REPORT OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL UNITS AND MEASUREMENTS (ICRU)

1959

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*In press.

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Report of the International Commission on Radiological Units and Measurements (ICRU)
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Preface

A. Scope

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 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 units and concepts.

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. There is still a need for the quantity "exposure dose", however, and although the use of the term "dose" in both these quantities could lead to confusion, this can be avoided by giving each its full name. The Commission therefore recommends no change in the names of these quantities at the present time.

Detailed discussions have centered about a complete review of all of the quantities and units involved in radiation measurements. While a report on this subject will be issued in the near future, it is not proposed to make any major change in the position adopted in the 1956 report as described above.

Hitherto the definitions and recommendations of the ICRU have been made with little explanation of the philosophy on which they were based. The Commission recognizes that this neglect has given rise to confusion in the past, and in the present report has continued to develop the necessary technical information to facilitate the interpretation and application of the recommendations. This section of the report represents a considerable enlargement over the 1956 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. The objectives of this program and factors involved in achieving the requisite accuracy have been outlined in section 5.

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 measurement, 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 effect 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 Geneva in September 1958 following the second International Conference on the Peaceful Uses of Atomic Energy. At the opening session, the Commission agreed upon a general position with regard to programs on standards and dosimetry.* This is given in Appendix IV.

The second day of this meeting was held jointly with representatives from the World Health Organization (WHO), United Nations Educational, Scientific and Cultural Organization (UNESCO), International Atomic Energy Agency (IAEA), International Bureau of Weights and Measures (IBWM), Joint Commission on Applied Radioactivity (JCAR), International Union of Pure and Applied Chemistry (IUPAC), International Union of Pure and Applied Physics (IUPAP), International Organization for Standardization (ISO), and International Labor Organization (ILO).

The 1956 report called attention to some basic problems in the establishment of radiation quantities and units. These were the subject of major discussion 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 the scrapping of 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.

It was therefore decided to setup a small ad hoc committee for the purpose of examining this problem critically and in detail. The ad hoc committee has been working actively on the problem and held a two-week meeting in Geneva in April 1960 to resolve or reach a compromise on any differences between the members. It is contemplated that the report of the ad hoc committee will be published in the open literature in order to invite comment and discussion before receiving further consideration and possible adoption by the ICRU.

Meetings of the Main Commission, Committees, and Subcommittees were held in the days preceding the 9th International Congress of Radiology in Munich in July 1959. These meetings were mainly for the purpose of making final editorial changes in the 1958 draft report.

Two joint committees have been established between the ICRU and the ICRP. One, Methods and Instruments for Radiation Protection; this will be under the ICRU Chairmanship. The other committee, dealing with values for RBE, is under the ICRP Chairmanship.

As a result of a request from the United Nations Scientific Committee on the Effect of Atomic Radiations, the ICRU and the ICRP agreed to undertake joint studies on problems relating to the dose administered in the course of medical procedures. This program involved several meetings of selected members from each of the two commissions.

The first study resulted in a joint report entitled, "EXPOSURE OF MAN TO IONIZING RADIATION ARISING FROM MEDICAL PROCEDURES, an Inquiry into Methods of Evaluation." This was published in PHYSICS IN MEDICINE AND BIOLOGY, Vol. **2**, p. 107, October 1957.

The second study dealing with the medical and physical parameters in clinical dosimetry is underway at this time.

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 report 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 not been seriously debated by the Commission, as yet, but is being suggested for future consideration.

D. Operating Funds

Throughout most of its existence, the ICRU has operated on a purely voluntary basis. Costs of traveling, secretarial services, mailing, etc., have been handled by the various individuals involved or by their parent organizations. It has become increasingly obvious that this method of financing cannot continue indefinitely, particularly with the stepped-up activity of the Commission during the past few years.

At the close of the Congress in 1956, the Commission received from the International Society of Radiology (ISR) the sum of $525 to assist in defraying expenses during the following 3-year period. In 1959 the contribution from the ISR was increased to $3,000.

Substantial assistance for the work of the Commission has been received from the Rockefeller Foundation, which provided the sum of approximately $7,000 for the purpose of holding the meet-

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*ICRU Position Plan—this program was reviewed at the 1959 meeting of the ICRU in Munich at which time the progress of several of the subcommittees was noted. It was agreed that this work was of sufficient importance to warrant its continuation and expansion and the program is presently going forward.

ing in September 1958. The Rockefeller Foundation made an additional grant of about $4,000 for defraying most of the expenses of the Ad Hoc Committee meeting held in Geneva in April 1960.

Assistance by the World Health Organization has also been very substantial. The WHO has provided the Commission with meeting space, secretarial assistance, reproduction services, etc., for the 1958 and 1960 meetings. In addition, the WHO at the beginning of 1960 allocated the sum of $3,000 to the ICRU, for carrying forward its work. It is expected that this sum will be allocated annually, at least for the immediate future.

In connection with the Joint Study with the ICRP, the United Nations allocated the sum of $8,000 for the joint use of the two Commissions. This was handled for both Commissions by the ICRU. The sum of $10,000 has been granted by the United Nations for the joint study currently underway by the two Commissions. This is being handled by the ICRP.

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

While all of these contributions have been of great assistance to the Commission in its work, it should be pointed out that the total still falls far short of its estimated needs for the next several years. For this reason, endeavors are being made to seek additional funds from Foundations or from other sources having an interest in the Commission's activities. A valuable indirect contribution has been made by the National Bureau of Standards, where the secretariat resides. The Bureau has provided substantial secretarial services, reproduction services, and mailing costs to an extent which amounts to several thousand dollars a year.

The Commission notes with satisfaction that the World Health Organization and the International Atomic Energy Agency are interested in using some of their resources to hold conferences on current problems of radiation dosimetry and in support of research in this field. The Commission has never felt it to be within its scope to organize large scientific meetings or to sponsor research.

From time to time the Commission arranges for a meeting of one of its committees. These are meetings of experts who advise the Commission on the scientific aspects of matters which will become the subject of Commission recommendations. They are an integral part of the committee work which is carried on continuously and largely by an exchange of documents between members. The committee meetings thus form an important part of the Commission's activities for which annual support is essential.

E. Rules

The International Commission on Radiological 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 2 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.

F. Organization of the ICRU Subcommittees

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.

G. Composition of the ICRU

(a) 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	Great Britain.
H. O. Wyckoff, Secretary	United States.
A. Allisy	France.
R. H. Chamberlain	United States.
F. Ellis	Great Britain.
G. Failla	United States.
H. Holthusen	Germany.
H. E. Johns	Canada.
F. E. Leborgne	Uruguay.
W. J. Oosterkamp	Netherlands.
B. Rajewsky	Germany.
R. Sievert	Sweden.

(b) The membership of the Main Commission as confirmed during the International Congress of Radiology in Munich is as follows:

Lauriston S. Taylor, Chairman.	United States (National Bureau of Standards, Washington, D.C.).
L. H. Gray, Vice-chairman.	United Kingdom (British Empire Cancer Campaign Research Unit in Radiobiology, Mt. Vernon Hospital, Northwood, Middlesex).
H. O. Wyckoff, Secretary	United States (National Bureau of Standards Washington, D.C.).
A. Allisy	France (Ecole Normale Supérieure, Paris).
R. H. Chamberlain.	United States (University of Pennsylvania Hospital, Philadelphia, Pa.).
F. Ellis	United Kingdom (Churchill Hospital, Headington, Oxford).
H. E. Johns	Canada (Ontario Cancer Institute, Toronto).
W. J. Oosterkamp	Netherlands (Philips Research Laboratory, Eindhoven).
B. Rajewsky	Germany (Max Planck Institut fur Biophysik, Frankfurt).
K. K. Aglintzev	U.S.S.R. (Academy of Sciences, Leningrad).
H. Franz	Germany (Physikalisch-Technische Bundesanstalt).
H. H. Rossi	United States (Columbia University, New York, N.Y.).
M. Tubiana	France (Laboratoire Des Isotopes Et Betatron, Institut Gustave Roussy, Villejuif, Seine.)
H. Holthusen, Member Emeritus.	Germany (St. Georg Hospital, Hamburg).
R. Sievert, Senior Adviser.	Sweden (Institute of Radiophysics, Stockholm).
G. Failla, Senior Adviser.	United States (Argonne National Laboratory, Lemont, Ill.).

Members of Committee I.

W. E. Perry (Chairman), National Physical Laboratory, Teddington, Middlesex, England.
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N. G. Trott

Subcommittee 3. Techniques for measuring radioactivity in hospitals and biological laboratories, and equipment for clinical applications.
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Report of the International Commission on Radiological Units and Measurements (ICRU) 1959

(Revision of NBS Handbook 62)

Part 1. General Considerations [1]

1. Definitions of Quantities and Units [2]

1.1.[3] **Absorbed dose** of any ionizing radiation is the energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest.

Note.—(a) In the definition of absorbed dose the concept of "energy imparted to matter" refers to all of the energy which appears as ionization, excitation or changes of chemical bond energies during the period of observation and within the specified mass of material. Energy as here defined includes, for example, the energy of lattice displacements but not the energy associated with changes of rest mass (e.g., after neutron capture). If nuclear excitation occurs, the fraction of the energy released and which remains in one of the above forms within the specified mass, is counted at the time of decay.

(b) The quantity "absorbed dose" can apply to any material. Moreover, in a constant radiation field its magnitude will be different in different materials and hence the material should always be specified.

(c) Various methods may be used to determine the magnitude of the absorbed dose. Among these are:

(1) *Calorimetry* (see section 11). Absorbed dose may be determined from measurements of the temperature rise of the material during irradiation. This method has been used with good accuracy for absorbed dose rates above approximately 50 rads per minute. However, corrections of a few percent are required to account for endothermic and exothermic chemical reactions, crystal lattice displacements, etc.

(2) *Ionization* (see section 8). Absorbed dose may be determined from ionization measurements by means of the cavity ionization relationship. While such determinations are possible for a large range of dose rates, uncertainties in the value of some of the conversion factors limit the accuracy at present to the order of a few percent.

1.2. The unit of absorbed dose is the rad. 1 rad is 100 ergs/g.

1.3. **Integral absorbed dose** in a certain region is the energy imparted to matter by ionizing particles in that region.

1.4. The unit of integral absorbed dose is the gram rad. 1 gram rad is 100 ergs.

1.5.[3] **Absorbed dose rate** is the absorbed dose per unit time.

Note.—It should be pointed out that there are special situations when the absorbed dose rate should be expressed more explicitly. For example: when the absorbed dose rate is not constant during the time of irradiation, it may be desirable to specify also the instantaneous absorbed dose rate. It is recognized that the term "instantaneous" may not always be sufficiently explicit and that perhaps a statement should be added referring specifically to an absorbed dose rate of pulsed radiation averaged over a single pulse.

1.6. The unit of absorbed dose rate is the **rad per unit time.**

1.7.[3] **Exposure dose of X- or gamma radiation** at a certain place is a measure of the radiation that is based upon its ability to produce ionization.

Note.—(a) The translation of "exposure dose" into German is "Bestrahlungsdosis", into French "dose d'exposition", and into Spanish "dosis de exposición."

(b) Although the definition of exposure dose was purposely stated in loose terms, a more physically specific definition might be as follows: "the exposure dose is measured by the ion charge, ΔQ, of either sign produced in air by the secondary electrons, which are produced by X- or gamma radiation in a small mass, Δm, of air divided by Δm." Note that according to the above definition, ΔQ is *not* the charge measured in Δm. However, under electronic equilibrium conditions, the charge produced in Δm is approximately numeri-

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[1] The ICRU is presently studying all definitions used in its report (see preface). This study is to be completed before the next report.

[2] Symbols and nomenclature. There are numerous national and international bodies that have reached varying degrees of acceptance of the use of symbols and units for physical quantities. However, there is no universal acceptance of any one set of recommendations. It is suggested that each country modify the symbols used herein, in accordance with its own practices. Thus one may write: kev, keV, or Kev; ^{14}C or C^{14}; rad per unit time, rad per time, or rad divided by time; rad/sec, rad/s, or $rad \cdot s^{-1}$; etc. The most generally accepted system of symbols and units may be that contained in Document UIP 6 (1956) prepared by the International Union of Pure and Applied Physics. These are in fairly close agreement with the recommendations of the International Standardization Organization project ISO/TC 12, the Conférence Générale de Poids et Mesures, Union Internationale de Chimie Pure et Appliquée, and the International Electrotechnical Committee.

[3] See notes following this definition.

cally equal to ΔQ (see sections 1.8(a), 3.2.a, and 8.2.)

(c) The wording of this definition leaves open the possibility of later defining exposure dose for radiations other than X- or gamma rays.

1.8.[3] The unit of exposure dose of X- or gamma radiation is the **roentgen**(r). One roentgen is an exposure dose of X- or gamma radiation such that the associated corpuscular emission per 0.001293 g of air produces, in air, ions carrying 1 electrostatic unit of quantity of electricity of either sign.

NOTE.—(a) According to the definition, a dose of one roentgen is obtained at a point if the electrons generated in 0.001293 g of dry air at that point produce along their track 1 esu of ions of either sign. Accurate measurements in roentgens are not obtained by actually measuring these ions. Instead the concept of electronic equilibrium is used so that one can measure the ionization per 0.001293 g of air. According to this concept the ionization produced outside of a small mass, m, by high-speed electrons generated inside of m is compensated by ionization produced inside m by electrons generated outside of m (Fano, 1954).

(b) The corpuscular emission shall not include contributions due to secondary X- or gamma radiation produced in the quantity of air in which the corpuscular (electron) emission referred to is generated. This follows from a consideration of the actual definition of the roentgen and the concept of electronic equilibrium.

(c) It becomes increasingly difficult (because of electronic equilibrium limitations) to determine the exposure dose in roentgens as the quantum energy of the X- or gamma radiation approaches very high values. For practical purposes, 3 Mev is sometimes arbitrarily regarded as the useful upper limit of the energy range over which the roentgen should be used.

1.9.[3] **Exposure dose rate** is the exposure dose per unit time.

NOTE.—(a) Exposure dose rate can be used to specify a field of irradiation or the output from an X- or gamma-radiation source up to 3 Mev.

(b) For quantum energies above 3 Mev, the ICRU at present is not in a position to make a firm recommendation on the specification of output from a radiation source (see section 3.4). One may use either intensity, or the absorbed dose rate at the peak of the buildup curve in a phantom under specified conditions; the latter may be derived from ionization measurements.

(c) It should be pointed out that there are special situations when the exposure dose rate should be expressed more explicitly. For example, when the exposure dose rate is not constant during the time of irradiation, it may be desirable to specify also the instantaneous exposure dose rate. It is recognized that the term "instantaneous" may not always be sufficiently explicit and that perhaps a statement should be added referring specifically to an exposure dose rate of pulsed radiation averaged over a single pulse.

1.10. The unit of exposure dose rate is the **roentgen per unit time.**

1.11. **Intensity of radiation** (radiant energy flux density) at a given place is the energy per unit time entering a small sphere centered at that place per unit cross-sectional area of the sphere.

1.12. The unit of intensity of radiation may be **erg per square centimeter second,** or **watt per square centimeter.**

1.13. The unit of quantity of radioactive material, evaluated according to its radioactivity, is the **curie** (c). One curie is a quantity of a radioactive nuclide in which the number of disintegrations per second is 3.700×10^{10}.

1.14. **Specific gamma-ray emission** (specific gamma-ray output) of a radioactive nuclide is the exposure dose rate produced by the unfiltered gamma rays from a point source of a defined quantity of that nuclide at a defined distance.

1.15. The unit of specific gamma-ray emission is the **roentgen per millicurie hour** (r/mch) **at 1 cm.**

1.16. **Linear energy transfer** (LET) is the linear-rate of loss of energy (locally absorbed) by an ionizing particle traversing a material medium.

1.17.[3] Linear energy transfer may be conveniently expressed in **kilo electron volts per micron** (kev/μ).

1.18.[3] **Mass stopping power** is the loss of energy per unit mass per unit area by an ionizing particle traversing a material medium.

Note on Linear energy transfer—Mass stopping power.—Except for the factor of density, the quantities LET and mass stopping power are very similar, but the former emphasizes the process of energy transfer, while the latter emphasizes the effect on the particle. In both quantities the energy lost in the production of bremsstrahlung is to be neglected for the present purposes. However, in the case of stopping power, all energy losses resulting in the production of charged particles are commonly included, while in the case of LET, such losses are only included if they are in the vicinity of the primary charged particle track.

1.19. Mass stopping power may be conveniently expressed in **kilo electron volts per milligram per square centimeter** (kev cm^2/mg).

2. RBE (relative biological effectiveness), a recognized symbol

Relative biological effectiveness was first used (Failla and Henshaw, 1931) to compare the biological effect. The results of such experiments usually are given now in terms of the ratio of absorbed doses. The value so obtained depends not only on the type and degree of biological damage but also upon many subsidiary variables such as the absorbed dose rate, the dose fractionation, the oxygen pressure, the pH and the temperature. It may be computed from experimental data obtained with the same or different kinds of radiation.

The U.S. National Committee on Radiation Protection recommended (NCRP Handbook 59, 1954) the use of the RBE for the field of radiation protection in order to provide a mechanism for the addition of absorbed doses of different kinds of radiation. For this application it has become necessary to assign to different types of radiation certain agreed factors (RBE), so that the "effective" dose may be computed. These factors take into account the "critical organs", some RBE values for certain effects on these critical organs, and any other relevant considerations. It would be advisable to distinguish between such agreed RBE factors and the experimental values of RBE determined as described above but agreement has not yet been obtained for this.[4] The procedure to be followed when expressing the exposure of persons to radiation is as follows:

The absorbed dose, D (in rads), of any radiation must be multiplied by an agreed factor, RBE, whose values for different radiations are laid down by the ICRP. This product, called the RBE dose, is expressed in rems where

$$\text{RBE dose (in rems)} = (\text{RBE})(\text{D}) \qquad (2.1)$$

In the case of mixed radiations the total RBE dose is assumed to be equal to the sum of the products of the absorbed dose of each radiation and its RBE.

$$\text{RBE dose (in rems)} = \Sigma[(\text{absorbed dose in rads})(\text{RBE})]. \qquad (2.2)$$

Part II. Clinical and Biological Applications

3. Considerations Regarding Absorbed Dose and Exposure Dose of X- or Gamma Radiation

The definitions in section 1 are given as critically as possible but it is hoped that the explanation in this section may make their meaning clearer and facilitate their clinical use.

3.1. *The concepts of absorbed dose and exposure dose.*

It seems desirable to make a clear distinction between the X- or gamma radiation to which a patient or object may be *exposed* and the energy absorbed from radiation as this energy is made locally available in a patient or object. The former, which is a measure of the radiation field, is called *exposure dose* and is to be measured in *roentgens*. The latter, which is determined by the radiation field and by the composition of the object irradiated, is called *absorbed dose* and is expressed in *rads*. *Absorbed dose* can be calculated from the exposure dose, in many cases, from other measurements of ionization, from luminescence, or from chemical effects; or measured directly calorimetrically. Because a biological effect of ionizing radiations presumably can result only from energy imparted to matter, it is evident that biological effect should be correlated more closely with the absorbed dose than with the exposure dose. The term "absorbed dose" is used to distinguish it from the various "doses" hitherto often expressed in roentgens. The term "exposure dose" replaces the term "dose" used in the 1953 recommendations to distinguish it more clearly from "absorbed dose"; in view of long-standing clinical practice and use in ICRP recommendations and national codes for radiation protection, it was considered expedient to retain the word "dose" in this expression.

That the absorbed dose gives a truer picture of the actual situation may be illustrated by the example that though bone may be exposed to the same exposure dose (expressed in roentgens) as adjacent soft tissue its absorbed dose may be more than three times as great (see section 3.5).

3.2. *The concept of the roentgen.*

In many instances the absorbed dose can be easily calculated from the measured value of exposure dose. Thus the roentgen used as a unit of X- or gamma-ray exposure will remain as a basis for practical dosimetry.

Certain features of the concept of the roentgen may be more clearly understood from the following:

(a) According to the definition of the roentgen, the ionization charge of either sign in esu produced in air by the electrons released by the photon beam per 0.001293 g of dry air (1 cm^3 at NTP) is equal to the number of roentgens. Actually the electrons released in a given volume of air usually produce part of their ionization outside of that volume. However, some of the electrons released outside of that volume also contribute to the ionization inside of the given volume. Thus the loss of ionization in the given volume may be partially or wholly compensated by the gain of ionization. As carried out with present techniques, a measurement in roentgens requires complete compensation. This compensation is obtained in a free-air ionization chamber by providing sufficient air in front of (upstream in the photon beam) and behind (downstream) the measuring volume for such compensation, and sufficient collection space to the sides for complete utilization of the energy of the electrons. In a cavity chamber, the compensation is provided by surrounding the measuring volume with a layer of solid (ideally air-equivalent) material. However, in both cases, the surrounding material also produces secondary (annihilation, characteristic, or scattered) photons which may themselves produce electrons that contribute to the ionization in the measuring volume.

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[4] Suggestions for the quantity to designate the particular factors include RBE$_P$ (for protection), RHF (Relative Hazard or Health Factor), *H* (Hazard or Health Factor) and *P* (Radiation Potency Factor). The ICRU and ICRP are considering the problem and it is hoped will have definite recommendations on this subject for the next report.

In addition, the surrounding material attenuates the photon beam. If the measuring instrument is used as an absolute device, a correction must be made for the secondary photon contributions and the attenuation; but if it is calibrated by an instrument of known sensitivity, such corrections are not required.

Figure 3.1 assists in describing the method of correcting for the photon attenuation when using a cavity chamber with an air equivalent wall. It is assumed that the air cavity is small enough so that the number of ions produced by electrons generated in the air by the photons is negligible as compared with the number of ions produced by the secondary electrons entering the cavity. If there are no high atomic number materials nearby which are irradiated by the photon beam, the ionization charge per 0.001293 g of air in the cavity initially increases with cavity wall thickness, t, along a path such as AB. However, if the beam irradiates nearby materials of high atomic number, the initial portion may follow a curve such as CB. In either case, this initial portion is followed by more or less of a plateau, BD, and a decrease, DE. The slope of the latter portion is controlled by the attenuation of the photon beam in the wall material.

(b) When *t* corresponds to the portion of figure 3.1 labeled DE, the photon attenuation correction depends upon the distance, t′, from the outer wall to the mean position at which electrons that produce ionization in the cavity are generated. For low-energy photons such electrons are produced very near the inner wall so that $t' \simeq t$ and the ionization should be corrected for a thickness t. In such cases one usually assumes that the attenuation of the photons for the thickness t has the same slope as that of DE (fig. 3.1). Thus the exposure dose is obtained by extrapolating DE to zero wall thickness to give the exposure dose R (fig. 3.1). For higher photon energies the difference in attenuation for a thickness t′ and a thickness t becomes appreciable. It amounts to about 0.5 percent for cobalt-60 and 0.2 percent for cesium gamma rays (Wyckoff, 1960. See Burlin, 1959 or Roesch, 1958 for formulation).

Other authors (Tubiana and Dutriex, 1958 and Barnard and Axton, 1959) have shown an alternate approach to the wall attenuation correction. If a value of t is chosen so that the ionization is a maximum (in portion BD of fig. 3.1) the proper attenuation is for this entire thickness.

It becomes increasingly difficult (because of electronic equilibrium limitations) to determine the exposure dose in roentgens as the quantum energy of the X- or gamma radiation approaches very high values. For practical purposes, 3 Mev is sometimes arbitrarily regarded as the useful upper limit of energy range over which the roentgen should be used.

(c). It is to be noted in the definition of the roentgen (sec. 1.8) that the *roentgen* is the unit of an *exposure dose* of X- or gamma radiation based upon ionization in air but that it *does not* express a quantity of ionization; it is *not* a unit of ionization, nor is it an absorbed dose in air.

3.3 *Quantitative relationship between absorbed dose and exposure dose.*

This subject will be more fully dealt with in section 8. Included here are some of the conclusions of that section for X- or gamma radiation.

Under conditions of electronic equilibrium, the absorbed dose, D, is proportional to the exposure dose, R, according to the relation $D=fR$. When the absorbing medium is air and the average energy required to produce an ion pair in air (W) is 34 ev, f has the value of 0.87_7 rad per roentgen for all radiation qualities. For other materials, f varies with the atomic composition of the material and with the radiation quality as more fully explained in section 8.

Some typical f-values in rads per roentgen are given in table 8.2 and figure 8.1.

In *transition zones* where electronic equilibrium conditions are not satisfied (i.e., between different tissues and on skin-, bone-, and cavity-surfaces), the absorbed dose may vary appreciably over distances comparable with the range of the secondary electrons. Routine recording of bone doses should include a statement of maximum and minimum absorbed doses in rads.

Two examples illustrating the difference between exposure dose and absorbed dose are given below.

Example 1.

Tube potential, 250 kv.
HVL, 1.0 mm Cu.
Field area, 100 cm^2.
Conversion factor (f) roentgens to rads: 0.945 in soft tissue, 1.76 in compact bone (see table 8.2).
Backscatter factor, 1.36 (Glasser et al., 1952).
Exposure dose at position of skin surface (without backscatter) $=1000$ roentgens.
Absorbed dose in soft tissue at position of skin surface $=1000\times0.945\times1.36=1290$ rads.
Absorbed dose at 80% isodose level in soft tissues not near bone $=0.80\times1000\times0.945\times1.36=1030$ rads.
Absorbed dose in cells in or near bone at 80% isodose level. Maximum absorbed dose $=0.80\times1000\times1.76\times1.36=1920$ rads; Minimum absorbed dose $=0.80\times1000\times0.945\times1.36=1030$ rads.

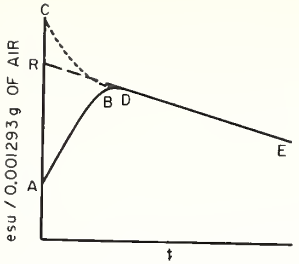

Figure 3.1. *The variation in ionization with wall thickness.*

The solid curve is the type of variation expected when no high atomic number materials are nearby and irradiated by the beam. The dotted curve may be obtained if there are high atomic number materials in the vicinity which are irradiated by the beam. The dashed curve is an extrapolation of DE which is used to obtain the exposure dose.

The dose varies from 1030 to 1920 rads adjacent to and within compact bone (see section 3.5 for details).

Example 2.

In the case of very energetic radiation (e.g., Cobalt-60) the corpuscular radiation produced by primary interaction is generated preferentially in the forward direction so that despite loss of direction by scattering, the absorbed dose builds up to a maximum below the surface in contrast with the exposure dose, which falls off in accordance with absorption and inverse square law. The situation therefore is as represented in figure 3.2.

3.4 *Routine measurement of the output of X- or gamma-ray sources.*

The output of the source or equipment that emits radiation with quantum energies up to 3 Mev or perhaps a little more may conveniently be specified by the *exposure dose rate* in roentgens per minute or per second at a fixed position, preferably measured in free air.

For quantum energies above 3 Mev the ICRU at present is not in a position to make firm recommendations as to a preferred method for measuring output of X- or gamma-ray sources. Roentgen measurements with ionization chambers must be made under electronic equilibrium conditions, which cannot, in principle, be fulfilled when the range of the secondary electrons becomes too great. Thus such measurements should only be applied to quantum energies below about 3 Mev.

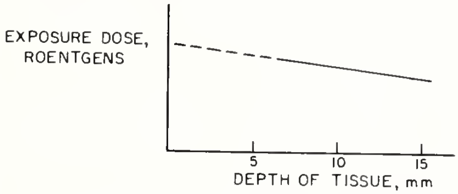

Figure 3.2a. *Variation of exposure dose with depth in tissue for cobalt–60 gamma radiation.*

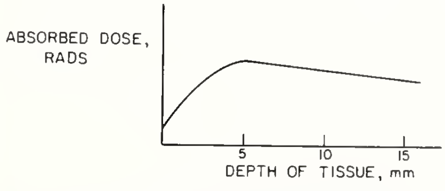

Figure 3.2b. *Variation of absorbed dose with depth in tissue for cobalt–60 gamma radiation.*

There are, however, several methods which have been employed successfully above 3 Mev:

(a) *Calorimetry.* The beam intensity in watts/cm^2 or erg/cm^2 sec may be measured by means of a total absorption calorimeter. Such a measurement is particularly useful for nuclear physics, but is usually not practical for routine measurements of output for radiotherapy.

(b) *Ionization chamber calibrated by comparison with a calorimeter.* In this method the convenience of a cavity ionization chamber is taken advantage of without the accompanying difficulty arising from its use as an absolute device where electronic equilibrium conditions cannot be attained. The cavity chamber should have walls which are thick enough to exclude secondary electrons originating outside (Laughlin and Genna, 1956; Pruitt and Domen, 1959). A lucite cube is frequently used to enclose a condenser ionization chamber (see for example NCRP–NBS Handbook, 55, 1954).

Such a chamber is located at a calorimetrically calibrated point in an X-ray beam of the quality for which the chamber will later be routinely used. The ionization in esu/cm^3 is observed, and a figure is thus established for esu/cm^3 in the cavity chamber per erg/cm^2 incident upon it. Subsequent routine measurements with the chamber can then be easily interpreted in terms of erg/cm^2 in the beam.

(c) *Absolute ionization measurement at the point of interest.* The above two methods, (a) and (b), are intended to measure the intensity of an X- or gamma-ray beam. For medical treatments one usually wants to know the absorbed dose at some point in tissue exposed to the beam. Knowing the intensity, and its spectral distribution, one may calculate the energy converted into kinetic energy of fast electrons at the point of interest, by summing the product of the flux at a given photon energy and the appropriate energy absorption coefficient ($_m\mu_{en}$) for tissue (see table 8.1). But because of the long range of these electrons, it is not a simple matter to compute the absorbed dose at the same point.

A more direct procedure for medical work makes use of the method outlined in section 8.4, case II. A cavity ionization chamber of nearly tissue-equivalent material and known volume is inserted into a phantom of the same material as its wall. The phantom should be of large enough area to more than cover the beam used therapeutically. The depth of the chamber in the phantom is usually adjusted until the maximum ionization is produced in the chamber. This is a relatively easy adjustment to make because the maximum is a broad one.

The absolute charge Q per unit volume is measured (in esu/cm^3 of air at 0° C and 760 mm

Hg) while the phantom is exposed to the X- or gamma-ray beam to be calibrated. Such an ionization measurement under a specified set of conditions constitutes a calibration of the beam. While it is not easily related to the beam intensity, it can be used straightforwardly to calculate the absorbed dose in the phantom material at the site of the chamber, by means of eq. 8.10.

3.5. *The special case of bone.*

(a). *General considerations.* The case of the boundary region between bone and soft tissue requires special consideration because of its importance in clinical radiotherapy at conventional peak potentials of 200 to 300 kv. The presence of bone in a treatment zone alters the distribution of dose in two ways. The macroscopic distribution is altered, with a reduction in the dose reaching soft tissues beyond the bone and, on a microscopical scale, the dose to soft tissues immediately adjacent to or enclosed within bone is raised by the action of secondary electrons arising from photoelectric absorption in the calcium and phosphorus of the bone mineral.

The anatomy of the situation is indicated in figures 3.3 to 3.5, and dose distributions are illustrated purely qualitatively in figures 3.6 and 3.7.

With 200 to 300 kv radiations, the dose enhancement produced by secondary electrons from compact bone will occur only within distances of the order of 100 microns or less from the bone surface. The tissues of interest in such a case are, for example, periosteal tissues at points B C D and corresponding endosteal sites B′ C′ D′ in figure 3.4. Within the bone, the tissues of interest such as E in figure 3.5a will be the soft tissue components of the Haversian systems: the osteocytes, the connective tissues lining the walls of the Haversian canals, and the contained blood vessels (see fig. 3.5b). In all these cases the absorbed dose depends upon the linear dimensions of the tissue inclusion relative to the range of the emitted photoelectrons.

The general features of the dose distribution in the region of compact bone are illustrated in figures 3.6 and 3.7. In figure 3.6 the exposure dose is seen to fall below that measured in a homogeneous medium and in figure 3.7 the absorbed dose is shown to rise within the bone itself. The condition at the interfaces between bone and soft tissue are not shown accurately to scale and the linear extent of the interface region is enlarged for the sake of clarity.

b. *Absorbed dose near an interface and within bone.* The absorbed dose in the compact bone itself is given by

$$D_{\text{mineral bone}} = fR \quad \text{rad} \tag{3.1}$$

(see eq 8.6) where R is the exposure dose at the site in question and f is the factor for compact bone for the photon energy (table 8.1) or averaged for the appropriate photon spectrum (table 8.2 and fig. 8.7). These figures indicate the values of the coefficient f averaged for some primary and scattered photon spectra within a water phantom and also the values of f measured directly by the method of equivalent gases (Allisy, 1958). This dose to compact bone, however, is not of direct clinical interest because dose levels sufficient to cause biological effects in the living soft tissue components of bone are unlikely to affect the purely mineral parts.

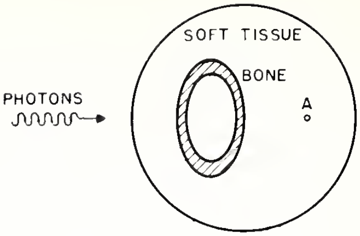

FIGURE 3.3. *Tissues of interest—point in soft tissue shielded by bone.*

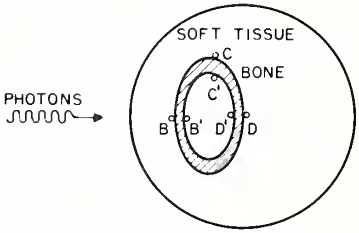

FIGURE 3.4. *Tissues of interest—points adjacent to bone.*

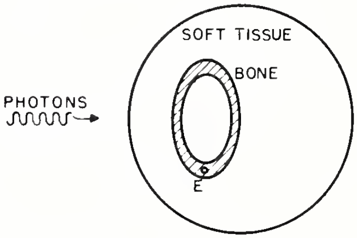

FIGURE 3.5a. *Tissues of interest—points within bone.*

In very small soft tissue inclusions the absorbed dose will approach that given by eq (3.1) because it will be produced mainly by secondary electrons generated in the compact bone. These electrons, however, lose energy at different rates in soft tissue and in bone, and eq (3.1) must be modified to include the ratio of the mass stopping powers for bone and tissue:

$$D_{\text{soft tissue in bone}} = \frac{fR}{(S_m)_{\text{bone}}/(S_m)_{\text{tissue}}} = FR \text{ rad} \tag{3.2}$$

where

$$F = \frac{f}{(S_m)_{\text{bone}}/(S_m)_{\text{tissue}}}$$

and

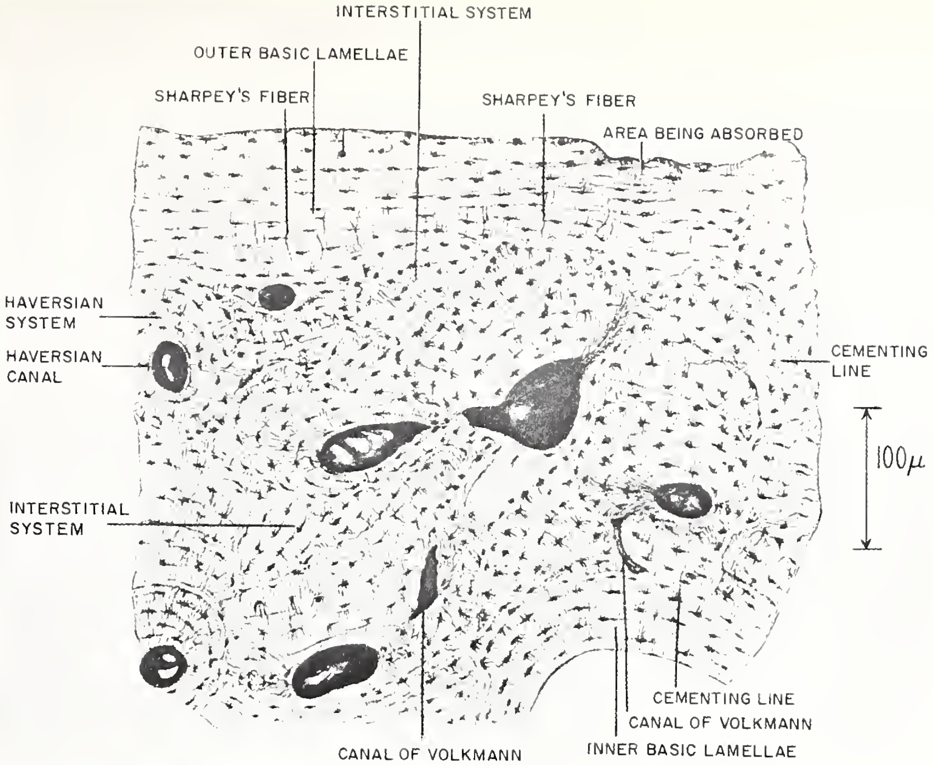

FIGURE 3.5b. *Photomicrograph of a ground portion of human metacarpal bone after staining with fuchsin and mounting in Canadian balsam.*

(After Schaffer in, Maximow and Bloom: Textbook of histology, 5th ed., W. B. Saunders Co., Philadelphia, Pa.)

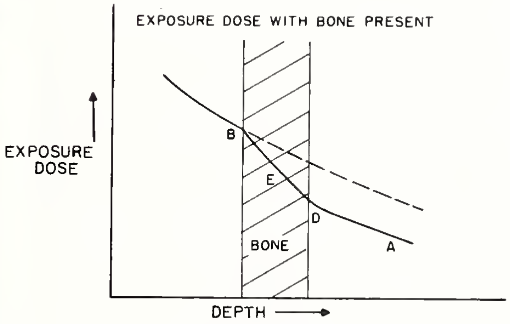

FIGURE 3.6. *Exposure dose with bone present (schematic).*

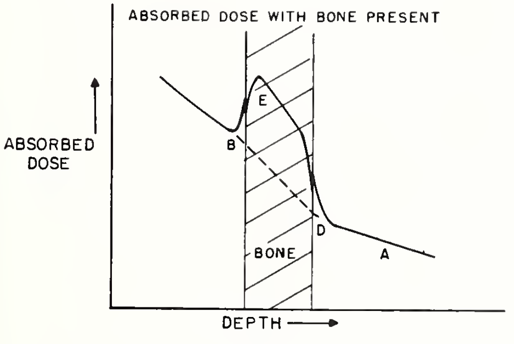

FIGURE 3.7. *Absorbed dose with bone present (schematic).*

$(S_m)_{\text{bone}}/(S_m)_{\text{tissue}}$ is the mass stopping power ratio (bone/tissue) for the average secondary electron energy. This approximation is justified because the $(\hat{s}_m)^{\text{bone}}_{\text{tissue}}$ is a slowly varying function of electron energy. The value of $(\hat{s}_m)^{\text{bone}}_{\text{tissue}}$ varies from 0.914 at an electron energy of 10 kev to 0.931 at 200 kev (calculated from table 8.3). Equation (3.2) holds only for soft tissue inclusions which are small compared with the secondary electron range; this means that for most of the energy range where the photoelectric effect in bone is important the tissue dimensions cannot be much more than 1μ (Spiers, 1951; see also fig. 3.10).

In the transition region near compact bone the absorbed dose varies with distance from the interface; approximate calculations for a plane slab of bone and other simplifying assumptions give values of the absorbed dose as illustrated in figure 3.8 (Spiers, 1951). At the interface itself, the value of F is approximately the mean of the values for muscle and bone derived from tables 8.1 and 8.2; at distances from the interface greater than the photoelectron ranges, the value of the factor falls to that for muscle. In figure 3.9 the distances are given at which the increased dose due to the proximity of the calcified layer is reduced to 10 percent of the value of the soft tissue dose.

If the tissue included within the compact bone has dimensions greater than about 1 μ, the dose received will be less than that given by eq (3.2)

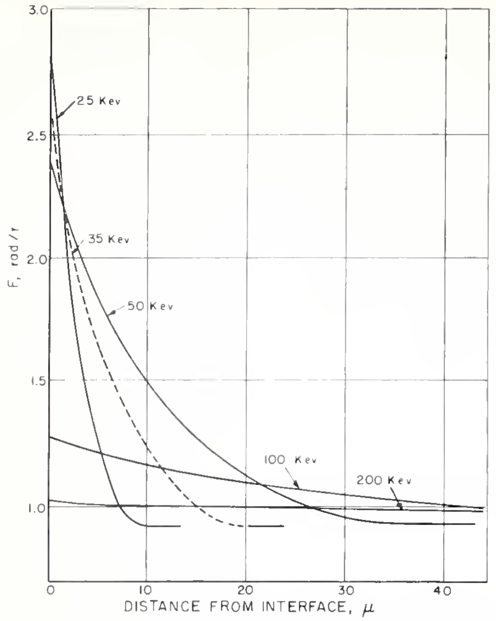

FIGURE 3.8. *Absorbed dose factor,* F, *in rad/r near bone.*

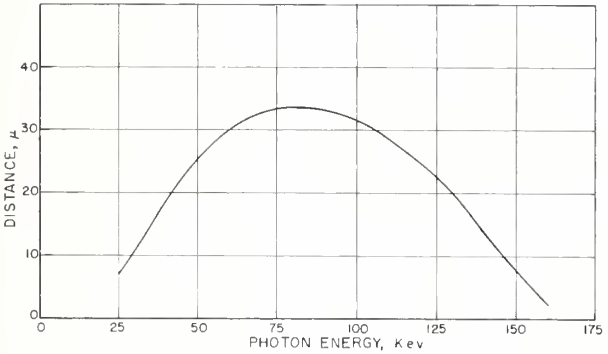

FIGURE 3.9. *Distance at which excess dose falls to 10 percent of soft-tissue dose.*

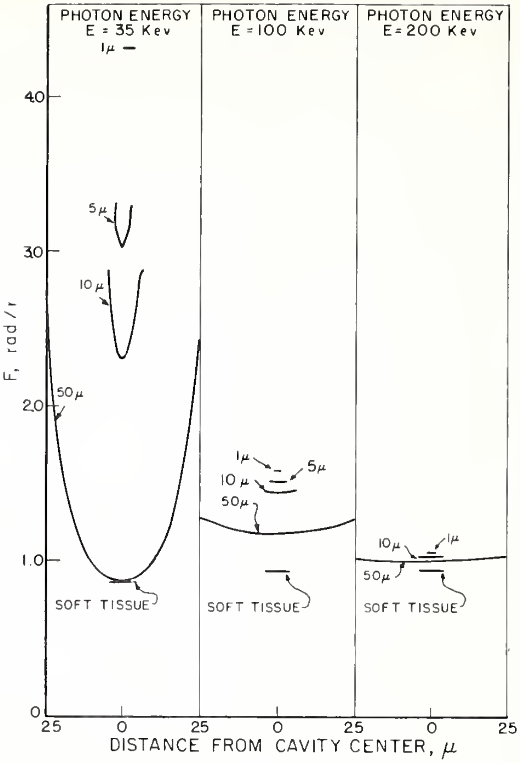

FIGURE 3.10. *Absorbed dose in tissue-filled cavities.*

and will vary from a maximum value at the edge of the cavity which contains the tissue to a minimum at the center. Variations in dose across tissue-filled cavities of different sizes for three photon energies are shown in figure 3.10. The 5-μ cavity may be taken as typical of one containing an osteocyte. The 50-μ and 100-μ cavities correspond to the dimensions of many Haversian canals. These calculations assume that the bone surrounding the tissue-filled cavities provides equilibrium numbers of photoelectrons; in the case of 200- to 250-kvp X-rays, electronic equilibrium is nearly complete if the thickness of bone is 100 μ.

In the Haversian systems two classes of tissue need to be distinguished: (1) Fixed tissues, and (2) blood cells in transport through the blood vessels. The latter group of cells, however, is only temporarily exposed to the higher dose rate in the canals and, because these cells mix with other blood cells, the absorbed dose they receive will not be significantly raised by the photoelectric effect in bone. The fixed tissues, on the other hand, will receive extra absorbed dose depending on their size and location. They include blood vessels having two types of wall: (1) Endothelial cells only—bone capillaries; and (2) mixed tissues—veins, arterioles, and arteries. It can be seen from figure 3.8 that the endothelium of the vessels in the second category will not receive more than about 10 percent excess absorbed dose at any quality of radiation, provided they lie centrally in canals or spaces greater than some 100-μ diameter. The endothelial cells of the bone capillaries in a 50-μ canal, however, could receive a minimum dose of about 140 rads for every 100 rads received in soft tissue remote from bone, when irradiated by 200- to 250-kvp X-rays. At 80 kvp (mean photon energy 35 kev), the absorbed dose to the endothelial cells could be as high as 280 rads (fig. 3.10).

It is clearly difficult to specify exactly the absorbed dose to the soft tissues in bone because of their variability in size and location. Nevertheless, a useful indication of the order of dose can be given, as in table 3.1, by calculating the absorbed dose for three types of tissue: (1) Average absorbed dose to an osteocyte of 5-μ diameter; (2) the mean of the average absorbed doses to two sizes of Haversian canal, one 10-μ and the other 50-μ in diameter; and (3) the average absorbed dose to tissues assumed to be 10-μ thick lining a Haversian canal of 50-μ diameter.

The osteocyte represents the tissue component which in all probability receives the highest absorbed dose, the mean absorbed dose in (2) is intended as an approximate measure of the average throughout the soft tissue inclusions in bone, excluding the marrow, (Spiers 1951), and the dose to the tissues lining a Haversian canal is given as a possibly significant clinical point.

(c). *Absorbed dose in bone marrow.* The spaces in trabecular bone which contain red marrow are, on the average, considerably larger than the small tissue-filled spaces in the Haversian systems with the result that the mean marrow dose is probably not greatly raised by the photo-electrons from the mineral bone. Engström et al. (1958) give distributions of the sizes of bone lamellae in human vertebrae and ribs as ranging from 60 to 100 μ with the dimensions of the marrow spaces ranging from 250 to 950 μ in vertebrae and 100 to 700 μ in ribs. Robertson and Godwin (1954), studying vertebrae and ribs, give a range of 20 to 250 μ for bone lamellae and 200 to 700 μ for the marrow spaces. Representative mean dimensions might be therefore 100 μ for bone lamellae and 400 μ for the marrow interspaces. Regarding trabecular bone as a formalized lattice as in figure 3.11, the marrow cells filling the spaces will receive a higher marrow dose depending on the size of the marrow space. The mean excess marrow dose, for a range of marrow spaces, has been calculated to vary with photon energy approximately as in figure 3.12. For 400-μ interspaces, the mean excess marrow doses are respectively 5, 13, 7, and 1 percent of the soft tissue dose at photon energies of 25, 50, 100, and 200 kev. Except at the highest energy, where the photoelectric effect is in any case small, the bone trabeculae will be mostly of sufficient thickness to provide electron equilibrium.

(d). *Further considerations and limitations of present analysis.* Tumors of bone have the same atomic composition as soft tissues and where macroscopic tumors have replaced calcified bone the absorbed dose will be the same as for soft tissue remote from bone. Where small extensions of tumor invade the bone structure, or islands of tumor develop in bone by metastatic spread, the absorbed dose will be governed by the same considerations of tissue size and electron range as in the calculations already given.

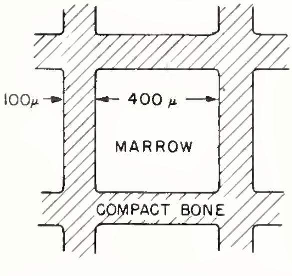

FIGURE 3.11. *Model and dimensions of lamellae and marrow spaces in spongy bone.*

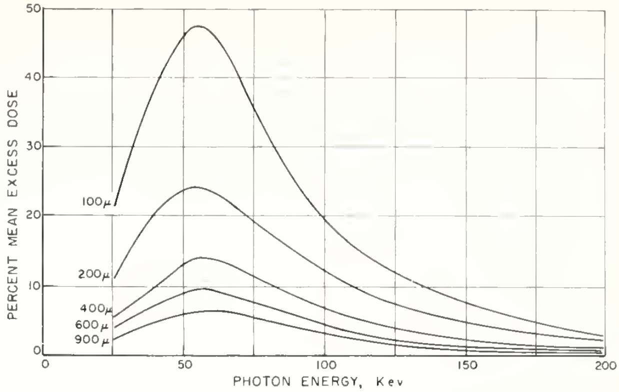

FIGURE 3.12. *Dose excess over soft-tissue dose averaged over marrow cavities.*

The evaluation of absorbed dose in bone has been made on the basis of geometrical models and simplifying assumptions such as single photon energies and isotropic emission of secondary electrons. More elaborate calculations have been made in special cases, using complete photon spectra and energy distributions of secondary electrons (Woodward and Spiers, 1953). Further refinements are desirable particularly in relation to the degradation of the primary photon energy spectrum with depth in tissue. Although the change of quality with depth makes little difference to the rad/roentgen conversion factors in soft tissue, in bone the effect can be considerable. For example, the factor $f=1.36$ for "average soft tissue" in bone (table 3.1) irradiated with X-rays of photon energy 100 kev without scatter (as for a finger), would change to 1.67 if a 10×10-cm beam were degraded by backscatter at 2 cm (e.g., a rib) and would increase to 1.87 if the same primary beam were directed at a bone 10 cm deep. (Data from Greening and Wilson, 1951.) Other data on energy degradation of the spectrum are now becoming available (see section 8.9(a)).

TABLE 3.1.—*Mean doses to soft tissue components of bone*

Mean dose factor F=rad/r			
Photon energy	Osteocyte 5-μ diam.	"Average soft tissue" in bone	10-μ lining of 50-μ Haversian Canal
kev			
25	2.80 (3.49)[a]	1.73 (1.94)[a]	1.50
35	3.12 (3.75)[a]	2.05 (2.42)[a]	1.76
50	3.25 (3.65)[a]	2.27 (2.71)[a]	1.89
75	2.40 (2.55)[a]	1.85 (2.10)[a]	1.60
100	1.52 (1.57)[a]	1.36 (1.47)[a]	1.26
200	1.05 (1.06)[a]	1.03_5 (1.05)[a]	1.02

[a] Kononenko (1957) has derived analytical expressions for the dose in cavities bounded by plane, spherical and cylindrical surfaces in a medium containing α-emitters. The solution for the case of the plane boundary is the same as that in which the numerical data in this section are based, including the approximate solution for cylindrical geometry to the photo-electric component of the X-ray dose. No solution was given which would enable the dose to the lining of the Haversian canal to the derived, but approximate calculation suggests that the differences between the "plane" and "cylindrical" geometries would be rather less than those in column 3.

Finally, consideration of the case of high-energy radiation is required. At high-photon energies the pair-production process results in greater absorbed dose in compact bone than in soft tissues. The magnitude of the effect can be estimated, for example, for a photon energy of 10 Mev at which the energy absorption per gram in compact bone is about 15 percent greater than in soft tissue and the positron and electron ranges are some 2 to 3 cm in soft tissue. For equilibrium the bone would have to be more than 1-cm thick and the tissues adjacent to it would then receive an additional absorbed dose of the order of 15 percent. In many cases the bone would be insufficiently thick to give this equilibrium dose, as bony structures represent only a part of the thickness of the anatomical bone; in such cases an average bony composition and density would have to be considered and the additional absorbed dose to adjacent tissues would be in fact less than 15 percent.

(e) *Shielding effect* (at distances greater than the range of secondary electrons from bone).

When bone is present in soft tissue between the radiation source and some position A, the dose rate at A is lower than if the bone is replaced by soft tissue. It is convenient to consider this lowering in terms of the relative reduction, $(D_1-D_2)/D_1$, where D_2 is the dose rate when the bone is in place and D_1 is the dose rate when the bone is replaced by soft tissue.

The dose rate at position A results from the radiation (a) that is transmitted directly to it from the source plus radiation, (b) which reaches A after one or more scatterings. For D_2 the latter radiation may or may not have passed through the bone while the former has passed through the bone.

The contribution to the relative reduction for radiation (a) depends on the relative attenuation of the radiation in the bone and in the same thickness of soft tissue. The relative attentuation of the radiation depends on the density of the bone compared to soft tissue and for low and very, very high (a few Mev) energy photons upon the chemical composition of the bone compared to soft tissue.

The density of bone varies. Spiers (1946) indicates a density of 1.85 for compact bone, Failla (1921), and Jacobson and Knauer (1956) found 1.15 for spongy bone. Haas and Sandberg (1957) found in 12 autopsy specimens of skull caps densities ranging from 1.53 to 1.92, 8 of them being between 1.6 and 1.69 and the average for the entire group being approximately 1.65. They found the same value of 1.65 in one specimen of the mandibular angle. Gest (1959) found 1.65 for a mandible, 1.45 for cervical-vertebrae (Atlas and 7th) and 1.3 for the wing of the iliac bone. Ellis and Jones (1957) found in adult post mortem vertebral bodies a density of 1.1 for the bulk of this bone, while the more anterior portion had an average density of about 1.4.

The problem is thus very complex and when high precision is required the attenuation coefficient must be calculated in each case considering: the characteristic of the bone, the energy of the primary X-rays and the characteristics of the scattered radiations which vary with area of the beam and the depth of the bone layer in the tissue. Furthermore, the change of the scatter contribution due to interposed bone has to be considered. A few examples of this type of calculation have appeared in the literature (Spiers, 1946; Meredith, 1951).

The actual magnitude of the shielding effect progressively decreases as the beam employed becomes harder. For high-energy X and γ radiation the variations of the mass attenuation coefficient with the composition of the bone and its depth in tissue are small and often can be neglected, and any change in the scatter contribution may also be neglected. For all field sizes the effective mass attenuation coefficient of bone is, to a first approximation with high-energy photons, equal to the effective mass attenuation coefficient of water or of soft tissue. Knowledge of the thickness of the bone and of its density makes possible the evaluation of the relative reduction of the dose beyond the bone.

At 200 kvp the relative reduction of the dose due to the interposition of bone varies between 10 percent and 30 percent for bones of thickness varying between 1 and 3 cm (table 3.2).

For a given quality of radiation the relative reduction of the dose varies slightly with the geometry as follows:

1. Beam area—The larger the field area, the smaller is the relative reduction. This is due to the fact that in any tissue the effective attenuation coefficient decreases when the field area increases. Furthermore, in the case of bone the relative scatter contribution originating in the bone layer increases with the size of the field. For instance, at 200 kvp, HVL 1.2 mm of Cu, and for a bone layer of 4 cm of thickness and 1.7 density, the relative reduction at a depth of 15 cm beyond the bone is 42 percent for a 3×3 cm field, 40 percent for a 5×5 cm field and, for a HVL of 2.0 mm of Cu, the relative reduction is, for the same-fields, 37 percent, 35 percent.

As this variation is so small, it may often be neglected (see A. Dutreix et al., 1959).

Table 3.2.—*Dose reduction due to bone shielding*

Reference	Bone density	Bone thickness	Depth of bone	Tube potential	HVL	Field area	Relative dose reduction
	g/cm³	*cm*	*cm*	*kvp*	*mm Cu*	*cm²*	%
Haas and Sandberg	1.65	1	0	400	2.12	48	11.4
Jones	1.65	0.9	0	-------	2	100	10
Spiers	1.85	3	6	200	1.5	100	27
Institüt Gustave-Roussy (Dutreix)	1.7	4	0	220	2	100	33

2. Bone depth—there is no variation with the depth of the bone in the tissue at 200 kv, but a small variation occurs at lower energies.

3. Distance between bone and position A—The way the relative reduction of the dose varies with the distance between the bone layer and position A is a function of the area of the field and the HVL, sometimes decreasing, sometimes increasing with the distance. For X-ray beams of HVL equal to or less than about 0.5 mm Cu, the relative reduction decreases with increasing distance between the bone layer and the point considered, the rate of decrease being augmented as the primary radiation becomes softer. For a 2.0 mm Cu HVL X-ray beam the opposite effect is observed (see fig. 3.13).

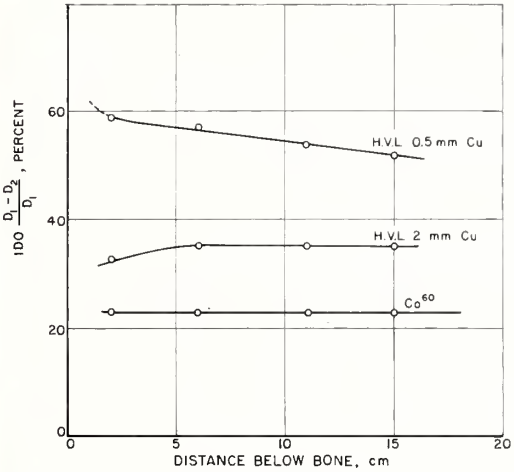

FIGURE 3.13. *Relative dose reduction due to a 4 cm thick bone at different distances beyond bone for various qualities of radiation.*

The processes responsible for these effects are very complex. These are:

(a) A filtration effect which changes the X-ray quality and increases the relative depth dose. This effect is small and negligible at 200 kvp. It is greater at lower energies.

(b) An increase of the forward scatter contribution originating in the bone layer due to the substitution of bone for water and the consequent increased density of electrons to act as scattering centers.

4. Specification of Radiation Dose

In applications of radiation to biological systems, the essential physical factors are related to: (1) Absorbed dose and its pattern (distribution in space); (2) type and energy of radiation; and (3) time distribution.

In both daily use and in reporting in the literature, it is important to consider how completely these values may be described for radiation therapy, radiobiology, diagnostic radiation procedures, and radiation safety or protection problems. In some instances only a small part of this information can be given. In other instances detailed expression of these values may seem unnecessary or impractical. Judgment should be exercised in deciding both the completeness and precision which are appropriate. The aim should always be to furnish as full detail as is pertinent so as to permit exact reproduction of the radiation technique and to allow others to analyze the radiation procedure in these terms. Information under the headings (1), (2), and (3) will now be dealt with in detail as applied to radiation therapy, radiobiology, and diagnostic radiology.

4.1. *Radiation therapy.*

A. *Beam therapy.*

(1) *Absorbed dose and its pattern.*

(a) The region of interest is usually the tumor or lesion which may be considered the target volume. The dimensions and volume of the region of interest should be specified. The maximum and minimum absorbed dose in rads in this region should be specified. In certain special cases the complete isodose distribution may be required in a plane through the tumor and in certain cases enough information given to enable the three dimensional nature of the distribution to be visualized. When high dose regions occur outside the target volume or when radiosensitive regions receive a significant dose, the distribution of radiation in these regions should be given. The factors involved (the depth dose tables, etc.) in estimating the absorbed dose should be specified. When corrections for bone, air, heterogeneity of the body have been used they should be specified.

(b) In some cases it may be impossible to specify absorbed dose values in rads. When this is the case, enough information regarding the technique of irradiation should be given to enable the radiation technique to be reproduced. Such information would include factors such as:

Target volume—dimensions, volume, and location.

Portals—number, dimensions, shape, location, or (for moving field) shape and dimensions of the beam at the axis of rotation or other appropriate level relative to the patient, angle of inclination relative to the axis of rotation, arc or pattern traversed.

Beam direction—source to surface distance, angle of beam, or (for moving field) distance from source to center of rotation or other pertinent point.

Dose—at some reference point.

Exposure dose—in roentgens at a specified place and extent of backscatter, if present, or surface absorbed dose, or other pertinent measured value at reference point.

Auxiliary devices—such as grids, wedge filters, bolus, etc.

(2) *Type and energy of radiation.*

The type of radiation (X-rays, gamma rays, electrons, protons or neutrons) should be specified. For X-rays the peak energy, the inherent and added filter, and the first half value layer should be given. For gamma rays the nuclide should be specified. For electrons or protons the energy should be given and for neutrons the mean energy.

(3) *Time distribution.*

The total number of sessions (treatment or group of treatments given during one patient visit) concerned with any one target volume and the total overall time over which the sessions are spread and their time distribution should be given. When unusually high or low exposure dose rates are used these should be specified.

B. *Surface, intracavitary, and interstitial therapy.*

These treatments are usually given with radioactive substances, but when X-ray machines are used, the information required is similar. A great variety of techniques may be used but the desirable information in reporting follows the same principles.

(1) *Absorbed dose quantities and pattern.*

(a) In most instances at least a maximum and minimum value for absorbed dose in the region of interest should be given. When possible, more complete expressions of pattern and range of dosage within the dimensions and volume of interest are desirable.

(b) The methods used for measuring or deriving dosage from computations and tabular values should be specified.

(c) When such dosage expressions are incomplete, the following data may be needed:

For discrete sources, the number, distribution, strength, irradiation time, and anatomical arrangement.

For extended sources, the features of volume, arrangement, irradiation time, and strength which contribute to dosage.

When appropriate, the methods of determining distribution and techniques which may influence it.

(2) *Type and energy of radiation.*

(a) The identity of the radioactive source. When pertinent, the types and energies of emissions, physical form, chemical form, and specific activity.

(b) The nature and thickness of material (including added filtration) traversed by the radiation before reaching the tissues.

(3) *Time distribution.*

(a) Half-life of the radioactive material.

(b) Number of applications, total duration of treatment, or pattern of temporary or permanent dosage.

C. *Systemic treatment with radioactive materials.*

Because of the difficulty in specifying a particular region as the tumor completeness of dose specification is difficult in this form of treatment.

(1) *Absorbed dose and its pattern.*

Information concerning the internal distribution of the isotope and the dose in rads delivered to various regions should be given. The methods by which this estimation is arrived at and the techniques of administration should be given.

(2) *Type and energy of radiation.*

The nuclide, the energies of its emissions and its specific activity should be given. When appropriate, the physical and chemical form should be stated.

(3) *Time distribution.*

The physical half-life and biological decay characteristics should be given.

4.2. *Radiobiology.*

In radiobiological studies care should be exercised in the design of the experiment to facilitate the statement of absorbed dose with greater precision than in radiotherapy. In studies with larger animal species the physical information may often be furnished in terms used in radiotherapy. For irradiation of very small animal systems, other methods of dose specifications may be required. For such systems the contribution to the total dose which arises from photo-electric emission from other material near the animals should not be overlooked.

4.3. *Diagnostic use of radiation.*

In most diagnostic procedures, it is unnecessary to specify absorbed dose values with the same detail as in section 4.1. However, in the development of new diagnostic techniques and in the comparison of these with standard ones, detailed specification of absorbed dose should be given. This should also be considered in techniques in which the doses are relatively large. For routine work, the following factors should be recorded:

(1) *Absorbed dose quantities and pattern.*

Location and thickness of body part examined.

Area and region of primary beam (with special reference to gonads in patients under 30 years).

Source to film or screen distance and any other useful geometric factors.

In fluoroscopy the exposure dose rate at some reference point for each kv used.

In radiography the exposure dose per ma-sec at same reference point for the kv used.

(2) *Type and energy of radiation.*

The kvp and total filtration and where possible the HVL.

(3) *Time distribution.*

In radiography the ma-sec per exposure.

In fluoroscopy the milliamperes and the cumulative time of exposure.

Cumulative number of examinations or exposures.

4.4. *Radiation safety and protection.*

In expressing dosage values in problems of radiation safety and protection, it is seldom possible to give complete information of absorbed dose quantities and pattern, but the concept of variations in this regard should be kept in mind. It should not be assumed that values derived from fragmentary monitoring methods apply to the whole body as well as to the gonads. The type and energy of radiation, which may be mixed, as well as its time distribution are highly pertinent. Whenever possible, the methods of measurement and the factors which are known should be recorded and reported. The appropriate sections of 4.1 and 4.3 are applicable for these purposes.

5. Report on an Inquiry on Current Errors in Clinical Dosimetry.

There are several methods of estimating the absorbed dose of radiation at any point in a patient, but whatever method is used the precision of the final figure is dependent on the accuracy of all the data used in arriving at the estimate. The necessary steps involve the calibration of dosimeters; calibration of output of radiation source; dose measurement; method of estimating body dose; planning of the treatment; and actual treatment of the patient.

In order to obtain information on the accuracy with which the dose to the patient can be determined, a questionnaire was submitted to hospital physicists and radiotherapists [5]. The data received are embodied in what follows:

5.1. *Calibration of clinical dosimeters.*

Reference chambers used as the standard in the clinic are calibrated against national standards. The dosimeters used routinely are then calibrated against the reference chambers.

(a) *Reference chambers* are *calibrated* against national standards to within ±2 percent. This *calibration* remains *constant* for many years *in most cases.* However, good quality commercial instruments, even though carefully handled, may show errors as follows:

(i) Sudden change following mechanical shock.

(ii) 5 to 10 percent change in 5 out of 13 cases, with no accident to account for these changes.

(iii) Variation for one quality but not for another, e.g., in one instance 1 percent for a cobalt-60 beam, but +6 percent for less energetic rays. In general, greater variations with softer radiation are observed but sometimes the reverse occurs.

(iv) Flaking of graphite from the inside of graphite coated chambers.

(v) Erosion of the electrode in one case.

(b) *Clinical Instruments* (*Chambers calibrated against reference chambers*).

(i) Dosimeters in clinical use may be calibrated with reference chambers, in which case the expected accuracy is less than that of the reference chamber.

(ii) More frequent use renders them more liable to variations in calibration.

(iii) Numerous commercial types are of different stability and precision.

(iv) Two instruments of the same make may show very different precisions.

(v) Commercial calibration of these instruments may be in error.

(vi) The variation of calibration at one quality may be different from that at another so that checking the precision of a chamber with a gamma-ray source may not be sufficient. In one instance, a change of 3–4 percent with 250 kvp X-rays occurred, but only a 1 percent change with 100 kvp X-rays.

(vii) In some clinical dosimeters a radium source is incorporated which checks the electrical circuit and instrumentation but not the chamber itself.

Recommendations.

Because most changes in calibrations are likely to arise through instrument use, the following recommendations are made to minimize errors in clinical dosimetry associated with calibration:

(i) A dosimeter called the reference chamber which is calibrated against the national standard, should be reserved solely for calibrating dosimeters used frequently in clinical dosimetry.

(ii) Such a reference chamber should be calibrated at least every two years [6] against the national standard for all qualities at which they are to be used. A constancy check should be made at regular intervals.

(iii) Dosimeters for regular clinical use should be calibrated with the reference chamber. Clinical dosimeters should have check calibration against a reference chamber for the range of qualities and output in which they are used, at least twice yearly.

(iv) A constancy check against a radioactive source kept for the purpose should be performed frequently on all thimble chambers.

(v) Graphite impregnated material is recommended in future new instruments, instead of graphite coating.

5.2. *Calibration of output from the radiation source.*

(a) *Methods of calibration.* Two methods are in general use for radiation energies up to 3 Mvp.

(i) Measurements in free air at a known fixed position relative to the source.

(ii) Measurement at the end of an applicator (cone).

Errors in these measurements, when made with a chamber calibrated for the energy and output for which it is used, may arise due to (a) scattered radiation from the chamber and other nearby objects; (b) inaccuracy in the position of the chamber relative to the radiation source; (c) stem leakage.

Recommendations.

(i) Exposure dose rate for each field size and applicator should be determined in the position of the central ray with suitable corrections for the relationship of the chambers to the applicator end.

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[5] The centers replying to this questionnaire are listed in section 5.11. In the text these are referred to by the numbers in parentheses.

[6] While this is desirable it is not feasible in many countries.

To minimize this correction the distance from the chamber to the applicator end should be small.

(ii) The geometrical arrangment of the chamber and source should be reproducible for constancy check measurements.

(iii) Measurements are recommended at various distances and with variable field sizes to obtain data to estimate errors due to scattered radiation and air attenuation.

(b) *Additional factors affecting the exposure dose rate estimation from an X-ray machine.*

These are:

(i) Fluctuations of the voltage and frequency of the power line. (The order of magnitude of these variations may differ considerably between installations and between different countries. In some countries line voltage fluctuations as high as ±15 percent have been reported.)

(ii) Changes in the temperature of the X-ray tube.

(iii) Fluctuations in target current.

(iv) Aging of X-ray tubes.

(v) Timing errors.

(vi) Shutter errors.

(vii) The build-up period to full voltage in tubes not having a shutter.

Timing and shutter errors are both of greater importance when short treatment times are used. In one case (24), the position of the tube head influenced the shutter error. The error in the case of tubes without shutters has been shown to be as great as 20r if the total exposure time is short (54), representing a possible error of 100 percent on a nominal dose of 20r and 10 percent on one of 200r.

In one center (60), when a recording monitor was initially installed on a 250 kvp therapy machine, it was found that in about 10 percent of the cases treated the error relative to the prescribed dose was greater than 10 percent. After the monitoring system called attention to this error, greater care was exercised and the error was reduced.

Recommendations.

(i) When line voltage fluctuations are large, continual adjustment of kv and current is necessary and allowance may be necessary for frequency variations and even wave-form changes.

(ii) A reliable monitoring system is desirable if input variations are considerable.

(iii) Output measurements are desirable under exactly reproducible conditions at intervals of 1 or 2 weeks.

(iv) Shutter time errors and errors resulting from build-up of the voltage may be eliminated in the calibration either by making two dose measurements for two different times or by the use of a dose rate meter. (The difference between the two dose readings provides the dose output for the time interval with no shutter or build-up error.)

5.3. *Monitoring chambers.*

(a) *Parallel plate monitoring chambers* (Thoraeus, 1956) in the beam between the X-ray tube and the filter or between the filter and the patient are frequently used. The stability of this type is reported as sometimes within ±2 percent (5 replies), but is frequently unreliable (8 replies). A variation of calibration of the chamber for different orientation of the tube head has been reported (24). In addition, when the monitor is not sealed its calibration depends upon the temperature of the air in the chamber; it may change considerably since the chamber is usually close to the tube.

(b) *Thimble chambers placed on the skin* introduce errors depending on the type of chamber, its dimensions and its position relative to the surface and the applicator. Variations of ±10 percent have been shown with some thimble-type chambers for small variation of the position of the chamber relative to the surface (50) and (52). This variation is smaller with chambers of smaller diameter.

(c) *Thimble chambers placed in a fixed position* in a socket in the end of the applicator show less variation in the readings than those placed on the skin.

Recommendations.

(i) Parallel-plate monitoring chambers should be sealed.

(ii) Frequent checks of the monitoring chamber are essential.

(iii) The monitoring chamber must be calibrated for each applicator and beam quality.

(iv) If thimble chambers are used as monitoring devices, they should not be placed on the skin surface at portal of entry but rather at some fixed position relative to the radiation source.

5.4. *Surface dose.*

(a) Surface dose is usually *determined* at the center of the field whether with or without an applicator (cone). The answers to the inquiry confirm the difficulties of making precise measurements of the surface dose, errors of ±10 percent being commonly encountered (50). Surface dose measurements are difficult to make because of the difficulty of obtaining a small enough detector for measuring the radiation. Errors could be due to geometrical factors if the center of the chamber is not in the plane of the surface, to scattered radiation coming through and from the ends of closed applicators or from the stem of the chamber, and to ionization in the stem (especially in small chambers). The differences in absorption of the ends of closed applicators also introduce the possibility of errors.

(b) The accuracy of delivery of prescribed surface dose has been checked on patients during treatment in 12 centers by skin measurement with small condenser chambers. Deviations from the expected dose were usually between 0 percent and 10 percent but it appeared that an error of 10 percent or more occurred in about 10 percent of the cases treated.

Recommendations.

(a) In general, surface dose measuremenss should not be attempted because of the technical difficulties and the surface doses should be computed from air dose measurements and *backscatter* factors. A backscatter factor is the ratio of dose determined with a chamber of very small dimensions half immersed in a suitable phantom to that obtained in air at the same position relative to the source.

Errors in the determination of the backscatter ratio with inadequate equipment are of the same order as in surface dose measurements. Computation introduces much smaller errors[7] provided an adequate set of data for backscatter factors is selected.

(b) If measurements are being carried out:

(i) The surface output for each applicator and field size should be measured.

(ii) The measuring chamber should be suitable and of small diameter (cavity not greater than 3 mm.).

(iii) For high accuracy, an extrapolation technique with a special chamber is necessary (or desirable).

(iv) Phantom measurements should be made under such conditions that maximum scatter is contributed by the phantom to the measuring chamber. For 250–300 kv this requires phantom material of 10 cm depth and extending for at least 10 cm outside of the field.

5.5. *Beam quality.*

Beam quality is characterized by peak kilovoltage (kvp) and half value layer (HVL). Errors in HVL measurement were estimated in 37 replies to lie between ±1 percent and ±10 percent (in the range 0.05 to 0.1 mm Copper). Errors are caused by (a) chamber quality dependence, (b) nonuniformity and impurity of the layers of absorbing material used to characterize the half value layer, (c) making the measurements without employing a monitoring chamber to compensate for variations in output, (d) improper arrangement of the chamber with respect to the filter and, (e) the use of too large a field.

Recommendations.

(i) A chamber with minimum quality dependence should be used for such measurements.

(ii) The purity and uniform thickness of the absorbing sheets must be assured.

(iii) The technique of measurement of the HVL is discussed in section 7.4.

5.6. *Depth dose determination.*

Depth dose is usually expressed as a percentage of surface dose and doses computed from depth dose data are therefore dependent on surface dose estimations. The variation between different sets of depth dose data is far from negligible (Boden and Cohen, 1952; Lindgren, 1957). Thus the estimation of a tumor dose may differ by 10 percent simply by using different central depth dose curves taken from the literature. These variations may be due to the fact that the reference point is taken at the surface, since it is in the surface that it is most difficult to make accurate measurements. The choice of a reference point at some depth such as 2 or 5 cm would considerably reduce the variation in the estimations of dose. As in surface dose measurements, errors may result from (i) inadequate size of phantom (ii) ionization chambers which are too large or show quality dependence. The choice of the most suitable published depth dose data is not easy.[8] Naturally, the data should be chosen to correspond to the physical factors such as half value layer, size and shape of field and focal skin distance which are actually used in treatment.

5.7. *Isodose curves.*

Isodose curves and depth dose curves are subject to similar inaccuracies. Published isodose curves for beams of the same kvp and HVL and with other factors similar in all respects may, nevertheless, exhibit considerable differences. Serious errors can result from the use of unsuitable isodose curves. Uniformity is considered desirable, should be possible, and could be attained, by adequate attention to the factors concerned in the measurements.[9] It would be a great advantage if agreed upon isodose curves could be used in all clinics.

5.8. *Clinical factors.*

The position and size of the target volume should be carefully defined. Thereafter errors may arise as follows:

(a) There may be dissimilarity between the body surface and that of the phantom used for measurement. Bolus is usually used to reduce the dissimilarity at low energies but not at high energies where "skin-sparing" is an important factor. Depth dose errors in the order of 30 percent may result from this practice (24) at the higher energies.

(b) There may be dissimilarity between the body volume and that of the phantom, the amount of scattering material being different in the two cases. With 200 kvp (HVL 1.0 mm Cu) and field areas ranging from 20 cm^2 to 100 cm^2, an overestimation of 13 percent in the tumor dose during a brain or neck irradiation (diameter 17 cm) has been shown to arise from the use of isodose curves obtained with a trunk-size phantom (51).

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[7] There are a number of sets of backscatter data in the literature which unfortunately do not agree too well. The ICRU is considering this problem and will present recommendations at a later date.

[8] The ICRU is considering this problem and will make recommendations at a later date.

[9] The ICRU is considering this problem and will make recommendations at a later date.

Recommendations.

(i) Bolus material should be used in the 200 to 300 kv range to reproduce the conditions in the region of interest under which the isodose curves were obtained.

(ii) Correction factors or special techniques should be used for high energy beams when bolus is not used (F. Ellis, 1959).

(c) The fields used may differ in shape from those for which isodose curves are available. With 200 kvp X-rays this could introduce errors in the order of 10 percent (51).

Recommendation.

This error may be avoided by suitable checks to determine a correction factor (D.E.A. Jones, 1959).

(d) Lack of adequate data for oblique and tangential beams may provide an additional uncertainty. The resulting error may be as high as 15 percent.

(e) Heterogeneity of body tissues introduces large errors that are often underestimated. When fixed field or rotation therapy is used in the treatment of tumors located in the thorax, dosage errors as large as 40 percent may be made if standard depth dose tables are used without corrections for the presence of air cavities and bone structures.

Direct measurements made with an ionization chamber placed in the esophagus during treatment with a field going through the lung shows that because of air cavities in the lungs the measured dose could be 20 to 40 percent higher than the value estimated with standard depth dose tables (50). During the treatment with a cobalt unit of a pelvis with a posterior field going through 9 cm of bone (sacrum), the dose could be 10 to 15 percent less than that estimated with standard depth dose tables. During the treatment of the brain through the skull, the reduction is in the order of 2 percent with cobalt but 10 percent with 200 kv (HVL 2 mm Cu). The reduction can be much higher, in the order of 30 percent, for a field going through the lower jaw.

Several methods have been described to make allowance for the variation due to a heterogenous medium. Correction factors have been used which are based on transmission dose measurement (Neumann and Wachsman, 1942; O'Connor, 1956; Fedoruk and Johns, 1957) or on information regarding the internal structure of the body derived from transverse tomograms in the plane of the axis of the fields for each patient (J. Dutreix et al., 1959); Further research is necessary.

(f) Incorrect positioning of the patient relative to the beam may cause more serious errors than all those so far discussed. Errors in F.S.D. and beam direction, and changes in the position of the patient from one session to the next and during a session, can all introduce serious errors.

Recommendation.

Discipline, training and suitable techniques should be adopted to minimize errors in positioning.

(g) Field positioning may be incorrect or the delineation of the target volume may be in error. The position of an internal organ such as bladder or cervix might be changed with regard to external landmarks as a result of change in position of the patient (50). Serious errors can arise in making field additions in such cases.

Recommendation.

A radiographic check should be made, whenever possible, of the field position relative to the volume to be treated.

(h) Errors may result from improper adjustment or operation of devices such as light beam devices, back pointers, timing devices, etc.

Recommendation.

The operation and performance of such instruments should be checked periodically.

5.9 *Special problems.*

(a) *Teletherapy with radioactive sources.*

(i) Such beams are most frequently calibrated at present using thimble ionization chambers o suitable design. Errors of up to ±15 percent ma result from the use of a chamber calibrated a 200 kv and used for a cobalt 60 or cesium 137 beam, even if its wall thickness has been suitabl varied to suit the types of radiation.

Recommendation.

Exposure must be measured with chamber calibrated for the energy of beam to be measured

(ii) Radioactive decay occurs exponentially and necessitates a suitable correction of exposure dose with time.[10] The more frequently this correction is made, the smaller the inaccuracy introduced by this factor.

Recommendation.

The necessary adjustment of exposure should be made at such intervals as to limit the error to ±1 percent. For cobalt 60 the error will not be larger than this if the adjustment is made every two months. If adjustment is made for cesium 137 every 12 months, the error will change from +1 percent to −1 percent during that period.

(iii) At a given distance from the source the *exposure* varies with the relative position and

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[10] The exponential decay correction may be complicated by the presence of more than one radionuclide in a source. For instance, according to Radioisotopes (1960), cesium 134 might be in the order of 15 percent of the total activity of a nominal cesium 137 source. As the half-life of cesium 134 is about 2.3 years and that of cesium 137 is about 30 years, the exposure dose must be measured frequently until the decay corresponds to the half-life of cesium 137.

aperture of the diaphragm. Depth dose and isodose curves are also dependent on source diameter, diaphragm size and position and source skin distance.

Recommendation.
Measurements of exposure, depth dose and isodose curves should be made under the conditions of treatment.

(b) *High energy X-rays (3–30 Mvp).* The ICRU is not yet in a position to recommend an instrument or method for determining the output of X- and gamma-ray source in this energy range (see section 3.4).

5.10. *Conclusion.*

Although the material for this survey came from radiotherapeutic centers of high standard and in most cases was based on measurements by hospital physicists, it was mentioned in many of the replies that the magnitude of the errors involved in each step of dose estimation had not previously been computed and surprise was expressed by some contributors at the result of their own estimations. It is important to realize the size of the errors even in the best conditions, and the necessity of work to reduce them. This survey emphasizes once more that to reduce these errors requires much time and a high standard of radiological physics technique. It has also shown that in many cases errors due to clinical factors, such as the heterogeneity of the body are often seriously underestimated. Errors in physical measurement may be overshadowed by clinical and human factors. Great care is required in each step of the planning and carrying out of treatment.

5.11. *Radiotherapeutic centers supplying data*[11]

EIRE

1. Radiotherapy Department, St. Luke's Hospital, Dublin.

GREAT BRITAIN

2. Radiotherapy Department, Derbyshire Royal Infirmary, Derby.
3. Radiotherapeutic Center, Addenbrooke's Hospital, Cambridge.
4. Radiotherapy Department, Royal Northern Hospital, London, N.7.
5. Radiotherapy Department, St. William's Hospital, Rochester.
6. Marie Curie Hospital, London, N. W.3.
7. Department of Radiotherapy, General Hospital, Northampton.
8. Radiotherapy Centre, St. Luke's Hospital, Guildford.
9. Radiotherapy Department, The London Hospital, London, E.1.
10. Stoke-on-Trent Regional Radium Centre.
11. Radiotherapy Department, Western General Hospital, Edinburgh.
12. Hogarth Radiotherapeutic Centre, General Hospital, Nottingham.
13. United Birmingham Hospitals.
14. Department of Radiotherapy, The Middlesex Hospital, London, W.1.
15. Physics Department, Westminster Hospital, London, S.W.1.
16. Radiotherapy Department, Norfolk and Norwich Hospital, Norwich.
17. Lincolnshire Radiotherapy Centre, Scunthorpe War Memorial Hospital.
18. Department of Radiotherapy, Royal Sussex County Hospital, Brighton.
19. Radiotherapy Department, Royal Berkshire Hospital, Reading.
20. Regional Physics Department, Glasgow.
21. Radiotherapy Department, University College Hospital, London, W.C.1.
22. Physics Department, Hammersmith Hospital, London, W.12.
23. Radiotherapy Department, Oldchurch Hospital, Romfort, Essex.
24. Department of Radiotherapy, The Churchill Hospital, Oxford.
25. Department of Radiotherapy, Charing Cross Hospital, London, S.W.1.
26. Department of Radiotherapy, Essex County Hospital, Colchester.
27. Radiotherapy Centre, Coventry and Warwickshire Hospital, Coventry.
28. Bradford Regional Radium Institute, Bradford.
29. Christie Hospital and Holt Radium Institute, Manchester.
30. Department of Radiotherapy, South Wales and Monmouth Hospital.
31. Physics Department, Lambeth Hospital, London, S.E.11.

FRANCE

50. Service de Radiothérapie, Institut Gustave Roussy, Villejuif (Seine).
51. Service de Radiothérapie, Centre anticancéreux de Lille, 1.
52. Service de Radiothérapie, Fondation Curie, Paris.
53. Service de Radiothérapie, Centre anticancéreux de Montpellier.
54. Service de Radiothérapie, Centre anticancéreux de Toulouse.
55. Service de Physique Medicale, Centre anticancéreux de Strasbourg.
56. Service de Radiothérapie, Fondation Bergonié, Bordeaux.

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[11] Numbering in each country denotes order in which information was received from each country.

U.S.A.

60. Department of Physics, Memorial Hospital, New York.

SWEDEN

65. Radiumhemmet, Karolinska Sjukhuset, Stockholm.
66. Radiotherapy Department, University Hospital, Lund.
67. Radiofysica laboratoriet, Sahlgrenska Sjukhuset, Gothenburg.

GERMANY

70. Krankenhaus St. Georg. Strahleninstitut, Hamburg.
71. Med. Strahleninstitut der Universität, Tubingen.
72. Strahlenklinik der Universität, Marburg (Lahn).

6. Considerations Regarding Integral Dose

The *integral absorbed dose* has been defined (section 1.3). Although its use in clinical and biological applications are of debatable value, it is desirable to encourage uniformity of conception and of methods of estimation.

6.1. If the whole body is the region over which the product of dose and mass is integrated, the integral should be referred to as the *total body integral dose* (Σ). Because of scattering of radiation, every part of the body receives some radiation when any part is purposefully irradiated, as in radiotherapy or radiodiagnosis, or when an individual is exposed to radiation unavoidably or accidentally. The estimation of body integral dose is difficult but was made by Grimmet (1942), and by Boag (1945) using multiple ionization chambers put together in the shape of a man. Ionization methods have also been used by Rossi et al. (1959), Reinsma (1959) and Bewley et al. (1959). Advances in chemical dosimetry (section 9) should facilitate the measurement and estimation of body integral dose for all types of radiation.

6.2. Integral dose (Σ_B) within the geometrical edges of the beam in beam therapy and radiodiagnosis is likely to be the most important part of Σ because of the high-dose rate and high dose. Its estimation is relatively simple and may be achieved graphically or by calculation (see appendix III).

6.3. Integral dose (Σ_R) outside the geometrical beam but within the body becomes a smaller proportion of Σ as the energy of radiation is increased.

$$\Sigma=\Sigma_B+\Sigma_R \tag{6.1}$$

Methods for the approximate evaluation of Σ have been devised which make possible an approximate rapid estimate of Σ_R, using calculated values of Σ_B in the above equation. A direct calculation of the part of Σ_R received in the tissues located between the edge of the beam and any chosen distance is possible (see appendix III). It is probable that this is the best way of estimating Σ_R because it facilitates the comparison of the biological importance of integral doses delivered at different dose rates.

6.4. Integral dose (Σ_T) within the target volume represents the integral dose due to radiation in the region to be treated as distinct from that in the rest of the body. In general it is desirable that Σ_T and Σ should differ as little as possible insofar as other conditions permit.

6.5. Methods are available (Mayneord, 1940, 1950) for estimating Σ for interstitial and intracavitary gamma-ray sources of radium or radioisotopes. In the case of β-emitting isotopes, the integral dose due to the β-rays is the product of the number of disintegrations and the mean β-ray energy emitted per distintegration expressed in appropriate units.

Part III. Physical Aspects of Dosimetry

7. Standards and Measurement of X- and Gamma-Ray Exposure Dose

7.1. *Standard instruments.*

Standard instruments are maintained in various national laboratories. These instruments are used to calibrate local standards or clinical instruments.

7.1.A. *Free-air chamber standards.* For X-rays of 500 kv or less the free-air chamber is usually the standard instrument. Two extensive reports on free-air chamber design for measurement of X-rays up to 500 kv have been published (Rajewsky et al., 1955; Wyckoff and Attix, 1957). Three papers (Ritz, 1959, 1960; Allisy et al., 1960) give additional information for the low-filtration 5 to 100 kv region.

One can estimate the accuracy with which a measurement with the free-air chamber can be made by estimating the accuracy of each of the factors involved. Table 7.1. gives the estimates included in the last ICRU report. The criterion of Wyckoff and Attix (1957) for item E agrees for field strengths above 100 or 200 volts cm^{-1} with the results for the similar theories of Boag (1956) and Hübner (1958) to within the accuracy estimate in table 7.1. Item F is the correction for ionization produced by X-rays scattered by air in the chamber. For X-rays generated by potentials of up to 500 kv, this correction is generally less than 1 percent. Ritz (1959) calls attention to the fact that this correction has been underestimated by Wyckoff and Attix (1957) by from 0.1 to 0.2 percent, but that with the new data the factor can be determined to within 0.1 percent.

TABLE 7.1.—*Experimental maximum error for quality range of 5 to 500 kv (constant potential), and filtrations of 1-mm Al to 3-mm Cu, respectively*

Experimental factor	Estimated maximum error (%)
A. Charge: (assumed measured by accurate potentiometer and capacitor, using electrometer as a null-detector)	±0.1
B. Air volume: (includes errors in diaphragm area, collecting plate alinement, and field distortion resulting from other causes)	±.3
C. Air density: (includes pressure and temperature measurements)	±.1
D. Humidity of the air: (includes measurement of the humidity and the effect of a slightly different value of the average energy per ion pair (W) for water vapor)	±.1
E. Saturation of ion collection	±.2
F. Scattering of X-rays by air in the chamber	±.1
G. Inadequacy of plate separation for electron path length	±.1
H. Photon attenuation in air between diaphragm and collector	±.2

The collecting region of the free-air chamber should be sufficiently wide that only a small fraction of the energy of the electrons produced by the X-rays in air is dissipated in the collecting and high-voltage electrodes. Item G indicates the maximum error which may arise in the determination of this small fraction.

From a measurement of the ionization in the collecting region, it is possible to compute the exposure dose at the center of this region[12] if the focal-spot-to-aperture and aperture-to-collecting-region distances are known. However, the free-air chamber is most often used to calibrate another instrument placed in the X-ray beam at the position previously occupied by the aperture of the free-air chamber. A computation of the exposure dose at the aperture from that at the center of the collecting region requires a knowledge of the X-ray absorption in air. Item H indicates the accuracy with which the absorption coefficient is known.

With the new data (Ritz, 1959; Allisy et al., 1960), it is possible to extend the region of applicability of table 7.1 down to 5 kv for moderate dose rates when the measurements are made at 0.5 m or more from the X-ray tube target.

During the 1956 ICRU meetings, the National Bureau of Standards (NBS) agreed to construct, calibrate, and make available to national laboratories a free-air chamber diaphragm, an air capacitor, and a small cavity chamber. Figures 7.1, 7.2, and 7.3 show cross sections of these instruments. A second set was constructed by the NBS for United Nations Educational, Scientific and Cultural Organization (UNESCO) and was also made available for loan. The diaphragm is usually the most critical factor in the free-air chamber. The various national laboratories may intercompare each other's diaphragms by comparing them with the diaphragm of the set. The capacitor permits the calibration of the laboratories' charge-compensating capacitor under d-c conditions. The three-terminal cavity chamber may be calibrated at the various national laboratories. A comparison of the calibrations is

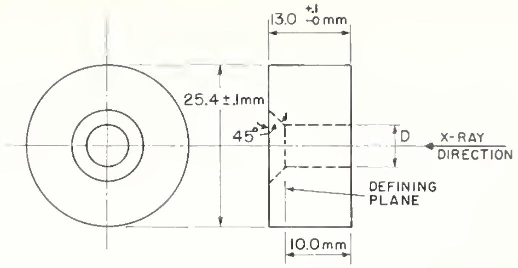

FIGURE 7.1. *Free-air chamber diaphragm indicating dimensions.*
The area of the hole of diameter, D, is supplied with the particular diaphragm (Wyckoff, 1957).

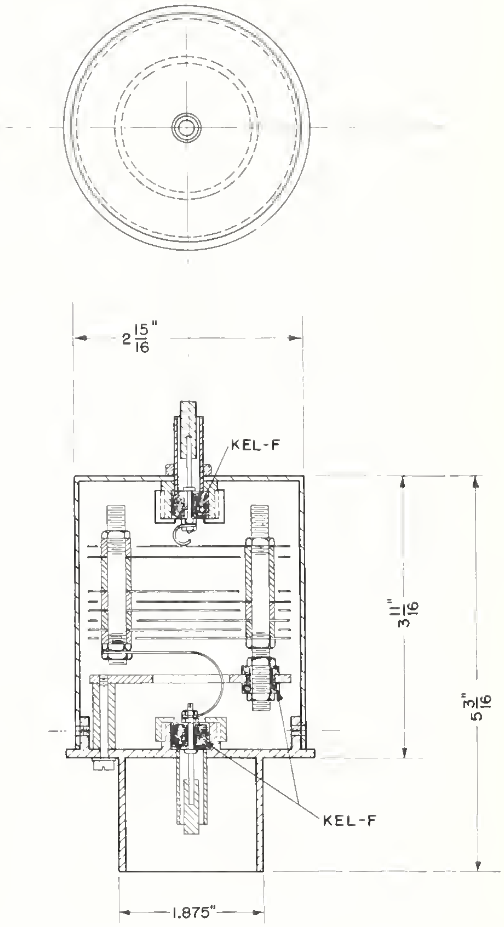

FIGURE 7.2. *Cross-sectional view of the capacitors supplied with the kit.*
These should not be disassembled as the capacitance might be changed thereby (Wyckoff, 1957).

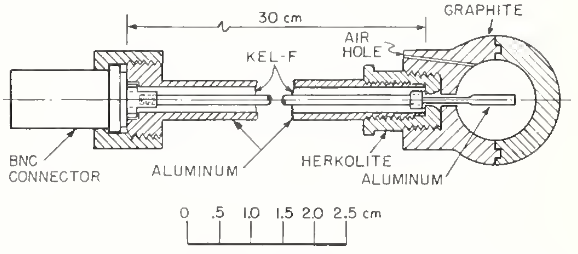

FIGURE 7.3. *Cross-sectional view of cavity chamber and stem (Wyckoff, 1957).*

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[12] For X-rays generated by potentials of 500 kv or less, the center of the collecting region is nearly the same as the mean position of origin of the electrons producing ionization in the collecting region.

a measure of agreement among the free-air chambers of these national laboratories. The long-term stability (~0.2 to 0.5 percent) of the cavity chamber indicates that radioactive sources for frequent checking the chamber sensitivity will not be necessary.

Table 7.2 shows the results thus far obtained with capacitors and diaphragms. These results indicate that the maximum errors listed in items A and B of table 7.1 are realistic.

Several intercomparisons of free-air chambers have been undertaken in the past four years. The results are listed in table 7.3. It would appear that there is generally an agreement to within about ±0.5 percent among all of these chambers. This is well within the maximum error indicated by the sum of all of the errors in table 7.1.

Table 7.4 lists the national standard free-air X-ray chambers of various countries according to the most recent information.

7.1.B. *Standards for the 0.5- to 3-Mev region.* Most roentgen measurements in this region at the present time are made with cavity ionization chambers. As the wall material is almost never truly air equivalent, a stopping power correction to the ionization current determination is necessary to compute the sensitivity. The uncertainty in this correction probably contributes the major uncertainty to such determinations. For this correction, the national laboratories of Canada and the United States of America, as a result of independent measurements (Attix and Ritz, 1957; Whyte, 1957) have agreed (Wyckoff, 1957; Garrett, 1958) on common values. The agreed values are based on the stopping power values obtained by Bakker and Segré (1951), an average excitation potential of 80.5 ev for air and the Sternheimer density correction. It should be noted that these values will give an exposure dose for cesium-137 and cobalt-60 gamma rays which differs by about 1 percent from that obtained by the use of the stopping power ratios given in table 8.6 of this report. Comparison of calibrations of a cobalt-60 and a cesium-137 gamma ray source by a free-air chamber and by a cavity chamber have now been completed (Wyckoff, 1960). Using a value of the relative stopping power, $s^{C}_{air}=1.003$ for cobalt-60 and 1.013 for cesium-137, the two instruments agree to within 1 percent.

Another major correction for such measurements, the photon attenuation in the wall, is considered in section 3.

7.1.C. *Intensity measurements above 3 Mev.* Above 3 Mev, where the roentgen is not defined, X- and gamma radiation is often described in terms of its intensity. Intensity measurements usually depend upon a calorimetric determination of the collimated X-ray beam power. However, such instruments are difficult to use and are not readily portable. Alternatively one may use a portable instrument whose response is proportional to the intensity.

TABLE 7.2a.—*Capacitor comparison*

	Capacitor A–5	Capacitor B–10	Method
	μμf	*μμf*	
NBS before shipment	$81.1_6 \pm 0.1$	988.6 ±0.5	Bridge.
NPL (Aston)	81.14	988.0	Balance with NBS capacitor.
	81.16±0.01	$988.3_5 \pm 0.1$	Bridge.
Rotterdam (Somerwil)	81.10	987.8	Balance with NBS capacitor.
PTB (Hübner)	81.1 ±0.2	988.4 ±0.3	Bridge.
	81.04	987.3	Ionometric comparison.
NBS after return	$81.1_5 \pm 0.1$	988.4 ±0.5	Bridge.
NBS before reshipment.	$81.0_8 \pm 0.1$	988.5 ±0.5	
NRC (Garrett)	$81.1_6 \pm 0.1$	$988.3_0 \pm 0.2$	Bridge.
NBS after retuurn	$81.0_9 \pm 0.1$	987.9 ±0.5	Bridge.

TABLE 7.2b.—*Comparison of diaphragm diameters*

	Diaphragm 7–B	Diaphragm 10–D	Method
	mm	*mm*	
NBS before shipment	6.989 ±0.001	10.008 ±0.001	Mechanical-optical.
NPL (Aston)	6.993	10.014	Ionometric.
Rotterdam (Somerwil)	6.996	10.018	Ionometric 4 mm Al HVL.
	----------	10.018	1 mm Cu HVL.
	----------	10.008	
PTB (Hübner)	$6.990_1 \pm 0.003$	$10.000_9 \pm 0.003$	Mechanical-optical.
	----------	10.009	Ionometric.
NBS after return	6.989 ±0.001	10.006 ±0.001	Mechanical-optical.

	Diaphragm 7–U	Diaphragm 10–U	Method
	mm	*mm*	
NBS before shipment	6.993±0.001	10.008	
NRC (Garrett)	6.994	10.000	Ionometric.
NBS after return	6.992±0.001	10.007	

TABLE 7.3.—*Results of intercomparisons*

Radiation quality		Ratio of exposure dose rates			
X-ray tube (constant potential)	HVL (approx.)	NPL[a]/NBS	IR[b]/NBS	IR[c]/MPIB	ENS[d]/NBS
kv	*mm Cu*				
52	0.05	----------	----------	0.999	----------
60	.08	1.005	0.995	----------	1.000
70	.10	----------	----------	.998	----------
75	.11	1.005	.992	----------	----------
100	.19	----------	.995	----------	1.000
100	.24	----------	----------	1.005	----------
110	.34	1.004	----------		----------
150	.61	1.003	.994	----------	1.004
150	.70	----------	----------	.995	----------
170	.83	----------	----------	.998	----------
200	1.2	1.003	.996	----------	0.999
250	2.1	1.005	.997	----------	----------

[a] G. H. Aston and F. H. Attix, 1956.
NPL=Nat'l Physical Lab.
Teddington, U.K.
NBS=Natl Bur. of Stds.
Washington, D.C., USA.

[b] R. Thoreaus and H. O. Wyckoff, 1956.
IR=Inst. of Radiophysics
Stockholm, Sweden
NBS=Nat'l. Bur. of Stds.
Washington, D.C., USA.

[c] R. Thoreaus et al, 1955.
IR=Inst. of Radiophysics
Stockholm, Sweden
MPIB=Max Planck Inst. für Biophik
Frankfurt, Germany.

[d] A. Allisy et al, 1957.
ENS=Ecole Normale Supérieure
Paris, France
NBS=Nat'l Bur. of Stds.
Washington, D.C., USA.

When the proportionality constant is determined by direct comparison with a calorimeter the instrument may be used to measure the beam intensity. A multilayered aluminum ionization chamber has been constructed at the United States National Bureau of Standards for this purpose and has been calibrated against a calorimeter (Pruitt and Doman 1959). The instrument and its calibration is being made available to other laboratories wishing to calibrate their own instruments.

7.2. *Accuracy of commercial exposure dose measuring instruments.*

7.2.A. *Introduction.* One of the important factors in the determination of the accuracy of the dose delivered to an irradiated object is the accuracy of the exposure dose determination. Although national standardization laboratories may agree to within about 0.5 percent, the accuracy of the clinical instruments is probably not as good. As there have been very few reports in the published literature on the accuracy of clinical instruments, it was decided to attempt an analysis of the calibration results for such instruments. A number of the major calibration centers in Canada, England, France, Netherlands, Sweden, U.S.A., and West Germany assisted in the accumulation of information. It was realized that this would probably be a biased sample, but that it might give some indication of the accuracy to be expected of the instruments now in use. Of those centers from which calibration information was requested, less than half were able to supply any large quantity of information. For this reason, the analyzed results reported here represent those from only a few percent of the instruments actually in use.

TABLE 7..4—*Primary X-ray standard chambers*

Standard No.	Country	Institute and custodian	Radiation range *kv*	Type	Compared with standard	Literature reference
1	Australia	Commonwealth X-ray and Radium Lab, Melbourne (D. J. Stevens).	40 to 200	Free air, parallel plate (Measuring system modified since reference).	--------	Eddy, 1937.
2a	Canada	National Research Council Ottawa (C. Garrett).	50 to 250	Free air, parallel plate	--------	
2b	do	do	50 to 250	do	--------	
2c	do	do	10 to 50	Free air, parallel plate (under construction.)	--------	
3a	Deutsch-Bundes-republik.	Physikalisch-Technische Bundesanstalt, Braunschweig (W. Hübner).	50 to 400	Free air, cylindrical.	4a	Rajewsky, B., et al. (1955). Thoreaus et al. (1955).
3b	do	do	10 to 120	Free air, parallel plate	--------	
3c	do	do	5 to 40	do	--------	
4a	do	Max Planck Institute für Biophysik, Frankfurt (B. Rajewsky).	30 to 400	Free air, cylindrical	3a, 12a	Rajewsky, B., et al. (1955).
4b	do	do	3 to 60	Free air, parallel plate	11, 12b	Bunde, E. et al. (1955).
4c	do	do	3 to 10	Total absorption "Wilhelmy chamber".	--------	Bunde, E. et al. (1954).
5	France	Laboratoire Central des Industries Electriques, Paris (P. Olmer).	50 to 250	Free air, parallel plate	--------	
6a	do	Laboratoire de Dosimétrie, Ecole Normale Supérieure, Paris (A. Allisy).	50 to 250	do	14a	Allisy et al. (1957).
6b	do	do	5 to 50	Free air, parallel plate (special design).	--------	Allisy et al. (1960).
7	Italy	Istituto Superiore di Sanita, Roma (M. Chiozzotto).	50 to 250	Free air, parallel plate	--------	Trabacchi, G. C. (1938).
8a	Japan	Electrotechnical Lab., Tokyo (M. Naito).	40 to 250	do	--------	Ito, T. and Ito, G. (1941).
8b	do	do	5 to 40	Free air, guard wire	--------	
9a	Netherlands	Radiotherapeutisch Inst. Rotterdam (A. Somerwil).	50 to 250	Free air, parallel plate	--------	deWaard et al. (1958), Somerwil (1959).
9b	do	do	10 to 50	Free air, nylon mesh	10	Somerwil, A. (1954, 1952); Oosterkamp, W. J. and Somerwil (1952).
9c	do	do	10 to 50	Free air, parallel plate	--------	
10	do	Philips Research Lab., Eindhoven (W. J. Oosterkamp).	10 to 50	Free air, parallel plate (modified since references).	4b, 9b	Thoreaus, R., et al. (1955); Oosterkamp, Somerwil (1952); Oosterkamp, Proper (1952).
11	New Zealand	Dominion X-ray and Radium Lab., Christchurch (G. E. Roth).	50 to 300	Free air, parallel plate	--------	
12a	Sweden	Inst. of Radiophysics, Stockholm (R. Thoreaus).	50 to 250	Free air, cylindrical	4a, 13b, 14a	Thoreaus et al. (1955); Thoreaus (1954) (1932); Thoreaus (1946); and Wyckoff, H. O. (1956).
12b	do	do	8 to 50	do	4b	Thoreaus et al. (1955); Thoreaus (1954) (1932).
13a	United Kingdom	National Physical Lab., Teddington (G. H. Aston).	5 to 50	Free air, parallel plate	--------	
13b	do	do	30 to 100	do	12a, 14a	Wyckoff, H. O. et al. (1954).
13c	do	do	100 to 300	do	14a	Smith, E. E. (1955); Aston and Attix (1956).
14a	United States	National Bureau of Standards Washington, D. C. (H. O. Wyckoff).	60 to 250	do	6a, 13b, 13c, 12a	[a] Allisy et al. (1957).
14b	do	do	20 to 100	do	--------	Ritz (1959) (1960).
14c	do	do	--------	12 atm parallel plate	--------	Wyckoff (1960).

[a] Wyckoff, H. O. et al. (1954); Aston, G. H., and Attix, F. H. (1956); Thoreaus, R., and Wyckoff, H. O. (1956).

It is assumed for purposes of the analysis that the instrument is in satisfactory operating condition, and that the user of an instrument understands its operation so that inaccuracies due to instrument condition or to improper use are not introduced. However, the latter may not be a valid assumption unless the user has been supplied with adequate instructions on the proper use of the instrument and on means for checking its operation and constancy of calibration.

7.2.B. *Types of instruments.* For this study, only instruments that may be considered as laboratory standards were included. This would exclude instruments used principally as radiation detectors. The instruments considered are all of the ionization-chamber type and are intended for the measurement of exposure dose, although some can be used to measure exposure dose rate as well. Most of the instruments employed condenser ionization chambers with some form of electrometer to measure the loss in charge on the condenser when the chamber is exposed to radiation. To obtain chambers with different roentgen ranges, some instruments employ different capacitances while others use approximately the same capacitance but different-sized air cavities.

7.2.C. *Factors affecting accuracy.*

a. *Energy dependence.* The ionization within the gas of the cavity chamber is principally produced by the secondary electrons generated in the chamber walls and the center electrode. Thus the amount of ionization produced in the cavity chamber for a given exposure dose rate depends upon the atomic number of the walls and of the center electrode. Most manufacturers attempt to produce an energy independent response; for example, by making the major portion of the wall of somewhat lower atomic number than air and the center electrode of higher atomic number. At best these are approximations and there is some deviation of response with energy. In addition, the walls of the cavity must be of sufficient thickness to provide electronic equilibrium for the highest quantum energies to be measured. The larger attenuation in this wall for low energy radiation is sometimes partially compensated for by using a thin layer of higher atomic number material on the inside of the wall. Over the range of X-rays produced by 60 to 250 kv corresponding to half-value layers of about 1 mm aluminum to 3 mm copper, variations of up to 10 percent due to energy dependence have been observed. In some cases, by proper choice of materials and adequate wall thickness, this energy dependence is not exceeded even for energies as high as those from cobalt-60.[13]

b. *Calibration.* The manufacturer may supply a calibration certificate when the instrument is purchased, giving the chamber correction for each of several radiation qualities, or he may attempt to adjust the instrument so that its response is within a given accuracy limit for a certain radiation energy range. Although most manufacturers claim an accuracy of about ±2 percent, individual instruments submitted to calibration centers have been found to have correction factors differing from the nominal value by as much as 20 percent. Such large corrections may be due to an improperly functioning instrument or to other causes such as the use of an electrometer other than the one with which the chamber was initially calibrated.

c. *Scale characteristics.* The accuracy with which the scale can be read depends on the illumination of the scale, the sharpness of the scale markings, the sharpness of fiber image or instrument pointer, and the size of the smallest division. For instruments covered by this report, scales can be read to better than 0.5 percent of full scale, and scale linearity is within 1 to 2 percent of full scale. Reproducibility of successive readings at the same part of the scale, other factors being held constant, are usually within about ±0.5 percent of full scale.

d. *Dose rate dependence.* Saturation conditions are assumed to be met if loss of ions is 1.0 percent or less. These conditions will depend on the ionization chamber size and shape, the collecting potential, and the exposure dose rate of the radiation incident upon the chamber. With condenser chambers having full scale range of the order of 25 r, saturation conditions are usually met for dose rates that will produce half-scale deflection in about 1 second. For ordinary use, exposure dose rates will normally be somewhat less than those for the saturation limit.

e. *Leakage.* Chamber leakage (25 r or similar range) in normal background radiation was found to be negligible for the majority of instruments reported but ranged up to about 0.7 percent per hour for others. Use of a dessicator cabinet to store instruments is usually helpful in reducing leakage under conditions of high humidity. Chamber leakage should be routinely tested although the leakage is usually negligible for instruments in good repair.

Radiation-induced stem leakage for condenser ionization chambers may be significant for gamma rays and heavily filtered X-rays. These effects may range up to nearly 25 percent for high range chambers on cobalt-60 gamma rays (Braestrup and Mooney, 1958).

f. *Atmospheric temperature and pressure.* The ionization chambers, considered here are generally not hermetically sealed,[14] so the mass of air contained in the cavity must be corrected for temperature and pressure. After moving the instrument to a new location, the instrument must be permitted time to attain temperature and pressure equilibrium before measurements are made.

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[13] By empirical adjustment of the amount of aluminum in the center electrode of the instrument shown in fig. 7.3, or for a similar instrument constructed with air equivalent material (Shonka et al., 1958), the energy dependence from 60 kv X-rays (HVL, 3mm Al) to cobalt-60 gamma rays does not exceed ±3 percent.

[14] Attention is called to the possibility that chambers usually not sealed may inadvertently become sealed because of a very tight-fitting outer wall.

g. *Consistency of readings.* Reproducibility of readings is generally about 0.5 percent of full scale when temperature and pressure equilibrium conditions of chamber have been reached, and after the required warmup time for other components.

h. *Overall accuracy.* If an instrument is in good repair and has been adjusted by the manufacturer to read as nearly correctly as feasible, its correction factor is usually within 5 percent of unity for the X-ray quality range of 0.2 to 3 mm Cu HVL.

i. *Operation and constancy checks.* Two types of tests may be used to test the stability of condenser ionization-chamber instruments over extended periods of time. The response of the instument depends upon the electrometer sensitivity, the mass of air in the cavity, and the capacitance of the chamber and electrometer. Voltages may be applied to the electrometer to test its sensitivity. The voltage sensitivity of a number of instruments has been followed by various groups over a period of several years (Adams, 1958; Wood, 1955, 1958). These studies have indicated a good correlation between voltage sensitivity and instrument response. Wood found that the sensitivity was dependent upon the electrometer's charge history, but Adams found no such effect.

Instruments which have been returned for recalibration and found to have a shift in the correction factor, have usually shown a corresponding shift in voltage sensitivity although the percentage change in the correction factor has not always been the same as the percentage shift in voltage sensitivity.

The overall stability of both the electrometer and condenser ionization chamber may be checked with a radioactive source in a fixed geometry. The radioactive material may be in the form of a radium foil sealed within a double-walled cylinder which is itself contained within a lead protective container and so arranged that when the ionization chamber is inserted it will be close to the cylindrical radioactive source. In this way a source of only about 1 mg of radium may be sufficient to discharge a 25-r chamber in about 5 min. A larger source in a different geometry has been used by Thoreaus (1956). Strontium-90 imbedded in a plastic cylinder has also been used as a check source, the energetic beta rays and bremsstrahlung providing the ionizing radiation. However, accurately reproducible placement and orientation of the chamber is required, particularly for beta sources.

In condenser-chamber instruments, the "discharge check" may be useful for spotting changes in the voltage sensitivity of the electrometer if the ratio of the capacity of the chamber to the capacity of the chamber and electrometer has remained constant. Though less searching than the tests already mentioned, it has the advantage of requiring no extra equipment. To carry out this test, the chamber and electrometer are first fully charged, then the electrometer alone is discharged. The charge remaining on the chamber is then shared with the electrometer and the reading noted. If the chamber and electrometer are again separated and the latter discharged, the charge remaining on the chamber can be shared a second time with the electrometer and a new reading obtained. This procedure, continued a few times, gives several readings across the scale of the electrometer which should be reproducible at any time if the capacity of the system and the voltage sensitivity of the electrometer are unchanged.

The desirable frequency of calibration for the instrument will depend to some extent on the care with which it is handled. If there is an unexplained change in the sensitivity as indicated by the checking source, it is considered good practice to have the instrument recalibrated.

7.2.D. *Summary.* The fraction of the total number of instruments in use which have been submitted to calibration centers after their purchase varies considerably from country to country. In some countries, nearly all of them have had some subsequent checks; in others, possibly less than 10 percent of them have been calibrated after their purchase. Judging from the instruments which have been tested, correction factors may differ from unity by 20 percent for individual instruments although the average difference is more nearly 2 to 5 percent mainly due to energy dependence for radiation having HVL's of from 3 mm Al to 3 mm Cu.[15] Compensating factors introduced into the design of the chamber have reduced energy dependence to this level. Successive readings can be made to about ±0.5 percent of full scale which is about the limit to which the scale can be read. Saturation conditions are satisfactorily met for chambers having ranges of the order of 25 r at the dose rates usually encountered. Chamber leakage is ordinarily negligible but should be checked. If chambers are not hermetically sealed, atmospheric temperature and pressure corrections are required.[16] Instruments in good repair and properly used may usually be expected to maintain their calibration to within about 2 percent. Some form of operation and response constancy check is desirable in order to determine whether or not the sensitivity of the instrument has changed and it should be recalibrated.

7.3. *X-ray quality.*

An X- or gamma radiation field, at a point, is defined by the quantity, direction, and spectral distribution of radiant energy which enters per unit time through the surface of a small sphere surrounding the point. The quantity and direction of the radiation is often known but it is still necessary to have some measure of the relative spectral distribution to be sure that two radiation fields are equivalent. The spectral distribution of the energy can either be measured by instru-

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[15] The difference is usually considerably larger if the quality range is extended downward.

[16] See footnote 14.

ments such as scintillation spectrometer, crystal diffraction spectrometer, Compton magnetic spectrometer, etc., be approximated by using thick-target bremsstrahlung theory or obtained by analyzing attenuation curves of the radiation in materials such as aluminum, copper, etc.

Computations and most measuring techniques mentioned above are long and often tedious. Even the recently developed method of scintillation spectrometry cannot be carried out routinely in every clinical laboratory.

Therefore, an indirect means of describing the spectral distribution is necessary even though it is very approximate. This description is based on the fact that two equal quantities of radiant energy, the spectral distributions of which are different, generally have different penetrability in matter; in this case, one says the "qualities" of the two radiations are different. As the attenuation of X- or gamma radiation by material is easy to measure, it is the preferred phenomenon for distinguishing radiations of different qualities.

In order to obtain an unambiguous determination, it is best to make the measurement under "narrow beam" condition so that measurement of an unknown portion of the scattered photons is avoided. It is then possible to roughly characterize the radiation quality by the thickness of matter necessary to have a specified attenuation.

In radiology the thickness of material necessary to obtain an attenuation of 50 percent of the exposure dose is used. This thickness is called the half-value layer (HVL).

The substances used for the determination of HVL vary with the spectral distribution of the radiation. Their selection is dictated by practical considerations: On the one hand it is not desirable to use small thicknesses which may be difficult to measure and maintain; on the other hand, it is better to work with materials whose attenuation coefficients vary with photon energy as rapidly as possible and have no discontinuities, for those will produce the greatest distinction between nearly similar spectra.

Taylor and Singer (1934) advocated the use of complete attenuation curves. It is clear that this method characterizes a heterogeneous radiation better than the HVL. Two heterogeneous radiations having the same attenuation curve can be considered fairly similar while, if they only have the same HVL, they can be widely different.

This is demonstrated by the example given in figures 7.4a and 7.4b (Hettinger and Starfelt, 1958). The curve, S_1, of figure 7.4a shows the spectral distribution of the exposure dose rate measured for the output of a tube (100 kv, inherent filtration equivalent to 4 mm aluminum and an additional filtration of 1.67 mm aluminum +0.87 mm copper). The curve, S_2 in figure 7.4b represents the spectral distribution of the exposure dose rate emitted by the same tube (220 kv, inherent filtration equivalent to 4 mm aluminum without additional filtration). Both kinds of radiations have almost the same first HVL; viz, 0.59 and 0.55 mm copper respectively. The relatively low value of the HVL of the 220-kv radiation is partially explained by the important contribution of characteristic radiation. Because of the high absorption coefficients for the characteristic line and the low energy tail of the spectrum (Fig. 7.4b) and the large contribution to the exposure dose in this region, this part of the spectrum largely governs the magnitude of the HVL. The resulting calculated spectral distribution of the exposure does rate for the 220-kv radiation after having passed through the HVL is given by the curve S'_2, and for the 100-kv radiation by the curve S'_1.

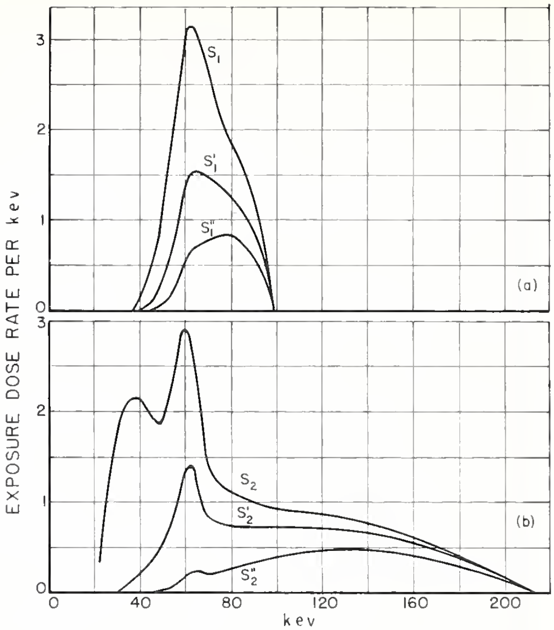

Figure 7.4 a and b. *Spectral distribution of X-rays produced by 100 kv accelerating potential (curve* S_1*) and 220 kv (curve* S_2*) (Hettinger and Starfeld, 1958).*

Curve S′ and S″ are the calculated spectral distributions after the radiation has passed through 1 and 2 HVL of copper respectively.

To obtain a more complete description of radiation quality than is given by the HVL, several authors proposed the use of a second HVL in addition to the first. The ratio of the first to the second HVL is an indication of the homogeneity of the beam. This ratio, which is sometimes called the homogeneity coefficient, is unity for monochromatic rays. The heterogeneity coefficient, which is the reciprocal of the homogeneity coefficient, is sometimes used. In the example of figure 7.4a and 7.4b, the curves S''_1 and S''_2 represent the calculated spectral distributions of the radiation after having passed through the second HVL. The corresponding homogeneity coefficients are 0.88 and 0.40, respectively.

The presence in the X-ray beam of an appreciable amount of characteristic radiation of energy above about 50 kev gives a small first HVL layer and therefore a small value for the homogeneity coefficient: which is illustrated by the spectrum in figure 7.4b.

However, when the X-ray beam contains an appreciable amount of characteristic radiation of low energy another factor becomes important. The large energy-absorption coefficient of air strongly weights the spectrum in this energy region. As Zieler (1954) has shown, the presence of low energy characteristic radiation produces first a maximum and then a minimum of the homogeneity coefficient with increasing filtration. It is obvious that under these conditions the homogeneity coefficient is no longer a useful concept.

7.4. *Practical measurements of HVL.*

a. For clinical use, the first and second half-value layers are, as a rule, sufficient for characterizing the quality of X-radiations.

b. Suitable materials for use as attenuators are.:
Aluminum for radiations from 10 to 120 kv
Copper for radiations from 120 to 400 kv
Tin or lead for radiations from 400 to 1000 kv
Lead for radiations above 1000 kv.

These materials, especially the aluminum for low quantum energy radiation, must be of sufficient chemical purity. Hübner (1958) has shown the influence of small amounts of impurities.

c. Successive layers of these materials are inserted between the radiation source and the measuring instrument until an attenuation of 50 percent of the original exposure dose is obtained. This thickness of the material is the HVL. If the layers of material at one's disposal are not sufficiently differentiated to obtain a precise attenuation of 50 percent, a graphical interpolation between adjoining thicknesses should be performed.

d. If necessary, the readings of the measuring instrument are to be corrected for the change in the quality of the radiation as it passes through the attenuator although this may be difficult at the softer qualities.

e. As the material used as an attenuator emits a certain amount of scattered radiation, care must be taken that the influence of this radiation is eliminated as much as possible. In the usual case of measuring the HVL of an X-ray beam, this can be done by (1) keeping the irradiated area of the attenuator as small as possible, and (2) placing the measuring instrument at such a distance from the attenuator that the scattered radiation diverges sufficiently before reaching it. If a diaphragm is used to limit the useful surface of the attenuator, the diaphragm must be of such a material and thickness that no appreciable amount of scattered radiation from this source can reach the measuring instrument. A slightly conically-shaped aperture with the top of the cone directed toward the dosimeter is advantageous in reducing the scattered radiation from the diaphragm which is measured by the dosimeter.

f. The dimensions of a suitable measuring arrangement can be derived from the curves of figures 7.5a, 7.5b, and 7.5c. The figure 7.5a (Somerwil 1957) indicates the variations of the

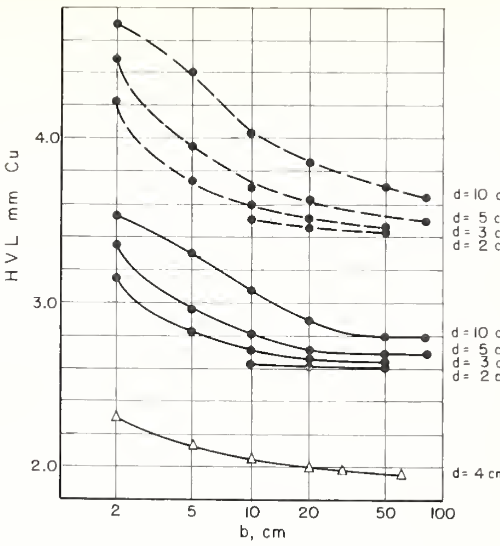

FIGURE 7.5a. *Variation of half-value layers with parameter* d (*diameter of the field at the attenuator*) *and* b (*chamber-attenuator distance*).

The focus-attenuator distance is F.

●—●—● First HVL 250 kv, 2mm Cu filtration, F=20 cm (Somerwil, 1957).

●——● Second HVL 250 kv, 2 mm Cu filtration, F=20 cm.

△—△—△ First HVL 300 kv, 1.3 mm Cu inherent filtration, F—21.5 cm (Farr, 1955).

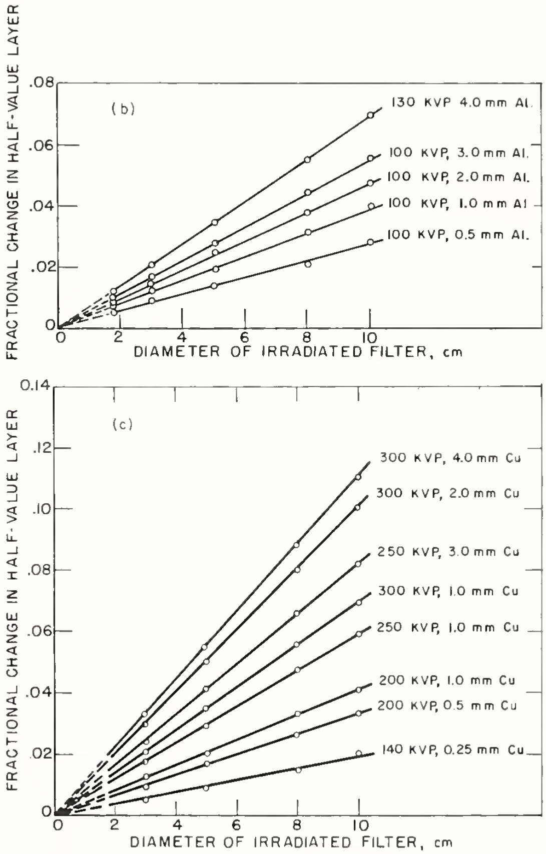

FIGURE 7.5 b and c. *Fractional change of half-value-layer for the indicated potentials and filtrations with* $F = 25$ *cm and* $b = 25$ *cm* (see fig. 7.5a for definition of F and b).

From Trout, E. D., Kelley, J. P., and Lucas, A. C., Determination of half-value layer, Am. J. Roentgenol. 84, 729 (1960). Courtesy of Charles C. Thomas, Publisher) first and second HVL in terms of the irradiated attenuator diameter and of the attenuator-dosimeter distance. Figures 7.5b and 7.5c (Trout et al., 1959) indicates the variations of relative HVL with irradiated attenuator diameter, kilovoltage and filter thickness. Both sets of curves were plotted in the way proposed by Trout, who found that a linear relation exists between the attenuator diameter and the HVL in most practical cases. For precision determination of HVL's, therefore, the value found at various diameters could be plotted against those diameters. By extrapolation to zero diameter the influence of the scattered radiation from the attenuator could be eliminated.

7.5. *Determination of the equivalent inherent filtration of X-ray tubes.*

The equivalent inherent filtration of X-ray tubes depends on the tube voltage and on the wave form. It also depends on the method used for determining it. All methods use an X-ray source with very low filtration (very thin glass or beryllium window) and a cutaway tube unit of typical component thickness (glass, oil, bakelite, etc.). One method (Taylor and Singer, 1934; Reinsma, 1958) consists in determining with a low filtration tube the copper and aluminum thickness that produces the same reduction of the exposure dose as that produced by the cutaway. Another method (Trout et al., 1956; Reinsma, 1960; Thoreaus, 1940) consists in determining with a low filtration tube the copper or aluminum thickness that gives a radiation of the same quality as that emerging from the cutaway. In this method, the radiation quality is characterized by the HVL. Table 7.5 (Reinsma, 1960) indicates the differences in the equivalent inherent filtrations obtained by the two methods. Figure 7.6 indicates the variation in HVL with total filtration aluminum equivalent of an X-ray tube for different constant and sine wave tube voltages. The influence of the wave form on the radiation quality is also given in table 7.6 (Reinsma, 1960).

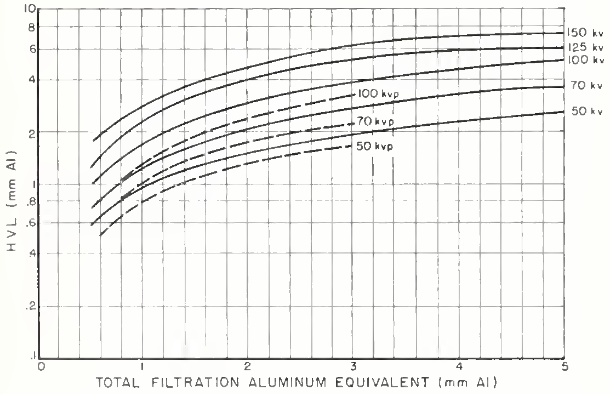

FIGURE 7.6. *Variation of HVL with total filtration.*

Solid curve—constant potential (Reinsma, 1950).
Dotted curve—sine wave potential (Trout et al., 1956).

Table 7.5.—Equivalent inherent filtration of a diagnostic X-ray tube

Tube voltage (constant potential)	Equivalent inherent filtration	
	From measurements of reduction in exposure dose	From HVL measurements
kv	*mm* Al	*mm* Al
50	1.7	1.4
100	1.9	1.8
150	2.1	1.8

Table 7.6.—Radiation quality of a diagnostic X-ray tube (without extra filtration)

Tube voltage	Half-value layer	
Wave-form	Constant potential	Full wave rectified
kv	*mm* Al	*mm* Al
50	1.25	1.16
100	2.75	1.9
125	3.2	2.3

8. Methods of Calculating the Absorbed Dose from Measured Ionization

8.1. *General principles.*

Although it is clearly recognized that the measurement of ionization is not the only approach by which the absorbed dose in rads can be determined, it nevertheless is the most common method now in use. This section is an attempt to summarize the usual techniques employed in such measurements, and to tabulate some of the data needed in carrying out the related calculations. These are selections from the current literature. It is anticipated that future experimental work will improve the accuracy with which some of these data are known. The ICRU plans to publish revised figures from time to time.

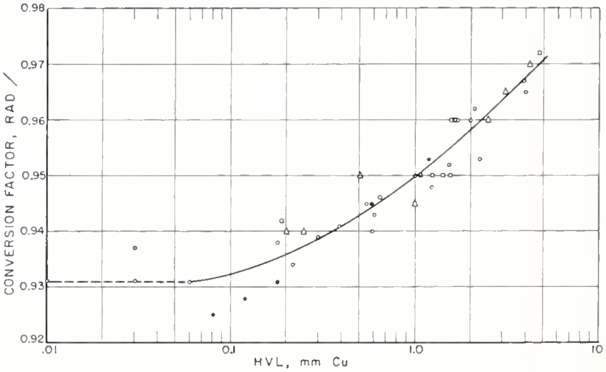

FIGURE 8.1a. *Plot of values of* $\overline{f}$ *muscle from data of table 8.2.*

△ ICRU 1956.
○ Calculated from primary spectra.
● Measured by means of gas equivalents.
□ Calculated from spectrum in phantom.

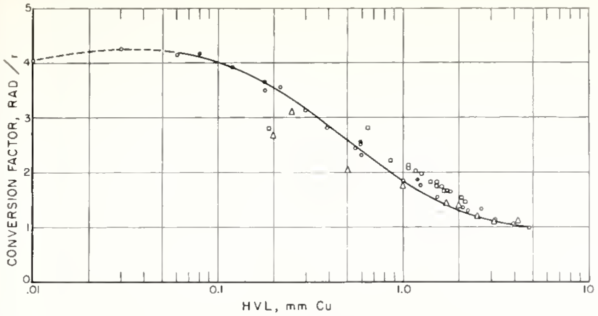

FIGURE 8.1b. *Plot of values of* $\bar{f}$ *bone from data of table 8.2.*
△ ICRU 1956.
○ Calculated from primary spectra.
● Measured by means of gas equivalents.
□ Calculated from spectrum in phantom.

In general, absorbed dose may be estimated by application of the relation between the ionization produced in a gas filled cavity at the place of interest in the material irradiated and the energy imparted to unit mass of the material (Gray, 1936). When the cavity is sufficiently small the gas will be subjected to the same flow of ionizing particles as the material under consideration. Then the energy, E_m, in ergs absorbed per gram of the material, is related to the ionization per gram of the gas, J_m, by the equation

$$E_m = J_m W s_m \tag{8.1}$$

where

W= the average energy in ergs [17] expended by the ionizing particles crossing the cavity per ion pair formed, and

s_m= the ratio of the effective mass stopping power of the medium to that of the cavity gas for these ionizing particles [18].

Furthermore, the absorbed dose, D, can be written as

$$D = 0.01\ E_m \text{ rad.} \tag{8.2}$$

The average value of W in air for X-, gamma, or neutron radiation probably lies between 33 and 35.5 electron volts. It is recommended that the value W=34 electron volts [19] ($5.4_4 \times 10^{-11}$ ergs) be used for calculations concerning X- and gamma radiation of quantum energy greater than 20 kev, and that 35 ev [20] be used for calculations involving fast neutron radiation.

The following paragraphs suggest convenient ways of applying the cavity relation to the measurement of absorbed dose. However, before proceeding further it will be worthwhile to define what is meant by the term "charged particle equilibrium" (CPE) and to describe the conditions necessary for its achievement.

8.2. *Charged-particle equilibrium (CPE) conditions.*

Charged particle equilibrium would exist at a point within a medium under irradiation if

(a) the intensity and energy spectrum of the primary radiation were constant throughout a region extending in all directions from the point, to a distance at least as great as the maximum range of the secondary charged particles generated by the primary radiation, and

(b) the energy absorption coefficient for the primary radiation (see sections 8.4, 8.5, and 8.6) and the stopping power for the secondary charged particles were constant in the medium throughout the same region as in (a).

If these two conditions could be fulfilled, then on the average for every secondary charged particle leaving an infinitesimal volume surrounding the point, another secondary charged particle of practically the same energy would enter. Thus, the energy dissipated within the volume would be equal to that which would have been dissipated if all the secondary charged particles originating there had spent their entire energy within that volume. This is, therefore, an alternative definition of strict CPE. The presence of a small amount of foreign material at or near the point of interest (such as the air within a cavity ionization chamber) will not significantly upset CPE provided that the quantity of material is so small that a negligible part of the energy absorbed within it results from direct interaction of the primary radiation with the molecules of the foreign material. The only situation in which true CPE can exist in a rigorous mathematical sense is in an infinite medium containing a uniformly distributed beta- or gamma-emitting nuclide. When the primary photon or neutron source is external to the irradiated material, the secondary particle flux may differ appreciably from the true equilibrium value. However, for primary photon energies of about one Mev or below and for neutrons the difference between the actual secondary particle flux and the true equilibrium value may not be important for many applications.

The energy spectrum of charged particles existing at a point under CPE conditions is called an *equilibrium spectrum.* If the charged particles involved are electrons, CPE is often referred to as "electronic equilibrium."

Some typical situations where CPE conditions do not obtain are (a) near a source of radiation, where the intensity of the primary radiation is changing rapidly; (b) near boundaries between materials of different composition, such as bone-tissue or air-tissue boundaries (see section 3.5); and (c) at high energy where the primary radiation is appreciably attenuated in the absorbing medium over a distance equal to the mean range of the secondary particles produced. In this latter case we may speak of the attainment of "transient equilibrium" (by analogy with transient equilibrium in e.g., Ra A decay) at a depth where the

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[17] W is often expressed in ev per ion pair where 1 ev=1.602×10^{-12} ergs.

[18] This term is a function of the charged particle energy at the point of interest, the material and the gas and to a smaller extent of the dimension of the air cavity. Values for this ratio under different experimental conditions are discussed later in this section.

[19] The mean of several recent determinations of W for electrons in air is 33.5 ev (Boag, 1959). See table 8.9 for recently published values.

[20] This assumes that negligible ionization is produced by heavy particle recoils.

ratio of primaries to secondaries reaches a constant value.

8.3. *Absorbed dose in air exposed to X- or gamma radiation.*

The absorbed dose at a point in air or "air-equivalent" material that is surrounded on all sides by such material to a thickness at least equal to the range of the secondary electrons and uniformly exposed to 1 roentgen of X- or gamma radiation is equal to

$$\left(\frac{1\ \text{esu}}{0.001293\ \text{g air}}\right)\times\left(2.082\times10^{9}\,\frac{\text{electron}}{\text{esu}}\right)\times\left(34\,\underset{\text{(or ion pair)}}{\frac{\text{ev}}{\text{electron}}}\right)\times\left(1.602\times10^{-12}\,\frac{\text{erg}}{\text{ev}}\right)\times\left(\frac{1\ \text{rad}}{100\ \text{erg/g}}\right)=0.87_{7}\ \text{rad} \qquad (8.3)$$

for all qualities of X- or gamma radiation with quantum energies greater than 20 kev, the energy range over which W is assumed constant.

D_{air} will be referred to as the absorbed dose at a point in an extended mass of air under CPE conditions.

It follows that if any ionization chamber or other measuring instrument that has been calibrated in roentgens against a standard free air chamber, or in any other way, records under conditions of electronic equilibrium in the wall of the chamber an exposure dose of R roentgens in any situation, then in the same situation:

$$D_{\text{air}}=0.87_{7}R\ \text{rad} \qquad (8.4)$$

8.4. *Absorbed dose in tissue or other material exposed to X- or gamma radiation.*

The estimation of absorbed dose presents two somewhat different problems dependent upon whether or not electronic equilibrium may be assumed. These will be treated separately.

Case I. Electronic Equilibrium:

(a) *Using an ionization chamber calibrated in roentgens.* The absorbed dose at some point within an irradiated medium can be measured indirectly by means of a roentgen-calibrated ionization chamber which can be positioned at the point within the medium. The calibration of the cham-

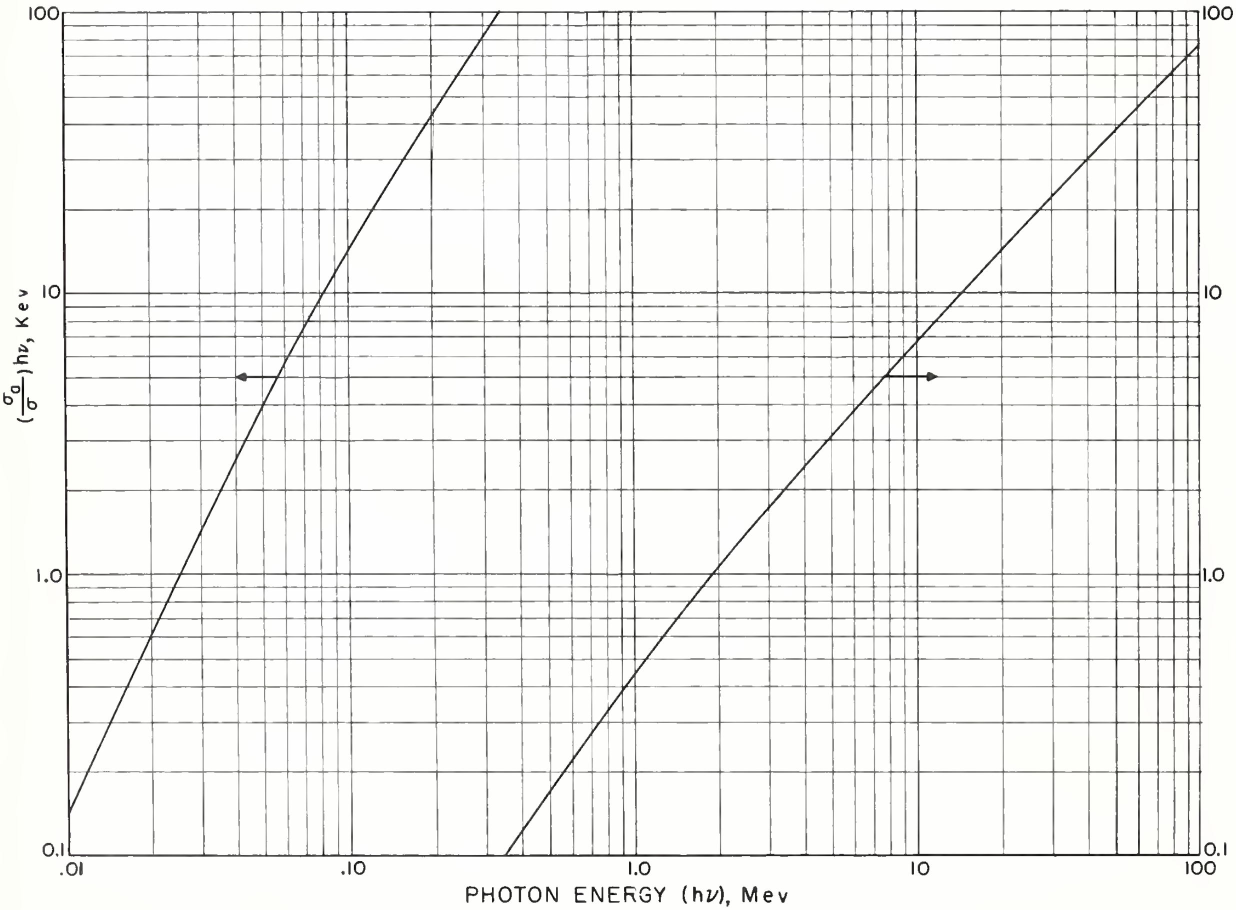

FIGURE 8.2. *Graph of* $(\sigma_a/\sigma)h\nu$, *the mean initial energy of the Compton recoil electrons produced by monochromatic* γ*-rays of quantum energy* $h\nu$.

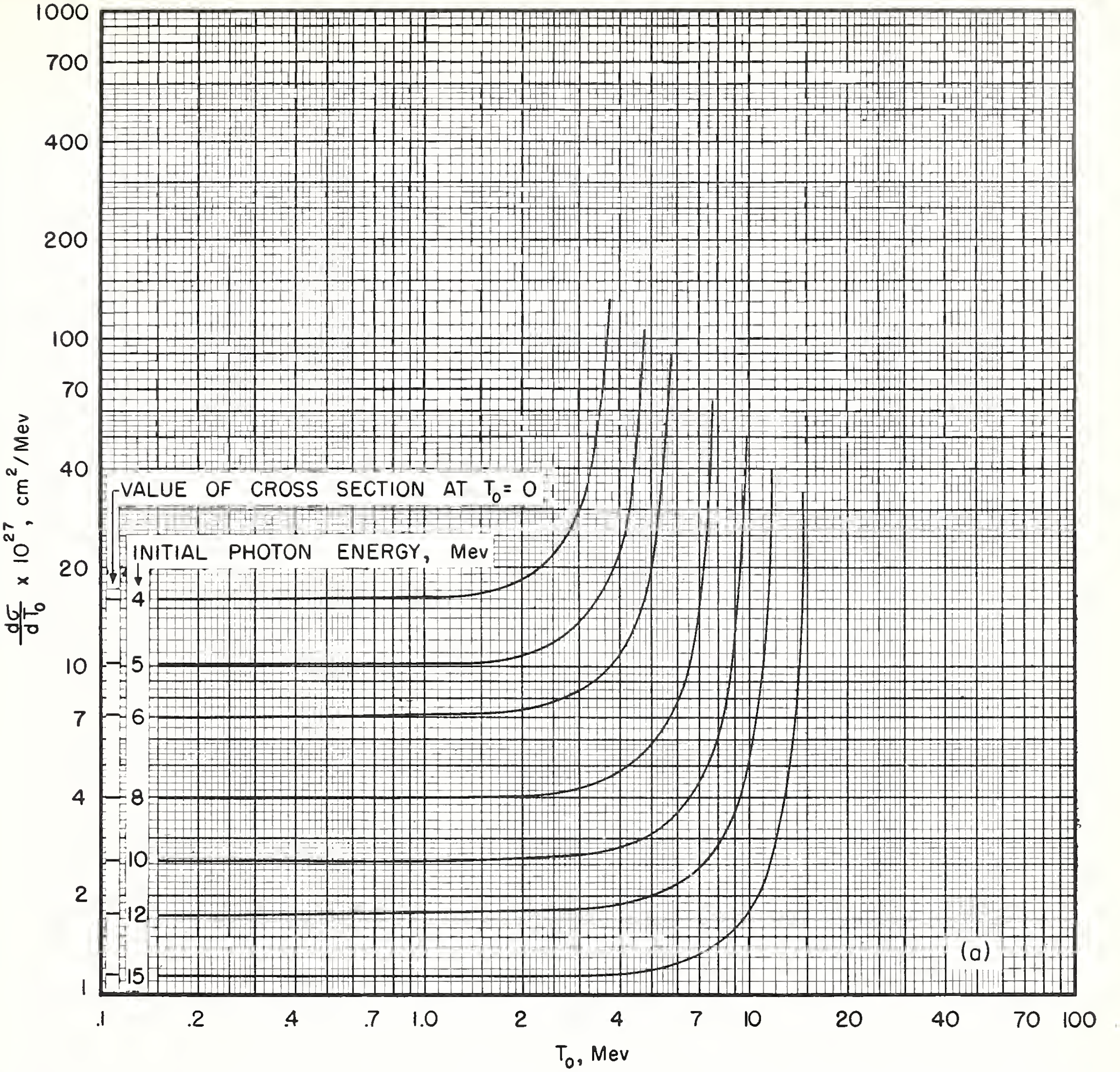

FIGURE 8.3a. *Starting-energy distribution for Compton-recoil electrons produced by photons with initial energies of 4 to 15 Mev (Nelms, 1953).*

The ordinate is the absolute differential cross section for giving a free electron a recoil energy in the inerval from T_0 to T_0+dT_0.

ber should be correct for the spectrum of X-rays present at the point in the medium, and the chamber walls thick enough to exclude electrons generated in the medium.

In position, the chamber measures a perturbed exposure dose R'. The exposure dose which would exist at the point in the absence of the chamber is

$$R=A_mR' \tag{8.5}$$

where A_m is a correction factor which accounts for all effects causing R and R' to differ, mainly (a) attenuation and scattering of X-rays in that amount of medium which is displaced by insertion of the chamber, and (b) X-ray scattering from the chamber connecting lead if one is used. (Modern ionization chambers can be made so small that A_m is very nearly equal to 1.)

Having thus established R at the point in roentgens, we obtain the dose in the medium, D_{med} in rad from the following relation, assuming electronic equilibrium to exist.

$$D_{\text{med}}=0.87_7R\frac{(_m\mu_{en})_{\text{med}}}{(_m\mu_{en})_{\text{air}}}=fR \tag{8.6}$$

where $(_m\mu_{en})$ is the energy absorption coefficient[21]

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[21] $_m\tau$, $_m s_n$, and $_m\kappa$ are the mass absorption coefficients for photoelectric effect, Compton effect, and pair production, respectively. For present purposes these coefficients shoud include corrections for energy losses through fluorescence, annihilation radiation, and bremsstrahlung produced by the electrons as they slow down. Such corrections are described by Fano (1953), and are small for the energy range considered here. The data in table 8.1 provided by G. White (1956) include these corrections.

TABLE 8.1—*Values of the mass energy-absorption coefficients and the factor f*

Photon energy	Mass energy-absorption coefficient, $(_{m}\mu_{en})$cm²/g.																		$f=0.87_7\left[\frac{(_{m}\mu_{en})_{medium}}{(_{m}\mu_{en})_{air}}\right]$		
	H	C	N	O	Na	Mg	Al	P	S	A	K	Ca	Poly-styrene $(C_8H_8)_n$	Lucite $C_5H_8O_2$	Poly-ethylene $(CH_2)_n$	Bakelite $(C_{43}H_{38}O_7)$	Water	Air	Com-pact bone	Muscle	$\frac{\text{Water}}{\text{Air}}$	$\frac{\text{Compact bone*}}{\text{Air}}$	$\frac{\text{Muscle*}}{\text{Air}}$
Mev																						
0.010	0.0099	1.94	3.42	5.50	15.4	20.9	26.5	40.1	49.7	62.0	77.0	89.8	1.79	2.92	1.66	2.43	4.89	4.66	19.0	4.96	0.92_0	3.5_8	0.93_3
.015	.0110	0.517	0.916	1.49	4.43	6.09	7.65	11.9	15.2	19.4	24.6	28.9	0.478	0.788	0.444	0.651	1.32	1.29	5.89	1.36	$.89_7$	4.0_0	$.92_5$
.020	.0133	.203	.360	0.587	1.77	2.47	3.16	5.00	6.41	8.31	10.5	12.5	.188	.311	.176	.257	0.523	0.516	2.51	0.544	$.88_7$	4.2_7	$.92_5$
.030	.0186	.0592	.102	.163	0.482	0.684	0.880	1.45	1.85	2.46	3.12	3.75	.0561	.0892	.0534	.0743	.147	.147	0.743	.154	$.87_7$	4.4_3	$.91_9$
.040	.0230	.0306	.0465	.0700	.194	.274	.351	0.570	0.731	0.974	1.25	1.52	.0300	.0426	.0295	.0368	.0647	.0640	.305	.0677	$.88_7$	4.1_8	$.92_8$
.050	.0270	.0226	.0299	.0410	.0996	.140	.176	.282	.361	.484	0.626	0.764	.0229	.0288	.0232	.0259	.0394	.0384	.158	.0409	$.90_0$	3.6_1	$.93_4$
.060	.0305	.0203	.0244	.0304	.0637	.0845	.104	.166	.214	.284	.367	.443	.0211	.0243	.0218	.0226	.0304	.0292	.0979	.0312	$.91_3$	2.9_4	$.93_7$
.080	.0362	.0201	.0218	.0239	.0369	.0456	.0536	.0780	.0971	.124	.158	.191	.0213	.0226	.0224	.0217	.0253	.0236	.0520	.0255	$.94_0$	1.9_3	$.94_8$
.10	.0406	.0213	.0222	.0232	.0288	.0334	.0372	.0500	.0599	.0725	.0909	.111	.0228	.0235	.0241	.0227	.0252	.0231	.0386	.0252	$.95_7$	1.4_7	$.95_7$
.15	.0485	.0246	.0249	.0252	.0258	.0275	.0282	.0315	.0351	.0368	.0433	.0488	.0264	.0267	.0280	.0261	.0278	.0251	.0304	.0276	$.97_1$	1.0_6	$.96_4$
.20	.0530	.0267	.0267	.0271	.0265	.0277	.0275	.0292	.0310	.0302	.0339	.0367	.0287	.0289	.0305	.0283	.0300	.0268	.0302	.0297	$.98_2$	0.98_8	$.97_2$
.30	.0573	.0288	.0289	.0289	.0278	.0290	.0283	.0290	.0301	.0278	.0304	.0319	.0310	.0311	.0329	.0305	.0320	.0288	.0311	.0317	$.97_7$	$.94_7$	$.96_5$
.40	.0587	.0295	.0296	.0296	.0283	.0295	.0287	.0290	.0301	.0274	.0299	.0308	.0317	.0319	.0337	.0312	.0329	.0296	.0316	.0325	$.97_5$	$.93_6$	$.96_3$
.50	.0589	.0297	.0297	.0297	.0284	.0293	.0287	.0288	.0300	.0271	.0294	.0304	.0319	.0320	.0339	.0314	.0330	.0297	.0316	.0327	$.97_4$	$.93_3$	$.96_6$
.60	.0588	.0296	.0296	.0296	.0283	.0292	.0286	.0287	.0297	.0270	.0291	.0301	.0318	.0319	.0338	.0313	.0329	.0296	.0315	.0326	$.97_5$	$.93_3$	$.96_6$
.80	.0573	.0288	.0289	.0289	.0276	.0285	.0278	.0280	.0287	.0261	.0282	.0290	.0310	.0311	.0329	.0305	.0321	.0289	.0306	.0318	$.97_4$	$.92_9$	$.96_5$
1.0	.0555	.0279	.0280	.0280	.0267	.0275	.0269	.0270	0280	.0252	.0272	.0279	.0300	.0301	.0319	.0295	.0311	.0280	.0297	.0308	$.97_4$	$.92_7$	$.96_5$
1.5	.0507	.0255	.0255	.0255	.0243	.0250	.0246	.0245	.0254	.0228	.0247	.0253	.0274	.0275	.0291	.0270	.0283	.0255	.0270	.0281	$.97_3$	$.92_9$	$.96_8$
2.0	.0464	.0234	.0234	.0234	.0225	.0232	.0227	.0228	.0235	.0212	.0228	.0234	.0252	.0252	.0267	.0247	.0260	.0234	.0248	.0257	$.97_4$	$.92_9$	$.96_3$
3.0	.0398	.0204	.0205	.0206	.0199	.0206	.0201	.0204	.0210	.0193	.0208	.0213	.0219	.0220	.0232	.0216	.0227	.0205	.0219	.0225	$.97_1$	$.93_7$	$.96_3$
4.0	.0351	.0184	.0186	.0187	.0184	.0191	.0188	.0192	.0199	.0182	.0199	.0204	.0197	.0198	.0208	.0194	.0205	.0186	.0199	.0203			
5.0	.0316	.0170	.0172	.0174	.0173	.0181	.0180	.0184	.0192	.0176	.0193	.0200	.0181	.0183	.0191	.0179	.0190	.0173	.0186	.0188			
6.0	.0288	.0160	.0162	.0166	.0166	.0175	.0174	.0179	.0187	.0175	.0190	.0198	.0170	.0172	.0178	.0168	.0180	.0163	.0178	.0178			
8.0	.0249	.0145	.0148	.0154	.0158	.0167	.0169	.0175	.0184	.0172	.0190	.0197	.0153	.0156	.0160	.0153	.0165	.0150	.0165	.0163			
10.0	.0222	.0137	.0142	.0147	.0154	.0163	.0167	.0174	.0183	.0173	.0191	.0201	.0144	.0147	.0149	.0144	.0155	.0144	.0159	.0154			

*See footnote 22.

(cm²/g) of the medium or of air, equal to $({}_m\tau+{}_m\sigma_a+{}_m\kappa)$ evaluated for the total spectrum of X- or γ-radiation arriving at the point of interest.

Table 8.1 gives values of the mass energy absorption coefficient for a number of elements and for water, air, bone, muscle, and some plastics. Table 8.1 also contains values of f in water, muscle, and bone for monochromatic photon energies.[22] Mean values, $\bar{f}$, of f integrated over several typical primary X-ray spectra (some measured and some calculated by Kramers' rule (1923)), or directly measured are given in table 8.2. For practical use, variations in $\bar{f}$ terms of HVL of some radiations have been plotted in figure 8.1. Table 8.2 gives information about kv and HVL for the plotted points.

(b) *Using an absolute ionization chamber.* In the event that no roentgen-calibrated ionization chamber is available, an air-filled cavity ionization chamber with nearly air-equivalent walls (e.g., carbon) can be employed as an absolute device.[23] When such a chamber is centered at the point of interest in the medium and the ionization is measured, it gives the absorbed dose in a thin inner layer of its own wall material, through application of the Bragg-Gray relation (8.1). Thus

$$D_{\text{wall}}=0.87_7Q(\tilde{s}_m)^{\text{wall}}_{\text{air}} \qquad (8.7)$$

where Q is the charge in esu carried by the ions of either sign produced per cm³ of air in the cavity 0 °C and 760 mm Hg(NTP) and $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$ is the ratio of the effective mass stopping power in the wall to that in air when the electrons initially have a Compton distribution but are slowed down to produce an equilibrium spectrum in the cavity.

If the measurement could have been made by means of a small air-filled cavity in the medium itself, D_{med} would have been obtained instead. D_{wall} differs from D_{med} for two reasons:

(i) The energy absorption coefficient in the actual chamber wall differs from that in the medium for the X-ray spectrum at the point, and

TABLE 8.2.—*Average absorbed dose per roentgen of exposure dose in compact bone and muscle for various X-ray spectra*

HVL		Tube potential	$\bar{f}$		Method (see footnotes below)	Reference
mm Al	mm Cu	kv	Compact bone* rad/r	Muscle* rad/r		
-------	-------	50	4.08	0.93_3	(a)	Ehrlich (1956).
-------	-------	50	4.10	$.93_3$	(a)	Ehrlich (1956).
-------	-------	50	4.12	$.93_3$	(a)	Ehrlich (1956).
0.25^{c}	-------	50	4.19	$.93_5$	(a)	Ehrlich (1956).
$.35^{c}$	$^{c}0.01$	50	4.03	$.93_1$	(a)	Wang, et al. (1957).
$.9^{c}$	$^{c}.03$	45	4.26	$.93_1$	(a)	Hettinger, Starfelt (1958).
1.0^{c}	-------	30	4.30	$.93_0$	(a)	Aitken, Dixon (1958).
1.2^{c}	$^{c}.03$	50	4.26	$.93_7$	(a)	Ehrlich (1956).
1.2^{c}	$^{c}.06$	75	4.15	$.93_1$	(a)	Hettinger, Starfelt (1958).
1.2^{c}	-------	40	4.30	$.93_4$	(a)	Aitken, Dixon (1958).
1.6^{c}	-------	50	4.26	$.93_5$	(a)	Do.
1.8^{c}	-------	60	4.23	$.93_5$	(a)	Do.
1.8^{c}	-------	70	4.16	$.93_4$	(a)	Do.
1.8^{c}	-------	80	4.04	$.93_6$	(a)	Do.
2.5	.08	60	4.17	$.92_5$	(b)	Allisy, Astier (1958).
-------	-------	-------	4.20	$.92_6$	(a)	Helle (1958).
3.2	.12	80	3.93	$.92_8$	(b)	Allisy, Astier (1958).
-------	-------	-------	3.90	$.92_7$	(a)	Helle (1958).
4.2	.18	100	3.66	$.93_1$	(b)	Allisy, Astier (1958).
-------	-------	-------	3.50	$.93_2$	(a)	Helle (1958).
-------	.18	100	3.50	$.93_8$	(a)	Hettinger, Starfelt (1958).
4.5^{d}	$^{d}.19$	140	2.81	$.94_2$	(a)	Cormack, Burke (1960).
-------	.2	150	2.69	$.94_0$	(a e)	ICRU Handbook 62 (1956).
-------	.22	80	3.57	$.93_4$	(a)	Kolb, Heitzmann.
-------	.25	100	3.10	$.94_0$	(a e)	ICRU Handbook 62 (1956)
-------	.30	140	3.15	$.93_9$	(a)	Hettinger, Starfelt (1958).
-------	.39	170	2.82	$.94_1$	(a)	Do.
-------	.50	200	2.05	$.95_0$	(a e)	ICRU Handbook 62 (1956).
-------	.55	220	2.46	$.94_5$	(a)	Hettinger, Starfelt (1958).
-------	.59	100	2.53	$.94_0$	(a)	Do.
-------	.59	150	2.55	$.94_5$	(b)	Allisy, Astier (1958).
-------	-------	-------	2.50	$.94_5$	(a)	Helle (1958).
-------	.60	120	2.31	$.94_3$	(a)	Kolb, Heitzmann.
-------	$^{d}.65$	250	2.58	$.94_6$	(a)	Hettinger, Starfelt (1958).
-------	$^{d}.86$	280	2.23	-------	(a)	Cormack (1958).
-------	1.0	160	1.86	$.95_0$	(a)	Kolb, Heitzmann.
-------	$^{d}1.0$	250	1.76	$.94_5$	(a)	ICRU Handbook 62 (1956)
-------	$^{d}1.07$	250	2.13	$.95_0$	(a)	Hettinger, Liden (1960).
-------	$^{d}1.07$	280	2.07	-------	(a)	Cormack et al. (1958b).
-------	$^{d}1.17$	280	2.02	-------	(a)	Do.
-------	1.2	200	1.89	$.95_3$	(b)	Allisy, Astier (1958).
-------	-------	-------	1.90	$.95_3$	(a)	Helle (1958).
-------	1.24	140	1.76	$.94_8$	(a)	Hettinger, Starfelt (1958).
-------	$^{d}1.25$	250	1.96	$.95_0$	(a)	Hettinger, Liden (1960).
-------	$^{d}1.42$	250	1.83	$.95_0$	(a)	Hettinger, Liden (1960).
	1.53	200	1.56	$.95_2$	(a)	Kolb, Heitzmann.
-------	$^{d}1.54$	280	1.76	-------	(a)	Cormack et al. (1958b).
-------	$^{d}1.55$	250	1.81	$.95_0$	(a)	Hettinger, Liden (1960).
-------	$^{d}1.63$	250	1.73	$.96_0$	(a)	Hettinger, Liden (1959).
-------	$^{d}1.68$	250	1.66	$.96_0$	(a)	Hettinger, Liden (1959).
-------	1.7	280	1.44	$.96_0$	(a)	ICRU Handbook 62 (1956).
-------	$^{d}1.71$	280	1.66	-------	(a)	Cormack et al. (1958b).
-------	$^{c}1.77$	280	1.65	-------	(a)	Do.
-------	$^{d}2.0$	250	1.42	$.96_0$	(a e)	ICRU Handbook 62 (1956).
-------	$^{d}2.07$	280	1.53	-------	(a)	Cormack et al. (1958b).
-------	$^{c}2.1$	220	1.36	$.96_2$	(a)	Greening (1947).
-------	$^{d}2.17$	280	1.46	-------	(a)	Cormack et al. (1958b).
-------	2.24	170	1.32	$.95_3$	(a)	Hettinger, Starfelt (1958).
-------	2.50	280	1.22	$.96_0$	(a)	ICRU Handbook 62 (1956).
-------	$^{c}2.63$	280	1.34	-------	(a)	Cormack et al. (1958b).
-------	3.1	280	1.13	$.96_5$	(a)	ICRU Handbook 62 (1956).
-------	$^{c}3.14$	280	1.18	-------	(a)	Cormack et al. (1958b).
-------	3.9	220	1.09	$.96_7$	(a)	Hettinger, Starfelt (1958).
-------	4.0	400	1.08	$.96_5$	(a)	Johns, et al. (1952).
-------	4.16	400	1.11	$.97_0$	(a)	ICRU Handbook 62 (1956).
-------	$^{d}4.8$	250	1.01	$.97_2$	(a)	Hettinger, Starfelt (1958).

* See footnote 22.

[a] Ratios calculated from spectral distribution.

[b] Ratios measured by method of gas equivalents (Astier and Allisy, 1959).

[c] HVL calculated from the spectral distribution of the primary beam.

[d] HVL calculated from the spectral distribution of the beam in a water phantom.

[e] Ratio calculated from theoretical spectral distribution of Kramers (1923).

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[22] For the purposes of calculation throughout this report the following percentage compositions by weight were assumed for muscle and bone (from data of Joyet *et al.* (1953), except for the omission of the small amount of Cl):

Element	Muscle (striated)	Compact bone (femur)
H	10.2	6.4
C	12.3	27.8
N	3.5	2.7
O	72.9	41.0
Na	0.08	-------
Mg	.02	0.2
P	.2	7.0
S	.5	0.2
K	.3	-------
Ca	.007	14.7

Air was taken to be 75.5 percent N_2, 23.2 percent O_2, and 1.3 percent A, by weight.

The values given in the literature for the calcium content of various bones vary greatly. For example, D'Ans and Lax (1949) give 20.2 percent Ca for the skull, 14.5 percent for femur, and 15.2 percent for ribs. Geygy (1955) gives 11 percent Ca for long bones and ribs. The data of Joyet *et al.* (1953) were arbitrarily chosen as roughly representative of bones in general. The errors resulting from this assumption are probably no greater than those resulting from (a) neglecting the scattered radiation in calculating f_{bone} (see table 8.2), and (b) ignoring the nonhomogeneous structure of bone.

[23] It will be assumed throughout this report, unless otherwise stated, that the cavity walls are thick enough to exclude any externally-produced electrons and that the cavity is small in comparison with the ranges of most of the electrons present. In the low-energy X-ray region, this latter requirement is difficult to fulfill, and the calibrated chamber method is preferable to the use of the cavity as an absolute device.

(ii) the X-ray intensity at the point is altered by the introduction of the chamber. The main cause of intensity change is usually the difference in X-ray attenuation in the chamber wall and in the layer of medium it displaces. Scattering of X-rays by the chamber and its connections may also contribute slightly to the change.

D_{med} and D_{wall} are related as follows:

$$D_{\text{med}}=A\frac{({}_m\mu_{en})_{med}}{({}_m\mu_{en})_{wall}}D_{\text{wall}}$$

$$=0.87_7\,Q\,(\tilde{s}_m)^{\text{wall}}_{\text{air}}\,A\frac{({}_m\mu_{en})_{med}}{({}_m\mu_{en})_{wall}} \qquad (8.8)$$

where A is a correction factor for the effects noted in (ii) above, which for a chamber of small wall thickness will have a value close to unity. A first approximation to $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$ is given in table 8.3[24] for H, C, N, O, P, Ca, and water relative to air (Nelms, 1956), not including density effect corrections. These data, labeled $(S_m)_{wall}/(S_m)_{air}$, are the actual ratios of mass stopping powers for monoenergetic electrons. They do not include modification of the electron energy for equilibrium nor the spectra of initial electron energies. For C, water, and a "tissue" composed of 13 percent of C and 87 percent of water, the $(S_m)_{wall}/(S_m)_{air}$ values are also given in table 8.3 with density effect corrections[25] (Sternheimer, 1956).

In table 8.4, the mean stopping power ratios, $(\bar{s}_m)^{\text{wall}}_{\text{air}}$, are given for C, water, and "tissue", relative to air. These were obtained by means of the formula at the bottom of the table. They apply to equilibrium spectra generated by monoenergetic initial energies, T_0.[26]

Tables 8.3 and 8.4 can be used approximately for protons also, by entering the tables at an energy 1/1836 that of the proton. For example,

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[24] The mean excitation potentials, $I=13Z$, were used in Bethe's stopping power formula for electrons (Segre, 1953). (Atomic binding corrections were not included.) This choice of I is based upon a recent survey of existing stopping power measurements (Caldwell, 1955; Sternheimer, 1956). However, it must be emphasized that an uncertainty of the order of 1 percent exists in s_m at the present time.

[25] Density effect calculations were not available for the other materials but the magnitude of this effect can be seen for C and H_2O in table 8.3. Density corrections are taken from Sternheimer (1956).

[26] s_m in eq. 8.7 and 8.8 may be evaluated at a mean energy for the spectrum of electrons crossing the cavity. Alternatively, s_m can be integrated as in table 8.4 to obtain the mean value $\bar{s}_m$. Strictly speaking, a further integration must be made over the initial spectrum of electrons produced by X- or gamma rays as has been done in table 8.5. However, it will suffice in many cases to take the mean initial electron energy from figure 8.2.

TABLE 8.3.—*Mass stopping power ratios*[a] *relative to air,* $(S_m)_{wall}/(S_m)_{air}$, *for monoenergetic electrons*

Electron kinetic energy, T	Without density effect							Including density effect		
	H	C	N	O	P	Ca	Water	C	Water	"Tissue"
Mev										
0.001	3.564	1.079	1.018	0.965	0.692	0.598	1.254	1.079	1.254	1.231
.002	3.223	1.062	1.014	.973	.752	.685	1.223	1.062	1.223	1.202
.003	3.084	1.055	1.013	.976	.776	.720	1.211	1.055	1.211	1.190
.004	3.004	1.051	1.012	.978	.791	.741	1.203	1.051	1.203	1.184
.005	2.949	1.049	1.011	.979	.800	.755	1.198	1.049	1.198	1.179
.006	2.904	1.045	1.010	.979	.806	.764	1.193	1.045	1.193	1.174
.007	2.877	1.045	1.011	.981	.813	.773	1.192	1.045	1.192	1.173
.008	2.851	1.044	1.010	.982	.818	.780	1.189	1.044	1.189	1.171
.009	2.830	1.043	1.010	.982	.821	.785	1.188	1.043	1.188	1.169
.01	2.811	1.042	1.010	.983	.825	.790	1.186	1.042	1.186	1.167
.02	2.709	1.037	1.009	.985	.843	.816	1.177	1.037	1.177	1.158
.03	2.661	1.034	1.008	.986	.851	.828	1.172	1.034	1.172	1.154
.04	2.630	1.033	1.008	.987	.857	.836	1.169	1.033	1.169	1.152
.05	2.609	1.032	1.008	.987	.860	.841	1.168	1.032	1.168	1.150
.06	2.592	1.031	1.008	.988	.863	.846	1.166	1.031	1.166	1.148
.07	2.579	1.030	1.008	.988	.866	.849	1.165	1.030	1.165	1.147
.08	2.568	1.030	1.007	.988	.868	.852	1.164	1.030	1.164	1.146
.09	2.559	1.029	1.007	.988	.869	.854	1.163	1.029	1.163	1.146
.1	2.550	1.029	1.007	.989	.871	.856	1.162	1.029	1.162	1.144
.2	2.502	1.027	1.007	.990	.879	.869	1.158	1.023	1.158	1.140
.3	2.476	1.025	1.006	.990	.884	.875	1.155	1.019	1.155	1.138
.4	2.458	1.024	1.006	.991	.887	.880	1.154	1.014	1.154	1.136
.5	2.444	1.024	1.006	.991	.889	.883	1.153	1.010	1.152	1.134
.6	2.433	1.023	1.006	.991	.891	.886	1.152	1.007	1.148	1.130
.7	2.424	1.023	1.006	.992	.893	.888	1.151	1.003	1.144	1.126
.8	2.417	1.022	1.006	.992	.894	.890	1.150	1.000	1.140	1.122
.9	2.410	1.022	1.006	.992	.896	.892	1.150	0.997	1.134	1.116
1.0	2.404	1.022	1.006	.992	.897	.894	1.149	.994	1.133	1.115
2	2.366	1.020	1.005	.993	.903	.903	1.146	.971	1.105	1.088
3	2.347	1.019	1.005	.993	.907	.908	1.144	.954	1.086	1.069
4	2.333	1.018	1.005	.994	.909	.912	1.143	.942	1.071	1.054
5	2.324	1.018	1.005	.994	.911	.914	1.142	.932	1.059	1.042
6	2.316	1.017	1.005	.994	.912	.916	1.141	.923	1.049	1.032
8	2.305	1.017	1.005	.994	.914	.919	1.140	.909	1.032	1.016
10	2.297	1.016	1.004	.995	.916	.921	1.139	.898	1.019	1.003

[a] Mean excitation potentials, I, assumed equal to $13Z$ (see footnote [23]). Courtesy of A. Nelms (1956).

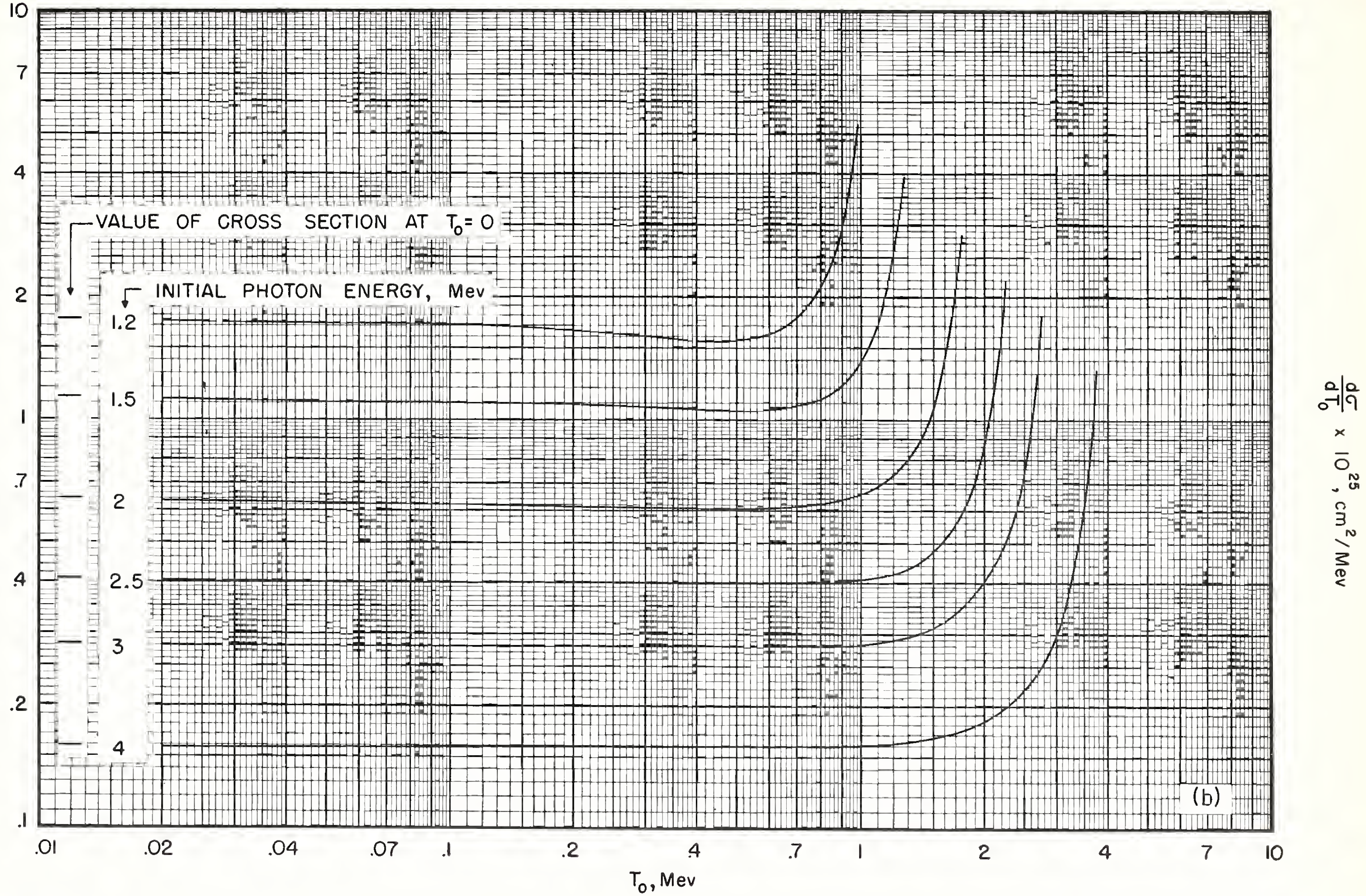

FIGURE 8.3b. *Starting-energy distributions for Compton-recoil electrons produced by photons with initial energies of 1.2 to 4 Mev (Nelms, 1953).* The ordinate is the absolute differential cross section for giving a free electron a recoil energy in the interval from T_0 to T_0+dT_0.

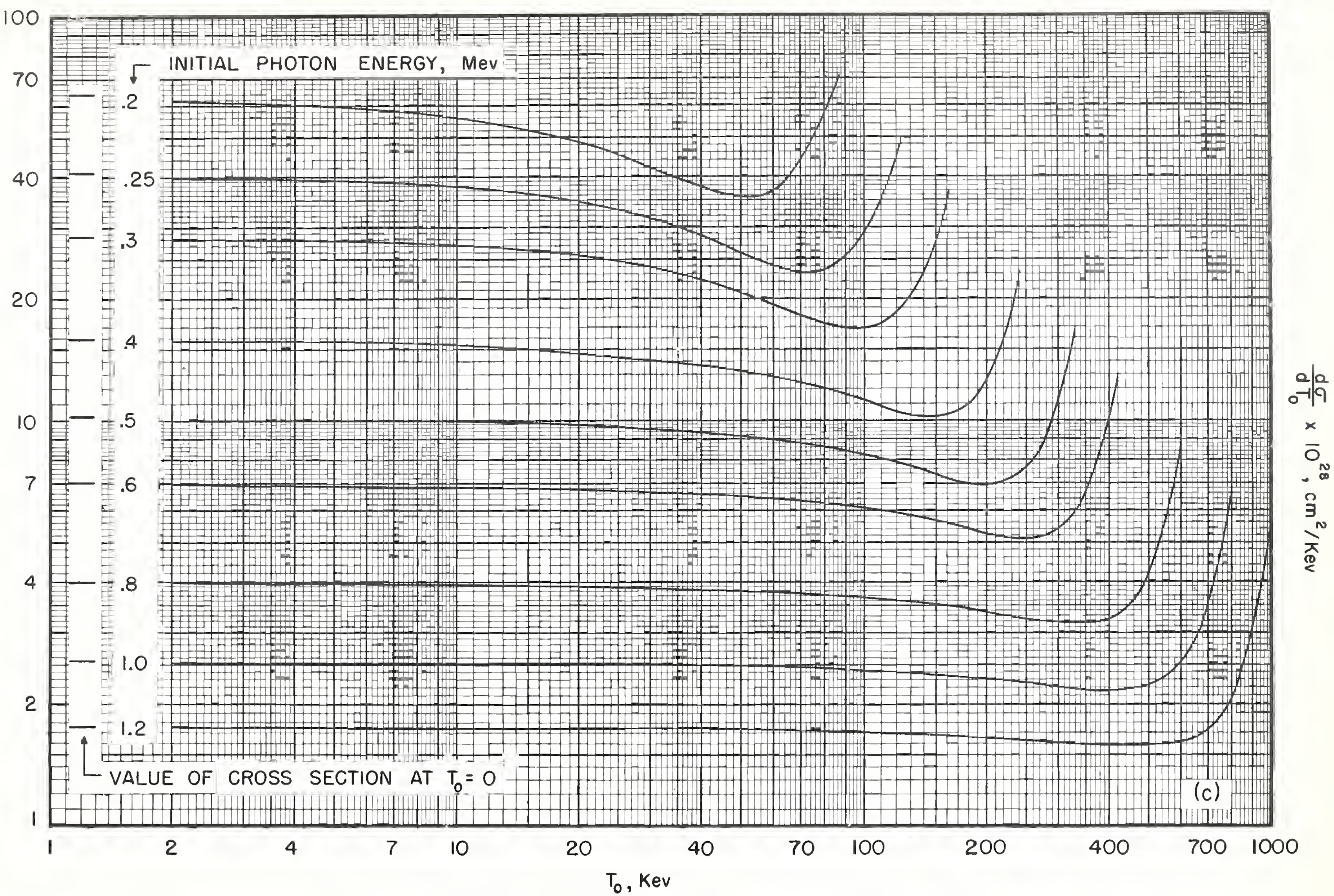

FIGURE 8.3c. *Starting-energy distributions for Compton-recoil electrons produced by photons with initial energies of 0.2 to 1.2 Mev (Nelms, 1953).*
The ordinate is the absolute differential cross section for giving free electron a recoil energy in the interval from T_0 to $T_0 + dT_0$.

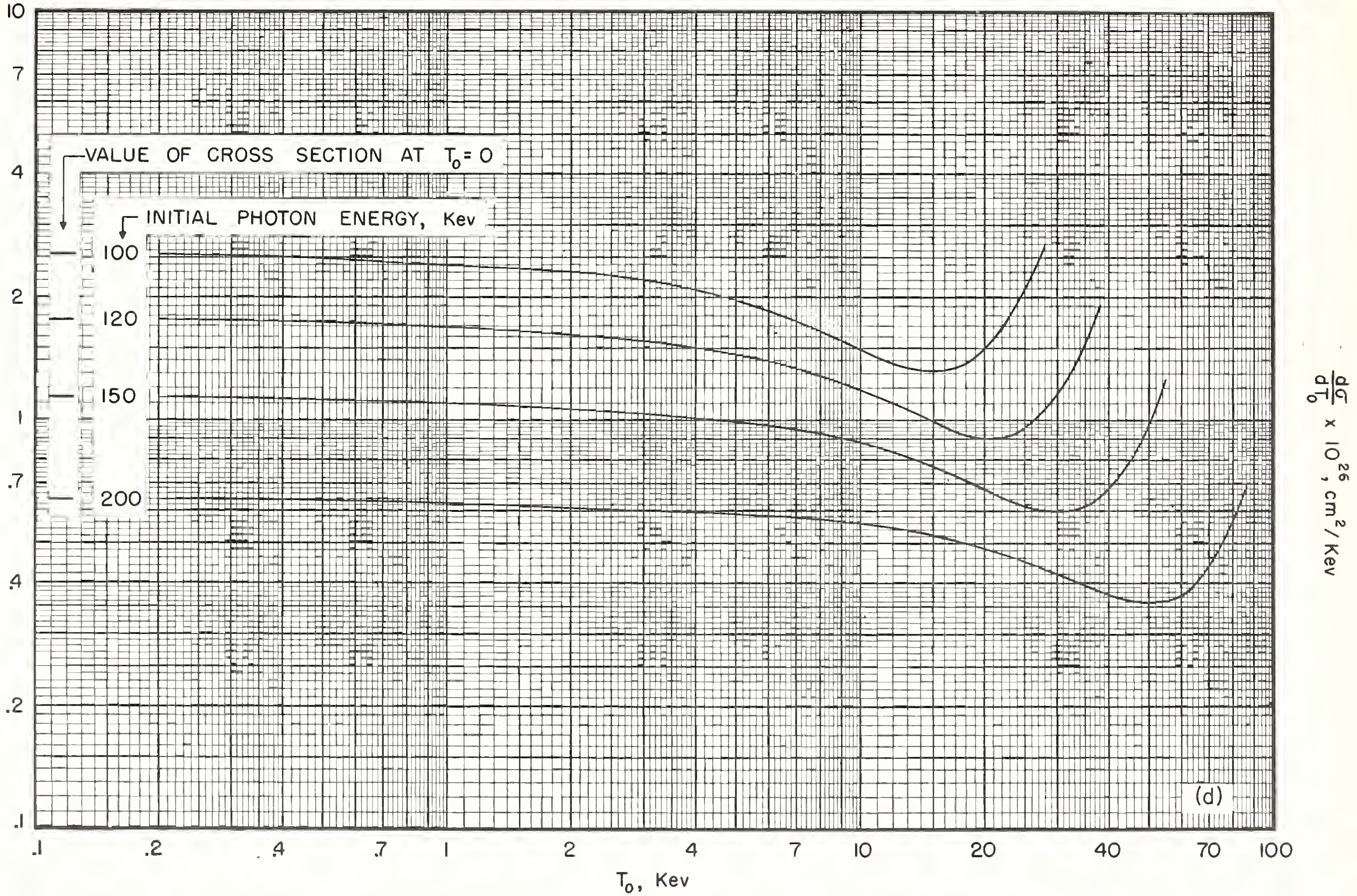

FIGURE 8.3d. *Starting-energy distributions for Compton-recoil electrons produced by photons with initial energies of 100 to 200 kev (Nelms, 1953).*
The ordinate is the absolute differential cross section for giving free electron a recoil energy in the interval from T_0 to T_0+dT_0.

an 18.36-Mev proton has the same velocity (and hence is subject to nearly the same stopping power) as a 10-kev electron.

In table 8.5a, the mean stopping power ratios, $(\bar{s}_m)^{\text{wall}}_{\text{air}}$, for C, water, and "tissue" relative to air, are given for equilibrium spectra generated by the Compton electrons initiated by the γ-rays of Co^{60} and Cs^{137}, including density effect. These were obtained from the following equation:

$$\frac{1}{(\bar{s}_m)^{\text{wall}}_{\text{air}}}=\frac{\int_0^{T_e} A(T_0,h\nu)\times T_0\times\frac{dT_0}{[\bar{s}_m(T_0)^{\text{wall}}_{\text{air}}]}}{\int_0^{T_e} A(T_0,h\nu)\times T_0 dT_0}, \qquad (8.9)$$

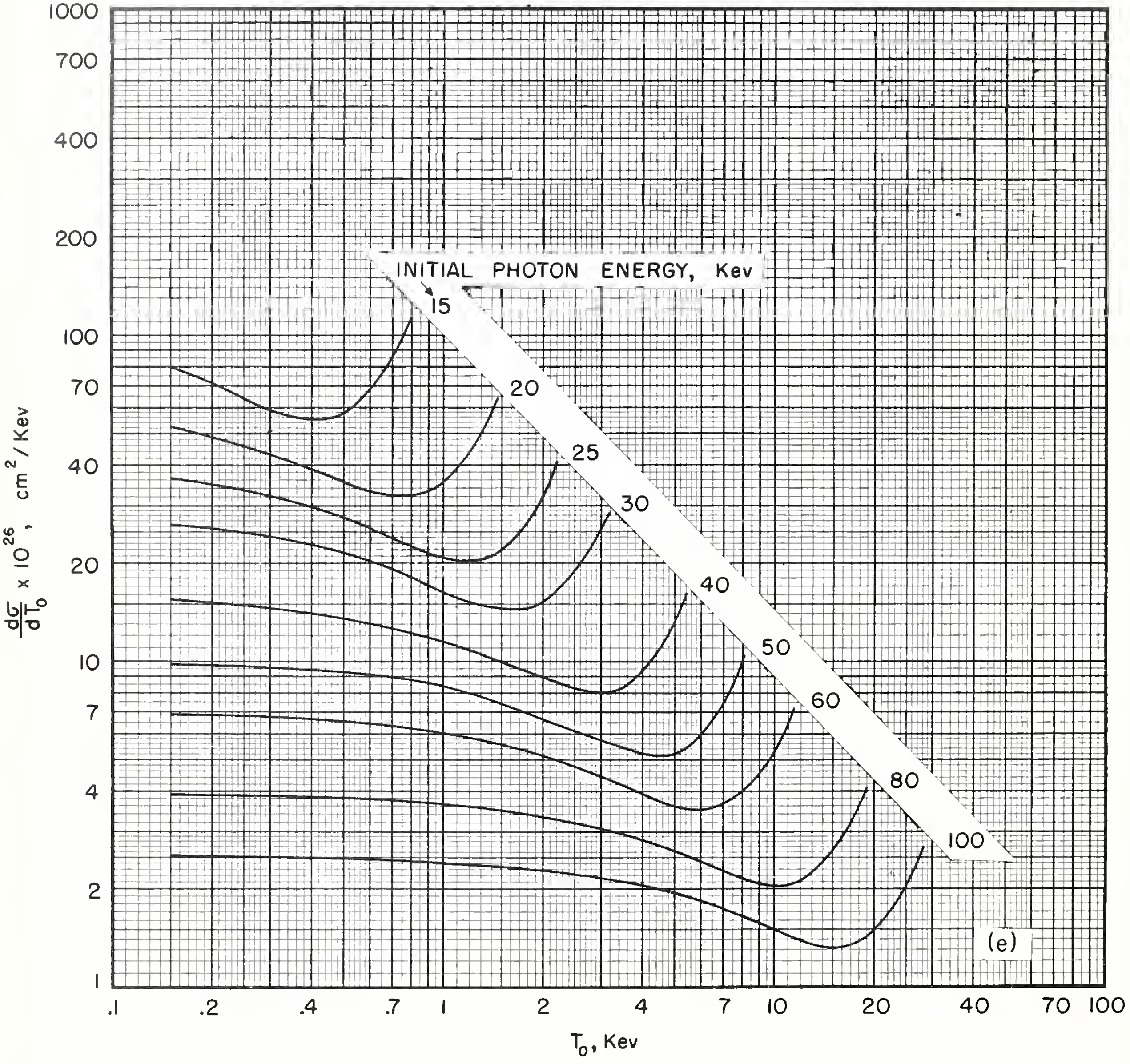

FIGURE 8.3e. *Starting-energy distributions for Compton-recoil electrons produced by photons with initial energies of 15 to 100 kev (Nelms, 1953).*

The ordinate is the absolute differential cross section for giving free electron a recoil energy in the interval from T_0 to T_0+dT_0.

TABLE 8.4.—*Mean mass stopping power ratios relative to air,*[a] $(\bar{s}_m)^{\text{wall}}_{\text{air}}$ *for the equilibrium electron spectra generated by monoenergetic initial electron energies,* T_0

Initial electron kinetic energy, T_0	Including density effect		
	C	Water	"Tissue"
Mev			
0.002	1.070	1.238	1.216
.003	1.064	1.226	1.205
.004	1.060	1.220	1.199
.005	1.058	1.215	1.195
.006	1.055	1.212	1.191
.007	1.054	1.208	1.188
.008	1.052	1.206	1.186
.009	1.051	1.203	1.183
.01	1.050	1.202	1.182
.02	1.044	1.191	1.172
.03	1.041	1.185	1.166
.04	1.039	1.181	1.163
.05	1.038	1.179	1.160
.06	1.037	1.177	1.159
.07	1.036	1.175	1.157
.08	1.035	1.174	1.156
.09	1.034	1.173	1.155
.1	1.034	1.172	1.154
.2	1.030	1.166	1.148
.3	1.027	1.163	1.145
.4	1.024	1.161	1.143
.5	1.022	1.159	1.141
.6	1.020	1.158	1.140
.7	1.017	1.156	1.138
.8	1.016	1.154	1.136
.9	1.014	1.152	1.134
1.0	1.012	1.150	1.132
2	1.001	1.139	1.121
3	0.985	1.121	1.103
4	.976	1.110	1.093
5	.968	1.108	1.084
6	.961	1.093	1.076
8	.950	1.080	1.063
10	.940	1.069	1.052

[a] Mean excitation potentials, I, assumed equal to $13Z$ (see footnote [24]).

$$\frac{1}{(\bar{s}_m)^{\text{wall}}_{\text{air}}}=\frac{1}{T_0}\int_0^{T_0}\frac{(S_m)_{\text{air}}}{(S_m)_{\text{wall}}}\,dT.$$ Courtesy of A. Nelms (1956).

TABLE 8.5a.—*Mean mass stopping power ratios,* $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$ *for equilibrium electron spectra generated by* Co^{60} and Cs^{137} γ-*rays, with the assumption that the electrons slow down in a continuous manner*

γ-ray source	Graphite	Water	"Tissue"
Co^{60}	1.01_7	1.15_5	1.13_7
Cs^{137}	1.02_7	1.16_2	1.14_5

TABLE 8.5b.—*Absorbed dose versus air cavity ionization*[a]

(Rad in wall medium per esu/0.001293g in air cavity)

γ-ray source	Wall material		
	Graphite	Water	"Tissue"
Co^{60}	0.89_1	1.01_3	0.99_7
Cs^{137}	$.90_0$	1.01_9	1.00_4

[a] These figures are equal to the corresponding figures in table 8.5a multiplied by 0.87_7 rad/r. Courtesy of A. Nelms (1956).

where $A(T_0, h\nu)$ is the numerical spectrum of Compton electrons having energies T_0 at production, for Co^{60} (γ-rays 1,250 kev) or Cs^{137} (662 kev). (Similar spectra for other γ-ray energies are shown in figures 8.3*a* to *f*.) T_e is the maximum value of T_0 in the spectrum, and $(\bar{s}_m)^{\text{wall}}_{\text{air}}$ is the mean mass stopping power ratio as a function of T_0, taken from table 8.4. Table 8.5b lists values of rad/(esu/0.001293g), the ratio of absorbed dose in the wall material to the ionization in an air cavity at NTP, which were obtained from table 8.5a by multiplying the corresponding figures by 0.87_7. Here the absorbed dose is that for a thin layer surrounding the air cavity.

In calculating $\bar{s}_m$ and $\tilde{s}_m$ in tables 8.4 and 8.5, it was assumed that the electrons traversing the wall and the cavity slow down in a continuous manner, without the production of energetic secondary electrons (delta rays). A modified treatment theory has recently been proposed, which takes into account those secondaries which are produced, (Spencer and Attix, 1955). This treatment has been found experimentally (Whyte, 1957; Greening, 1957; Attix and Ritz, 1957; Attix, DeLaVergne, and Ritz, 1958) to provide a better approximation. The cavity formula resulting from this treatment is the same as eq 8.1, except that s_m now includes the secondary electron effects and is a function of the cavity size.

Table 8.6 contains a tabulation of values of $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$ obtained with the modified treatment, for several wall materials and γ-ray energies. Δ is the energy needed by an electron just to cross a cavity of the size indicated (containing air at NTP). For comparison, the corresponding values of $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$ obtained by assuming continuous slowing-down (labeled "continuous slowing-down") are also given. This tabulation shows:

(a) For nearly air-equivalent materials (e.g., graphite) the difference between the two sets of results is small enough to be negligible for nearly any practical application.

(b) For walls of higher atomic number the differences become larger. Although the "continuous slowing down" results are meant to be applied to an infinitesimal cavity, they agree more closely with the Spencer-Attix modification at larger cavity sizes.

Calculations of electron spectra (including secondaries) are not yet available for low electron starting energies (T_0). Thus this theory has not been applied to lower quantum energies than that of Au^{198} (411.8 kev).

(c) Additional information which may be found useful for the evaluation of absorbed dose from X- and gamma radiation are given in figures 8.2, 8.3, 8.5, and in table 8.7.[27] Figures 8.7 *a* to *c* provide some information on LET.

For many purposes it will be sufficient to make the following approximations:

(1) The mean photon energy, $h\bar{\nu}$, is numerically equal to 40 to 45 percent of the kilovoltage applied to the tube in the case of moderate filtration.

(2) The initial energy of the photoelectron $\approx h\bar{\nu}$. The mean initial energy of the recoil electrons is $(\sigma_a/\sigma)h\bar{\nu}$, where σ_a and σ are the "true" Compton absorption coefficient and the total Compton coefficient, respectively, of radiation of quantum energy $h\bar{\nu}$. A graph of $(\sigma_a/\sigma)h\bar{\nu}$ is given in figure 8.2.

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[27] See section 8.9c for an explanation of this table.

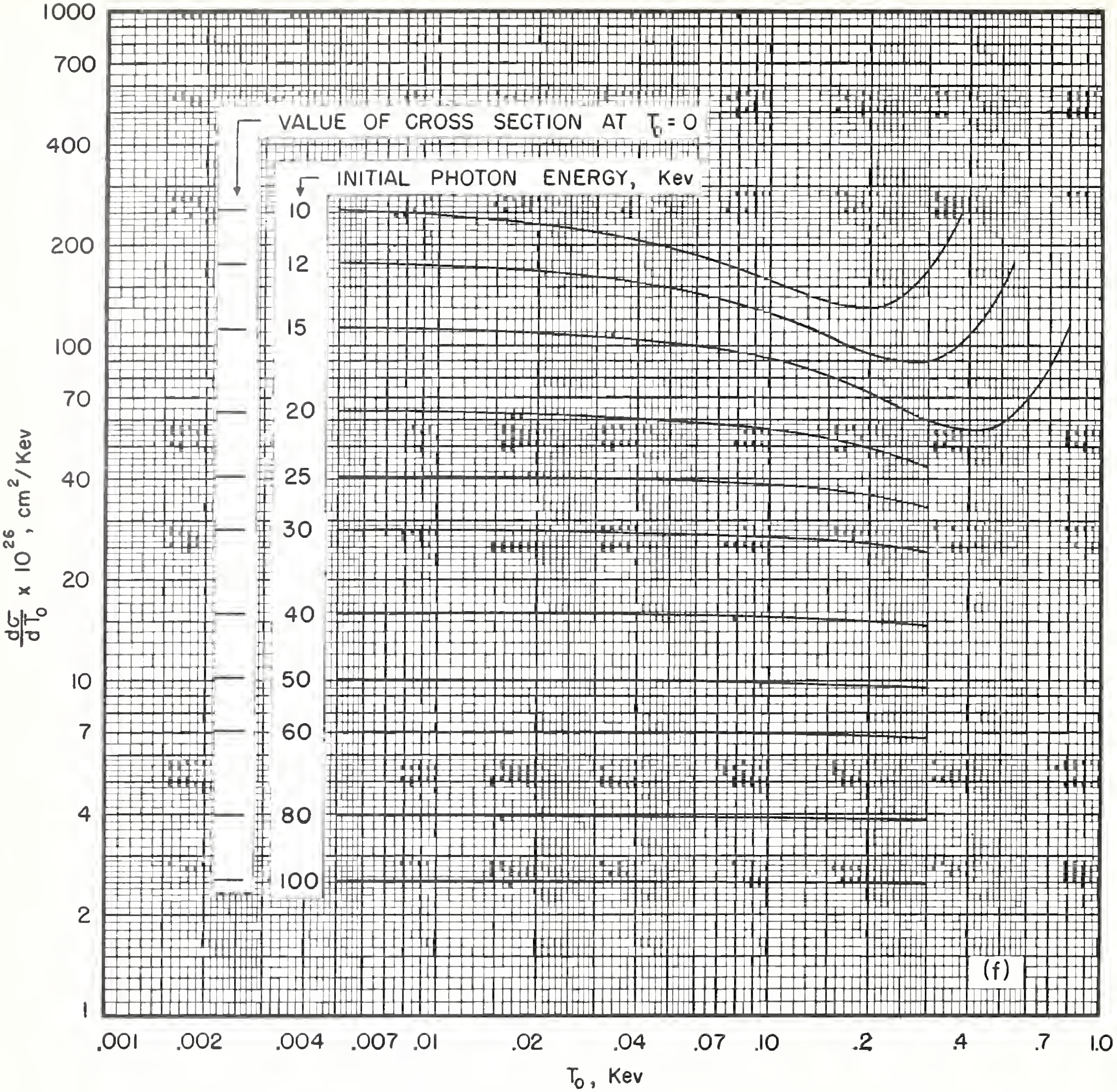

FIGURE 8.3f. *Starting-energy distributions for Compton-recoil electrons produced by photons with initial energies of 10 to 100 kev (Nelms, 1953).*

The ordinate is the absolute differential cross section for giving free electron a recoil energy in the interval from T_0 to T_0+dT_0.

The mean energy of the electrons and positrons resulting from pair production can be taken as $\approx(h\bar{\nu}/2-0.511)$Mev.

(3) The mean instantaneous electron energy (for purposes of calculating stopping power ratios $(\bar{s}_m)_{\text{air}}^{\text{wall}}$ is approximately 40 percent of the initial electron energy, T_0.

Case II. Electronic equilibrium conditions not satisfied (e.g., near tissue-air or bone-tissue boundaries).[28]

The ionization in this case is measured at the place of interest in an air-filled ionization chamber having walls that are very thin compared with

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[28] It should be pointed out that in such a region the spectral distribution of the electrons is not in equilibrium, and hence is not given by the reciprocal of the stopping power in the medium (neglecting δ-rays), as is usually assumed (Spencer and Attix, 1955). However, because s_m for near-air-equivalent materials does not vary rapidly with energy, it can be evaluated approximately by assuming that the equilibrium conditions apply (see section 8.2).

TABLE 8.6.—*Mean mass stopping power ratios,* ($\bar{s}_m$) (wall/air), *for equilibrium spectra of electrons generated by* Co^{60}, Cs^{137} *and* Au^{198} *γ-rays*[a]

Wall material	γ-ray source	Continuous slowing down	Spencer-Attix modification				
		Cavity size: Δ, cm ~0 Kev ~0	0.015 2.5	0.051 5.1	0.19 10.2	0.64 20.4	2.2 40.9
C	Au^{198} (0.411 Mev)	1.032	1.041	1.038	1.036	1.033	1.031
	Cs^{137} (0.670 Mev)	1.027	1.036	1.033	1.031	1.029	1.027
	Co^{60} (1.25 Mev)	1.017	1.027	1.024	1.022	1.020	1.018
Al	Au^{198}	0.878	0.852	0.862	0.870	0.876	0.882
	Cs^{137}	.886	.858	.868	.875	.881	.886
	Co^{60}	.892	.862	.871	.878	.883	.888
Cu	Au^{198}	.713	.653	.678	.696	.710	.722
	Cs^{137}	.733	.668	.692	.710	.723	.734
	Co^{60}	.754	.681	.704	.721	.734	.745
Sn	Au^{198}	.566	.497	.529	.552	.569	.584
	Cs^{137}	.602	.523	.555	.579	.596	.611
	Co^{60}	.634	.544	.576	.599	.617	.631
Pb	Au^{198}	.469	.412	.440	.460	.477	.491
	Cs^{137}	.506	.439	.469	.490	.507	.522
	Co^{60}	.541	.461	.491	.513	.530	.544

[a] Mean excitation potentials, I, assumed equal to 13Z.

TABLE 8.7.—*Approximate electron equilibrium spectral distributions in tissue*

Electron kinetic energy *T* [a]	Electron spectral distribution, neglecting δ-rays	Electron spectral distribution, including δ-rays (g/cm²)/Mev			Electron kinetic energy *T* [a]	Electron spectral distribution, neglecting δ-rays	Electron spectral distribution, including δ-rays (g/cm²)/Mev		
		T_0 [b] = 1.308 Mev	T_0 = 0.654 Mev	T_0 = 0.327 Mev			T_0 [b] = 1.308 Mev	T_0 = 0.654 Mev	T_0 = 0.327 Mev
Mev	*(g/cm²)/Mev*				*Mev*	*(g/cm²)/Mev*			
0.001	0.00825	----	----	----	0.1	0.244	0.346	0.283	0.251
.002	.01323	----	----	----	.2	.360	.410	.371	.360
.003	.01770	0.474	0.257	0.138					
.004	.0219	.439	.237	.129	.3	.427	.453	.427	.427
.005	.0260	.410	.225	.124	.4	.469	.480	.469	
					.5	.495	.498	.495	
.006	.0298	.394	.218	.121	.6	.514	.514	.514	
.007	.0336	.383	.212	.119	.7	.527	.527		
.008	.0372	.371	.206	.118					
.009	.0408	.361	.202	.117	.8	.535	.535		
.01	.0443	.355	.198	.116	.9	.542	.542		
					1	.546	.546		
.02	.0760	.315	.187	.127	2	.551			
.03	.1038	.303	.197	.144	3	.543			
.04	.1289	.305	.209	.160					
.05	.1518	.310	.223	.176	4	.535			
.06	.1730	.317	.235	.194	5	.528			
					6	.522			
.07	.1926	.324	.247	.208	8	.514			
.08	.211	.331	.259	.224	10	.507			
.09	.228	.337	.270	.237	----	----			

[a] For any value of *T* in column 1 taken for the starting energy T_0, the second column gives the relative electron spectral distribution (without δ-rays) at lower values of *T*, the degraded energy of the electrons. Columns 3, 4, and 5 give the relative spectral distributions for starting energies of 1.308, 0.654, and 0.327 Mev, respectively, including δ-rays.

[b] T_0 is the initial energy of the secondary electrons.

the range of the secondary electrons at the place of interest. If the walls are so thin that they make a negligible contribution to the total secondary electron emission that ionizes the air in the chamber, the material of which the walls are composed is not important. But because this condition is difficult to fulfill, it may be found to be advantageous to make the walls of a material that fairly closely matches in composition the immediately surrounding medium. The depth of the ionization chamber should be small in the direction of the gradient of the absorbed dose (see section 3.5).

If a charge *Q* esu is carried by the ions of either sign generated per 0.001293 g of air, then applying the general cavity relation

$$D_{\text{medium}}=0.87_7(s_m)_{\text{air}}^{\text{medium}}\times Q \text{ rads.} \qquad (8.10)$$

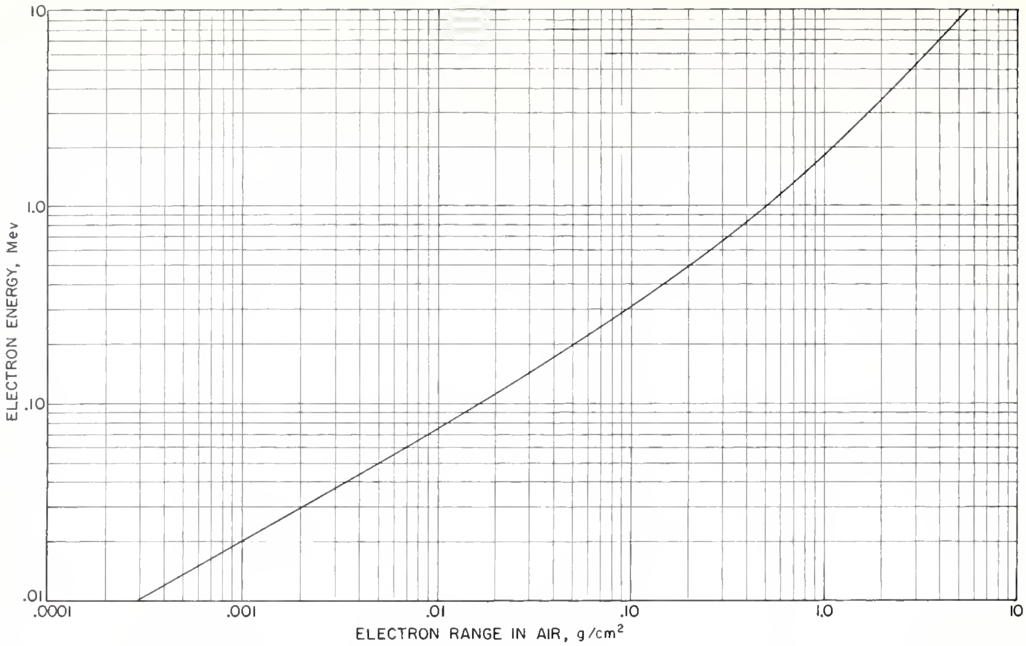

FIGURE 8.4. *Approximate range-versus-energy curve for electrons in air, calculated by Nelms (1956).*

8.5. *Absorbed dose in tissue or other material exposed to fast electrons.*

For electrons, as for X-rays, the condition of electronic equilibrium may or may not be satisfied. These cases will again be discussed separately.

Case I. Electronic equilibrium. If a β-ray emitter is homogeneously distributed throughout a medium in which the absorbed dose is to be measured, the conditions for electronic equilibrium will be almost ideally satisfied except near the boundaries of the medium. However, contrary to the above procedures for X-rays, a thin-walled ionization chamber should be used, as in the nonequilibrium X-ray case. The chamber walls merely act as electron absorbers, and should be minimized in thickness. The thickness, furthermore, should be varied in order to extrapolate the ionization current to zero thickness.

It is sometimes possible to eliminate the inserted cavity wall entirely, enclosing the air within walls of the medium itself. The inner surface of these walls must be made electrically conducting (if not already so) by a thin coating of a conducting material (e.g., graphite).

Equation (8.10) is used in calculating the absorbed dose, $(\bar{s}_m)^{\text{wall}}_{\text{air}}$, being evaluated from table 8.4. Depending upon the degree of accuracy desired, $(\bar{s}_m)^{\text{wall}}_{\text{air}}$ can either be taken for an average T_0 value appropriate to the β-ray source within the medium, or integrated according to eq (8.9) over the initial β-ray spectrum, $A(T_0)$.

Case II. Electronic nonequilibrium. If a medium is exposed to an externally applied beam of electrons, the electron flux and spectrum will certainly vary from point to point within the medium, particularly in depth. Thus, the electronic equilibrium conditions are not satisfied.

Here, again, a very thin- (or zero-) walled chamber should be employed. However, one cannot assume that the spectral distribution is the same as that for equilibrium so values of the stopping power ratio given in table 8.4 do not apply. Instead one must average the stopping power ratios for table 8.3 over the electrum spectrum at the place of interest to obtain the stopping power ratio, $(\hat{s}_m)^{\text{wall}}_{\text{air}}$ to be used. Near the surface of a material of low atomic number, most of the electrons traversing a small cavity still have energies close to their incident energy. Farther in they will have a lower average energy, which may be estimated by means of figure 8.4 for low atomic-number materials. Equation (8.7) is used to relate the energy absorbed per gram of the medium adjacent to the chamber to the ionization in the air (substitute $(\hat{s}_m)^{\text{wall}}_{\text{air}}$ in the equation for $(\tilde{s}_m)^{\text{wall}}_{\text{air}}$).

8.6. *Absorbed dose in tissue exposed to fast neutron radiation.*

The principles set out in section 8.1 for calculating absorbed dose from cavity chamber measurements are fully applicable to the determination of fast neutron doses by ionization methods.

The range, L, of the recoil protons (in air, 15 °C, 1 atm) of energies, E, between 0.1 and 10 Mev is given to a good approximation by

$$L=\frac{5}{4}E+E^2 \qquad (8.11)$$

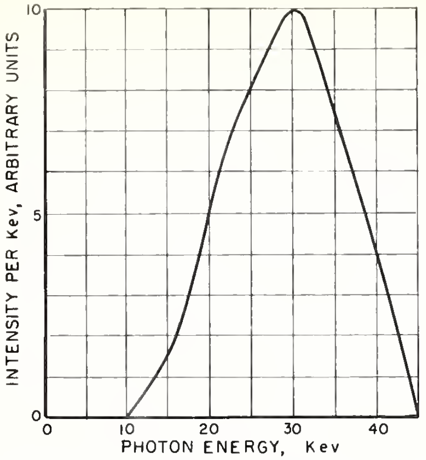

Figure 8.5a. *X-ray spectrum obtained with 45-kv tube potential, 0.7-mm Al inherent filtration, HVL=0.9 mm Al (Hettinger and Starfelt, 1958).*

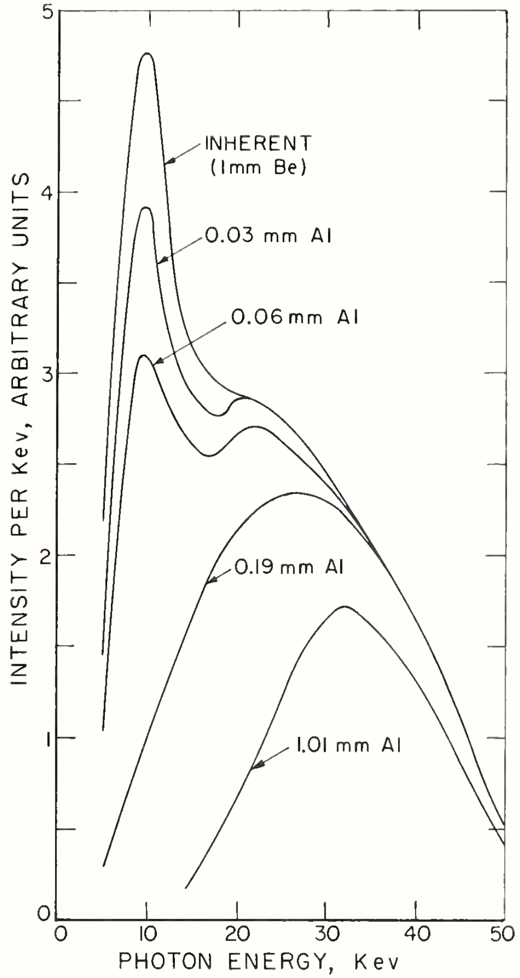

Figure 8.5b. *X-ray spectra obtained with 50-kv tube potential, an inherent filtration of 1 mm Be and the indicated added filters of aluminum. (Ehrlich, 1956)*

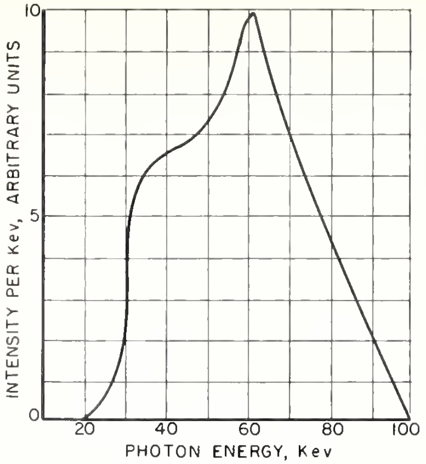

Figure 8.5c. *X-ray spectrum obtained with 100-kv tube potential, 4 mm Al inherent filtration, HVL=0.18 mm Cu (Hettinger and Starfelt, 1958).*

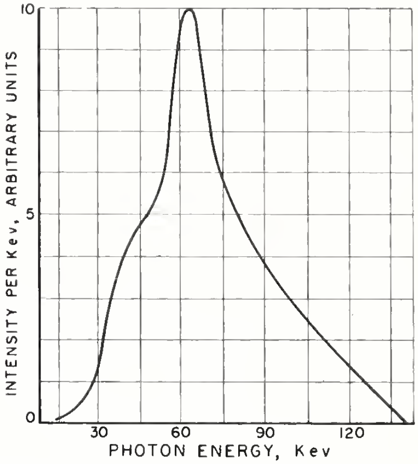

Figure 8.5d. *X-ray spectrum obtained with 140-kv tube potential, 4 mm Al inherent filtration, HVL=0.30 mm Cu (Hettinger and Starfelt, 1958).*

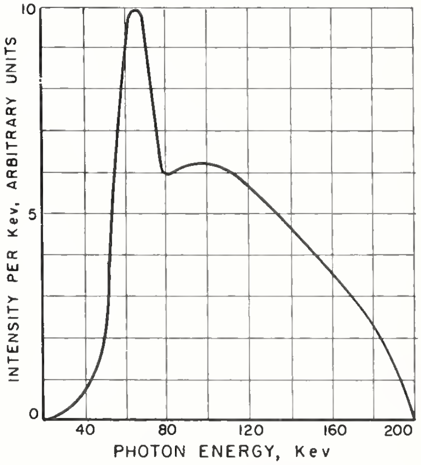

Figure 8.5e. *X-ray spectrum obtained with 200-kv tube potential, 2.5 mm Al inherent filtration, added 1.0 mm Al and 0.5 mm Cu. HVL=1.2 mm Cu (Helle).*

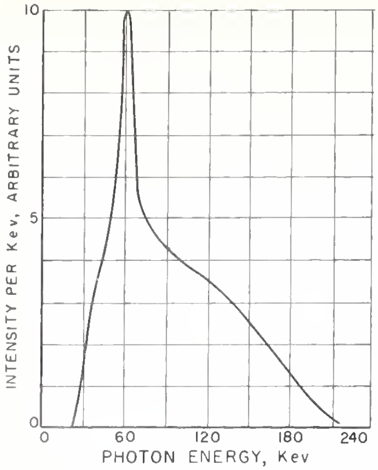

Figure 8.5f. *X-ray spectrum obtained with 220-kv tube potential, inherent filtration 4 mm Al, HVL=0.55 mm Cu (Hettinger and Starfelt, 1958).*

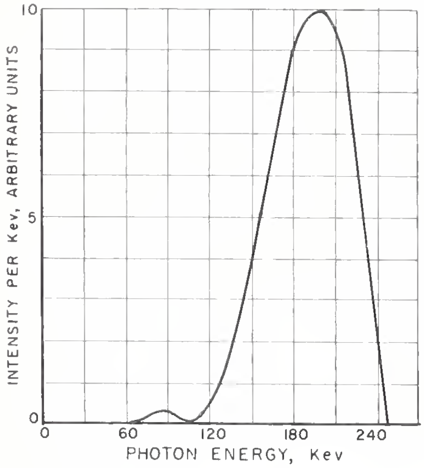

Figure 8.5g. *X-ray spectrum obtained with 250-kv tube potential, 4 mm Al inherent filtration, added 1.32 mm Al, 2.07 mm Cu, 1.10 mm Cd, 1.34 mm Pb, HVL=4.8 mm Cu (Hettinger and Starfelt, 1958).*

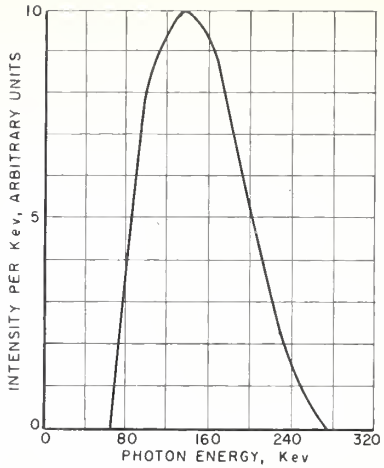

Figure 8.5h. *X-ray spectrum obtained with 280-kvp tube potential, with filtration to give HVL of 2.5 mm Cu (Cormack et al., 1955).*

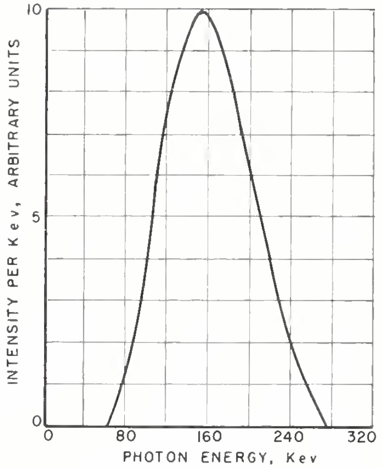

Figure 8.5i. *X-ray spectrum obtained with 280-kvp tube potential, with filtration to give HVL of 3.1 mm Cu (Cormack et al., 1955).*

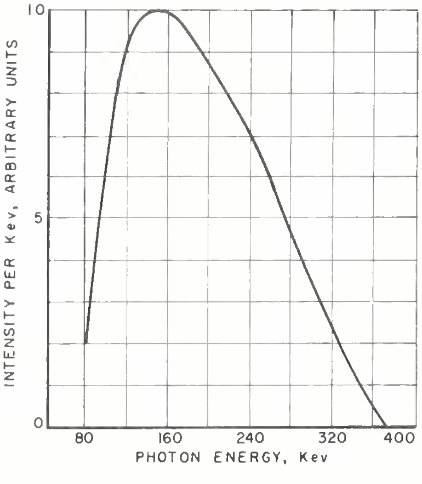

Figure 8.5j. *X-ray spectrum obtained with 400-kv tube potential, added filtration of 1 mm Sn, 0.5 mm Cu, and 1 mm Al; HVL=4.0 mm Cu (Johns et al., 1952).*

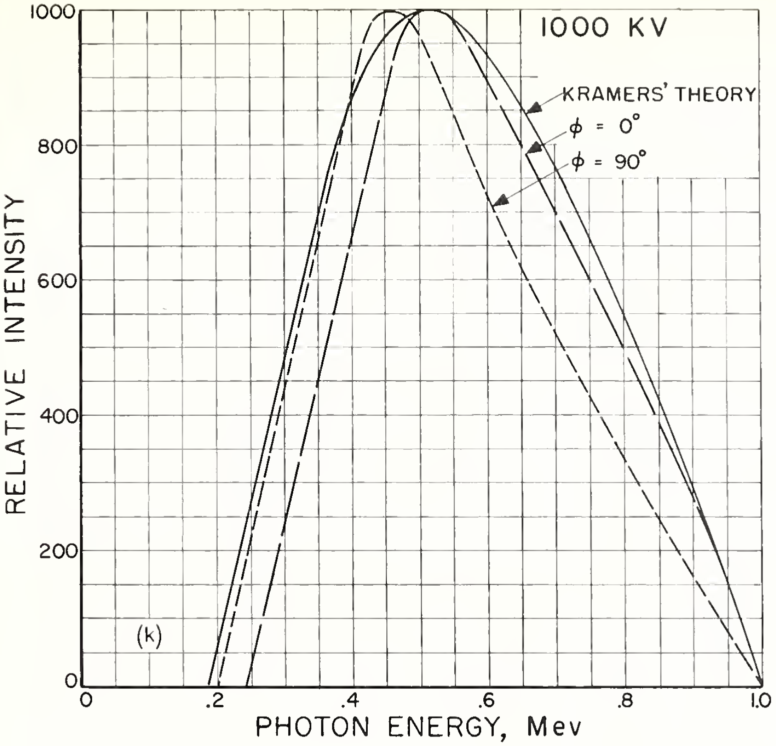

FIGURE 8.5k. *X-ray spectra obtained at 1000-kv tube potential.*

Filtration: 2.8-mm W+2.8-mm Cu+18.7-mm water+2.1-mm brass. Dashed curves: spectra measured by Compton spectrometer (the ordinate is photon energy multiplied by number of photons of that energy) (Miller et al., 1954). Angles refer to direction of X-rays relative to initial electron direction. Solid curve: spectrum calculated by Kramers' method (1923).

The average energy of the recoil protons is one-half the neutron energy producing them. Thus, for example, 2 Mev neutrons will give rise to recoil protons having ranges averaging about 2.25 cm in air.

For neutron energies up to 10 Mev roughly 90 percent of the neutron absorbed dose in tissue is due to recoil protons set in motion by collisions between the fast neutrons and hydrogen atoms. Hence, it is advantageous to use ionization chambers made of hydrogen-rich materials approximating tissue.

A rigid envelope may be lined with a suitable thickness of a gelatin mixture rather accurately reproducing the composition of wet tissue (Rossi and Failla, 1950). Such chambers can be entirely satisfactory if either polytetrafluoroethylene, or amber coated with ceresin, is used for insulation, despite the fact that the water vapor pressure in the chamber is high. Highly polished polystyrene insulators are also satisfactory. It is, however, generally more convenient to use a conducting plastic,[29] rather than gelatin for the chamber wall.

In practice the choice of gas to use in the cavity lies between air and a gas matching the hydrogenous wall as closely as possible in composition, e.g., ethylene in polyethylene. In the former case, it is customary to assume $W'_{\text{air}}=35$ ev$=5.6\times10^{-11}$ erg, where the prime by W indicates that the ionizing particles are protons. It should be borne in mind here that the cavity size restrictions are very stringent for neutrons because of the short ranges of the recoil protons. Cavity sizes should be less than about $\frac{1}{10}$ of the proton range in air, or 2mm at NTP for 2-Mev neutrons. An even more serious drawback encountered in using air as the cavity gas with tissue-equivalent walls is that the stopping power ratio, s_m, is not well known in the energy range of interest. For example, a 10-Mev neutron gives rise to protons with an average initial energy of 5 Mev, being subject to a stopping power roughly equivalent to that of a $T_0=3$ kev electron. At such an energy the Bethe stopping-power formula becomes less reliable because of atomic-binding effects, and becomes useless at somewhat lower energies. The stopping-power ratio $(\bar{s}_m)^{\text{wall}}_{\text{air}}$ the result of an integration from 0 to T_0, hence it will have little meaning for a T_0 as low as 3 kev for an electron. Table 8.4 gives values of $(\bar{s}_m)^{\text{wall}}_{\text{air}}$ only for $T_0\geq 20$ kev for electrons (or an initial energy of 36 Mev for protons). Thus, it is not applicable to neutron dosimetry except at very high energies. One can resort instead to experimentally determined proton stopping powers at low energies (Segre, 1953; Reynolds, et al., 1953).

Matching the walls and gas in composition avoids both the above difficulties, because not only is the cavity size restriction removed but

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[29] Such plastics having an atomic composition of approximately 10.1 percent of hydrogen, 3.5 percent of nitrogen, and the remainder carbon, are available commercially and may be molded into a variety of shapes.

$(\hat{s}_m')^{\text{wall}}_{\text{gas}}=1$ as well.[30] It is, however, necessary to know W' for the gas in question.

(a) *Air-filled chamber made of conducting plastic.* Measure the ionization when the chamber is exposed to the neutron beam within the tissue of interest or a phantom representing this tissue. Let the ions of either sign which are produced in the chamber (at NTP) carry a charge Q esu/cm³.

The absorbed dose in the wall material can be calculated from the relation

$$D_{\text{wall}}=Q\times 0.90(\hat{s}_m')^{\text{wall}}_{\text{air}}\ \text{rad},^{31} \quad (8.12)$$

where $(\hat{s}_m')^{\text{wall}}_{\text{air}}$ is the weighted average stopping power ratio for all recoils. The absorbed dose in tissue then becomes

$$D_{\text{tissue}}=D_{\text{wall}}\times\frac{(\Sigma\sigma_i k_i N_i)_{\text{tissue}}}{(\Sigma\sigma_i k_i N_i)_{\text{wall}}}, \quad (8.13)$$

where σ_i is the scattering cross section of the ith kind of atom, k_i is the average fractional loss of energy with the ith kind of atom, and N_i is the number per unit mass of the ith kind of atom.

Table 8.8 gives values of $(\sigma\times k)$ for H, C, N, O, S, and P for neutron energies between 1/10 and 10 Mev (Hughes and Harvey, 1955).

(b) *Gas and walls of matched composition.* Measure the ionization when the chamber is exposed to the neutron beam within the tissue of interest or in a phantom representing this tissue. Let the ions of either sign that are produced in the chamber (at NTP) carry a charge Q esu/cm³.

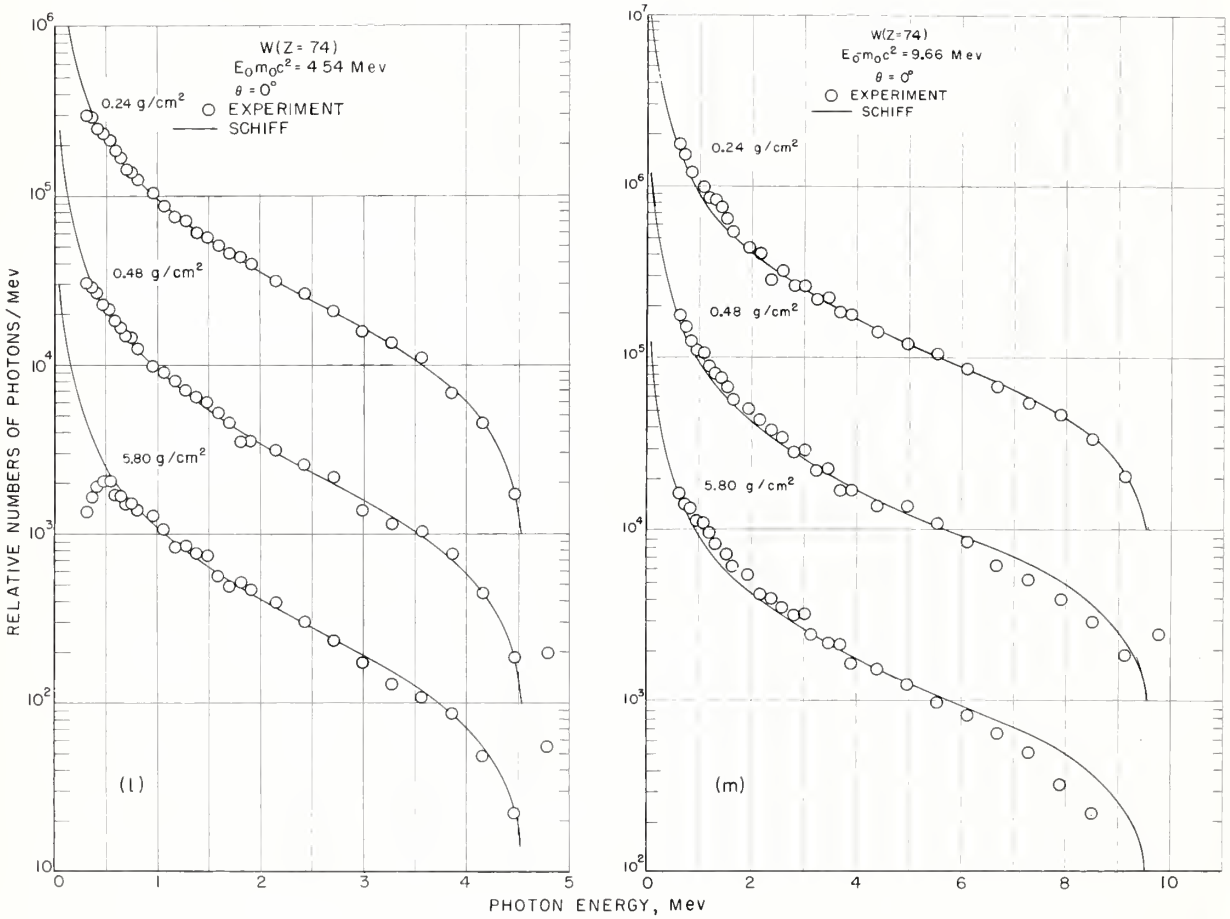

FIGURE 8.5 l and m. *Relative X-ray spectra (in photon Mev interval) produced by 9.66-Mev and 4.54-Mev electrons from a betatron (Starfelt and Koch, 1956).*

Curves are given for three tungsten target thicknesses: 0.24 g/cm², 0.48 g/cm², and 5.80 g/cm². Only the curve shapes, not their relative positions, are significant. The points were obtained with a scintillation spectrometer. The solid curves were calculated from the thin target formula of Schiff (1951), and normalized to the experimental data.

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[30] $(\hat{s}'_m)^{\text{wall}}_{\text{gas}}$ will be slightly less than 1, because of the "zero-energy" density effect correction (Sternheimer, 1956). However, this effect will be of the order of 1 percent which can usually be neglected in neutron dosimetry at its present state of development. The symbol for this stopping power ratio indicates that it is a mean value obtained by averaging over the spectrum at the point of interest; the prime indicates that the charged particles are protons.

[31] Note that the numerical factor is 0.90 instead of 0.877 because W is assumed 35 ev for protons instead of 34 ev as was used for electrons.

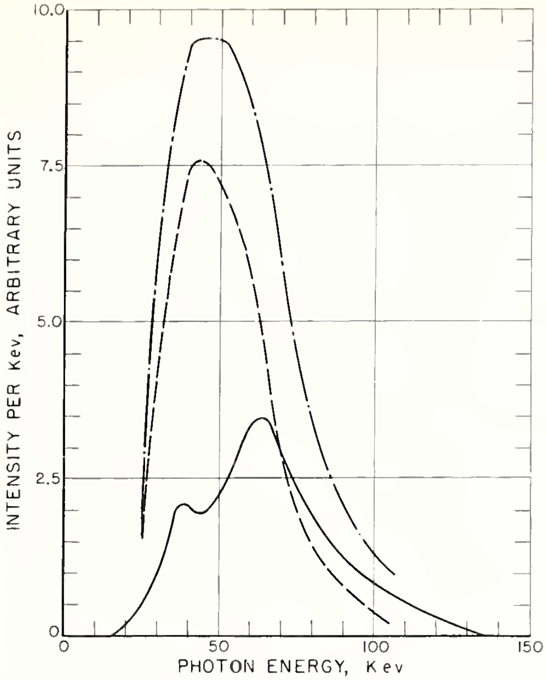


FIGURE 8.5n. *X-ray spectra obtained with 140-kv tube potential at a depth of 5 cm in a water phantom with a field of 300 cm.*[2]

The solid curve gives the primary spectrum, the dashed curve gives the spectrum of the scattered radiation and the dot-dash curve gives the sum of the two. The incident beam had a HVL of 1.45 mm Al.

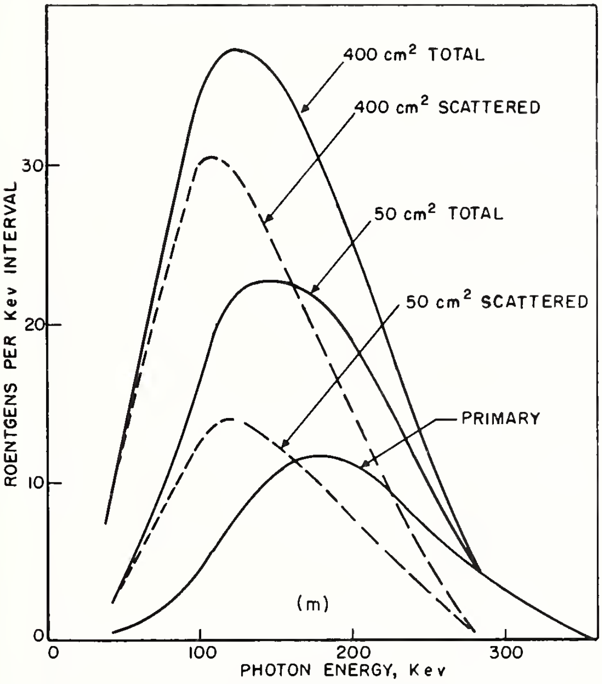


FIGURE 8.5o. *Spectral distribution of the exposure dose at a depth of 10 cm in a water phantom (Cormack et al., 1957), X-rays generated at 400 kvp (HVL=3.8 mm Cu in absence of phantom).*

The dashed curves apply to the scattered radiation only for field areas of 50 and 400 cm². The solid curves show the primary distribution and the total distribution combining both primary and secondary radiation. All curves are normalized to a primary surface exposure dose of 1000 r. (Reproduced by courtesy of the Editors of the British Journal of Radiology.)

TABLE 8.8.—*Product of scattering cross section and average fractional loss of energy, $\sigma \times k$ (barns)*[a]*, for neutrons traversing various elements*

Energy	Element..	H	C	N	O	P	S
	k [b]--------	0.500	0.1431	0.1254	0.1116	0.0611	0.0592
Mev							
0.1		6.3	0.66	(~.6)	0.39	0.21	0.42
.15		5.4	.63	0.51	.39	.15	.24
.2		4.8	.60	.46	.40	.13	.41
.3		3.95	.56	.40	.44	.15	.13
.4		3.45	.52	.36	.67	.13	.18
.5		3.1	.49	.30	.65	.17	.12
.6		2.8	.46	.20	.36	.14	.11
.8		2.4	.41	.23	.33	.15	.14
1.0		2.12	.39	.25	.89	(~.18)	.17
1.5		1.70	.30	.23	.25	(~.18)	.14
2		1.44	.24	.20	.18	.23	.17
3		1.14	.17	.21	.13	.23	.19
4		0.95	.27	.23	.21	.16	.17
5		.83	.19	.18	.13	.15	.15
6		.72	.14	.18	.16	.14	.15
8		.59	.21	.18	.13	.12	.12
10		.47	.16	.16	.13	.12	.11

[a] σ is the scattering cross section. The $\sigma \times k$ values in parenthesis are rough estimates, the cross sections not having been measured. See eq (8.13) for application.

[b] k is the average fractional energy loss of the neutron. These values assume the total cross section is all elastic scattering, isotropic in the center-of-mass system. The errors introduced by this assumption (which is strictly true only for hydrogen) should be small in this energy range.

The absorbed dose in the wall material (according to eq (8.1) is equal to

$$D_{\text{wall}} = \frac{J_m}{100} \times W'_{\text{gas}} \text{ rads}, \qquad (8.14)$$

where W'_{gas} is the energy in ergs for a proton to produce an ion pair in the gas. When W' is expressed in electron volts and d_{gas} is the gas density in g/cm³, then

$$D_{\text{wall}} = 0.90\ Q\ \frac{(W'_{\text{gas}})}{35} \times \frac{0.001293}{d_{\text{gas}}} \qquad (8.15)$$

8.7. *The measurement of absorbed dose in a medium exposed to both neutron and X- or gamma radiation, and the distribution of absorbed dose with respect to LET.*

The biological effectiveness of a given absorbed dose depends on the LET of the charged particles that deliver this dose. In radiation protection the permissible *absorbed* dose depends on LET and similarly in radiobiology a specification of radiation quality in terms of the LET distribution of dose is highly desirable. This is particularly true when significant portions of the dose are delivered over wide variations in LET in the range between 3.5 and 175 kev/μ in water. This occurs upon simultaneous exposure to neutrons and gamma rays or upon exposure to high-energy fast neutrons alone.

An alternative but less complete treatment of the problem consists in a separate assessment of the dose delivered by different types of radiations, particularly neutrons and gamma rays. This approach is only approximate since the RBE applicable in protection varies by about a factor of

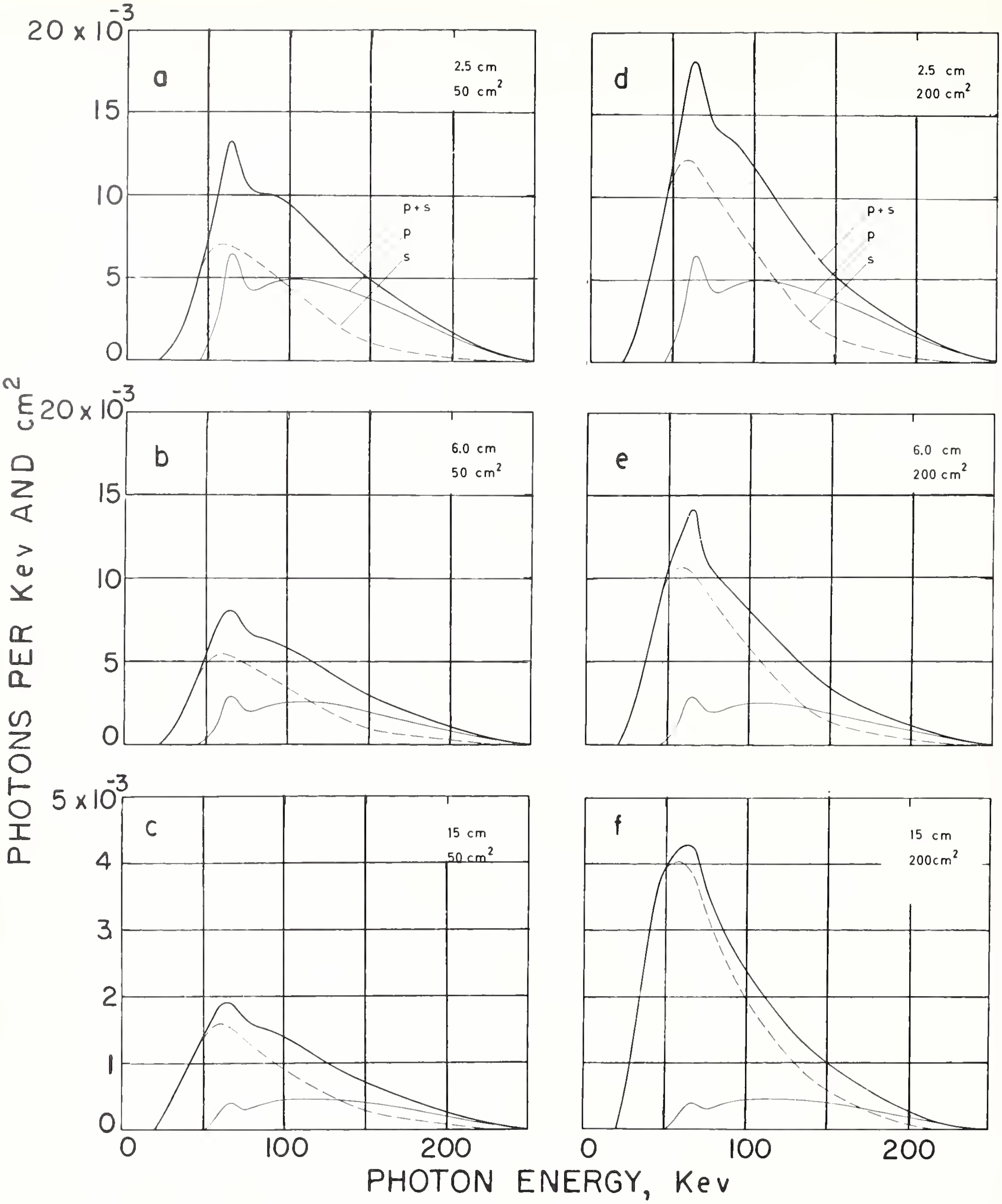


FIGURE 8.5p. *Spectral distribution of primary and secondary 250-kv radiation at different depths (2.5., 6.0, and 15 cm) when the field size at the phantom surface is 50 cm²* (a, b, *and* c) *and 200 cm²* (d, e, *and* f).

The focus surface distance is 67.5 cm; one primary photon per cm² is incident upon the phantom surface. The upper curve in each diagram corresponds to the total radiation; i.e., is the sum of primary and secondary photons. The ordinate scale in diagram c and f have been enlarged 4 times (Hettinger and Liden, Acta Radiol. **53,** 71, 1960).

two for fast neutrons alone and since the RBE determined in radiobiological experiments may show similar differences. Consequently high precision is of limited value in such determinations.

A method for the determination of the LET distribution of dose has been developed for particles that are heavier than electrons and that have ranges in excess of a few microns in water (Rossi and Rosenzweig, 1955). This method employs a specially designed spherical proportional counter and it permits the complete LET distribution to be measured within the irradiated material. In its present stage of development the LET thus determined is an average over one or more microns of track but this "resolution" is capable of further improvement.

The measurement of the total absorbed dose due to a mixed neutron and X- or γ-ray beam presents no special difficulty. The ionization chamber should have the same composition as for the measurement of neutron radiation. The absorbed dose may be calculated with fair accuracy from the ionization produced in this chamber by the application of the cavity relation as already described. Slight ambiguity arises, however, regarding the appropriate values of W and s_m because these are not the same for the two radiations. If the relative contributions to the absorbed dose from neutron and γ radiation are approximately known, weighted mean values of W and s_m may be used. If the wall material accurately matches the medium of interest, $D_{\text{medium}} = D_{\text{wall}}$ and no further calculation is required. If the match is not exact, an additional step as given in eq (8.12) for the neutrons, and a step as in eq (8.8) for the γ-rays will be required to obtain the absorbed dose in the medium. An approximate knowledge of the relative neutron and γ-ray contributions is needed to make this latter transformation.

In radiobiology, as well as in health physics, it is generally necessary to know the magnitude of the contribution of each radiation separately to the total absorbed dose. The following procedures have been used:

(a) The ionization is measured at the place of interest with two thick-walled ionization chambers, one having walls of a hydrogen-rich material and the other having walls of a material that contains no hydrogen, e.g., graphite, aluminum, or polytetrafluoroethylene.

The method of calculating separately the absorbed dose in tissue due to each radiation has been described by Ebert et al. (1955). There are two difficulties with this method. The first is that the gamma-ray chamber will also respond to neutrons. In the case of a carbon-CO_2 chamber, the neutron response (rads) is probably on the order of 10 to 20 percent of the gamma-ray response (rads) depending upon the neutron energy (Hurst et al., 1956; Sayeg et al., 1958). The second difficulty is that the gamma-ray responses of the two chambers are in general different, especially for gamma rays of low energy, thus rendering the subtraction technique difficult unless the gamma-ray spectrum and gamma-ray response are known. The first difficulty makes it hard to measure gamma-ray absorbed dose in the presence of many neutrons and the second difficulty hinders the measurement of neutron absorbed dose in the presence of strong gamma radiation.

Neutron and gamma-ray contributions to absorbed dose may be distinguished in a proportional counter by separation of large pulses due to neutron recoils from the small pulses due to secondary electrons from gamma radiation. In the neutron dosimeter (Hurst, 1954; Wagner and Hurst, 1958), the total ionization due to the recoil nuclei is measured separately in a proportional counter of polyethylene filled with ethylene and the integral of count times pulse height is registered. Small pulses due to secondary electrons from gamma rays are discarded by pulse height selection. At the same time one cannot avoid losing the small neutron pulses. By proper setting of the pulse height selector, the loss of neutron absorbed dose can be usually held to less than 5 percent, and can be estimated. If the pulse height is proportional to the number of ion pairs formed this method of dosimetry is equivalent to the ionization chamber, with the added advantage of being insensitive to gamma radiation. The proportionality between pulse height and number of ion pairs depends on two conditions: (a) There must be no electron attachment, and (b) the height of the pulse at the output of the

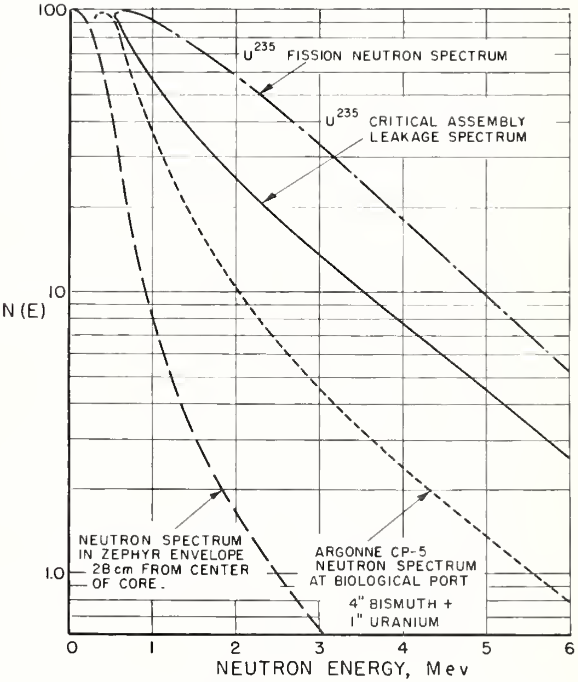


FIGURE 8.6a. *Neutron fission spectra from reactors.*

References: Watt, 1952; Rosen, 1956; USNRDL measurement on CP-5 spectrum (unpublished); Codd, Sheppard, and Tait, 1956.

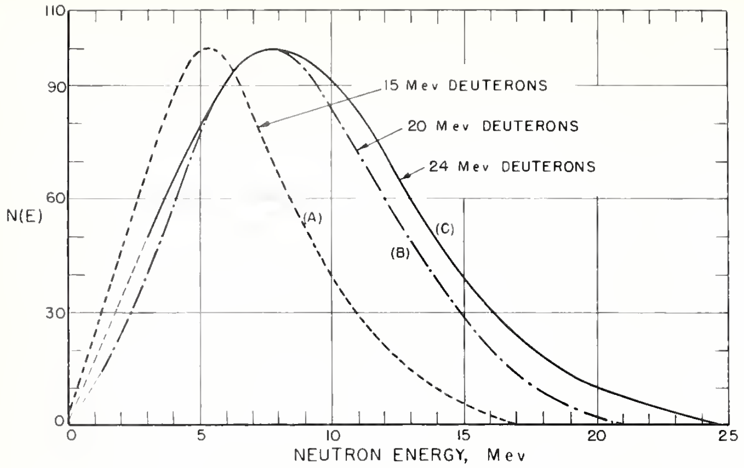
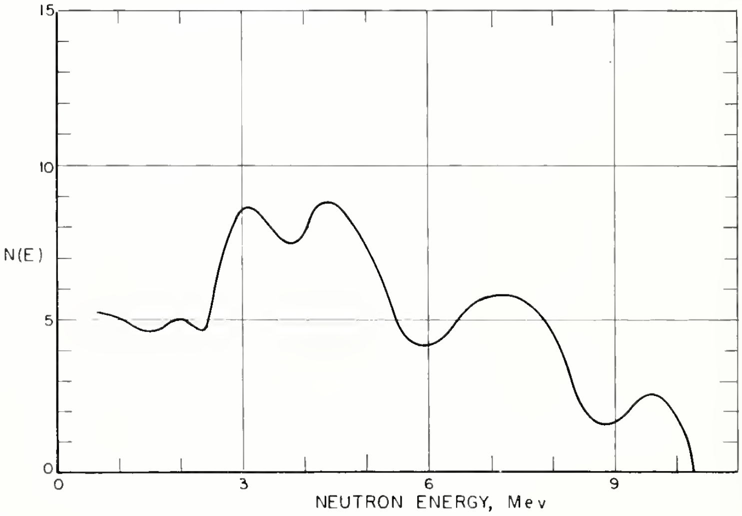


FIGURE 8.6b. *Neutron energy spectra for 15, 20, and 24 Mev deuterons on thick Be target.*

Measurement for 15 Mev was by Cohen and Falk, 1951; for 20 and 24 Mev by Tochilin and Kohler, 1958.

FIGURE 8.6c. *Neutron energy spectrum of Pu–Be* (,αn).

Nuclear emulsions were used for the measurement (Stewart, 1955).



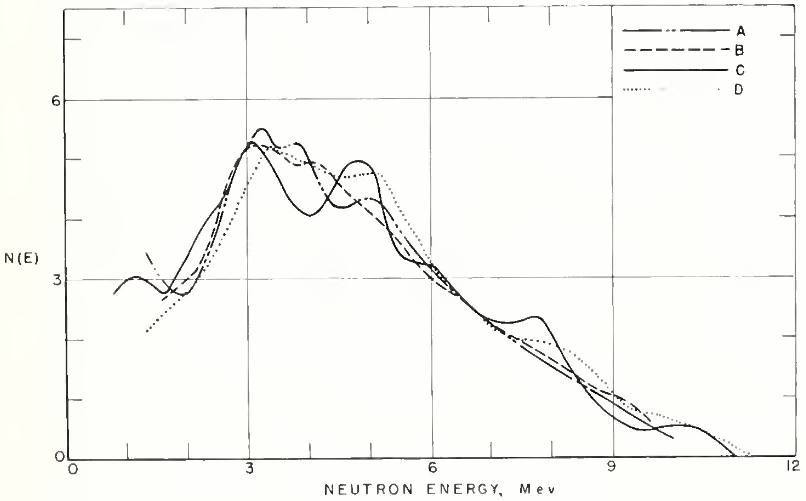


FIGURE 8.6d. *Neutron energy spectra of Po–Be* (α,n).

Curve A was measured with a proton recoil proportional counter telescope (Cochran and Henry, 1955); curve B was determined by a similar method (Breen, Hertz, and Wright, 1956); curve C with nuclear emulsions (private communication from author, 1960, indicates that the all quoted energies should be increased by 3 percent, Whitmore and Baker, 1950); curve D with Li^{6}I scintillation crystal spectrometer Murray, 1958).

linear amplifier must not depend on track orientation. Condition (a) may be fulfilled by excluding from the counter such gases as water vapor, oxygen, and some of the halogens, which have very large electron attachment cross sections (Healey and Reed, 1941). If the rise time and decay time constants of the linear amplifier are greater than the collection time of electrons in the counter, the pulse height at the output of the amplifier depends only slightly on the rise time of the proportional counter pulse and hence condition (b) will essentially be fulfilled (Hurst and Ritchie, 1953).

Inversely, a proportional counter may be used as a dosimeter for gamma rays in the presence of neutrons by pulse height integration of the small secondary electron pulses due to gamma rays and rejection of the large pulses due to neutron recoils (Caswell, 1957). This dosimeter is a graphite-walled He-CO_2 filled proportional counter, analogous to the carbon-CO_2 ionization chamber. Use of a graphite lining and a thin aluminum wall minimizes production of gamma rays in the walls by inelastic scattering of the incident neutrons (which would lead to neutron sensitivity of the counter). Neutron sensitivity in a 2.5 to 3 Mev $H^2(d,n)He^3$ neutron field is ≤ 1 percent of the absorbed dose in tissue.

Another method of measuring ionization which can distinguish between that produced by secondary electrons from gamma rays and recoil nuclei from neutrons is the single ionization detector reported by Auxier, Hurst, and Zedler (1958). In this instrument a cavity of very small linear dimensions and low gas pressure is used. The probability of an electron making an ionizing collision in the gas is low. The probability of more than one collision is correspondingly much lower. Hence, to a good approximation each ionizing event leaves behind one ion pair. If each ionization can be detected, then the number of counts equals the number of ion pairs and the energy absorbed in the cavity is simply W times the number of counts. On the other hand fast neutron recoils may lose a large amount of energy in the counter, producing many ion pairs, but still producing only one count per event. The device is therefore very inefficient for neutron recoils. In this way discrimination against neutron radiation is achieved.

(b) Discrimination between neutrons and gamma rays in mixed radiation fields may also be achieved with plastics loaded with granular scintillating materials (Hornyak, 1952; Brown and Hooper, 1958). Fast neutrons striking hydrogenous material in the detector button produce proton recoils, which in turn produce light when they strike the phosphor grains, which is then detected by a photomultiplier. Proton recoils produce large light pulses corresponding to large energy losses in the phosphor grains, whereas secondary electrons from gamma rays produce small pulses corresponding to the low energy per unit track length. The small gamma ray pulses may be discriminated against electronically. Empirical design of these detectors for measurement of dose in the tissue has been discussed by Muckenthaler (1957).

Other detectors have been designed to have a response versus neutron energy similar to the dose in tissue while achieving high discrimination against gamma rays in mixed radiation fields. These include the hydrogenous proportional counters (Hurst, Ritchie, and Wilson, 1951; Dennis and Loosemore, 1957); spherical scintillation detector (Skjoldebrand, 1955); and paraffin-moderated BF_3 counter (DePangher, 1957).

(c) The neutron contribution to the absorbed dose may be calculated from measurements of the number and length of the recoil proton tracks produced in photographic emulsion under controlled conditions.

(d) The neutron absorbed dose rate in any medium may be calculated from a knowledge of the flux and energy spectrum of the neutron beam at the place of interest. These two quantities may be inferred from the induced activity of a series of "threshold detectors" of Pu^{239} surrounded by B^{10}, Np^{237}, U^{238}, and S^{32} (Hurst et al., 1956).

8.8. *The determination of the volume of an ionization chamber by exposure to gamma radiation.*

Sometimes the volume of an ionization chamber cannot be calculated from its dimensions as these are not accurately known. The volume, V, then can be determined by measuring the total ionization, (QV) esu, produced by exposing the chamber to an exposure dose, R roentgens, of gamma radiation. The wall of the ionization chamber should be sufficiently thick to ensure electronic equilibrium at its inner surface. The dimensions of the chamber should be such that the range of the secondary electrons associated with the gamma radiation is great compared with the depth of the ionization chamber, e.g., 1 cm for Co^{60} radiation.

The V can be calculated by the relation

$$V=\frac{(QV)}{R}\times\frac{({}_m\mu_{en})_{\text{air}}}{({}_m\mu_{en})_{\text{wall}}}\times(\tilde{s}_m)^{\text{wall}}_{\text{gas}}\times\frac{W_{\text{gas}}}{W_{\text{air}}}\times\frac{0.001293}{\text{d}_{\text{gas}}}\times A \quad (8.16)$$

where $({}_m\mu_{en})$ signifies the energy absorption coefficient for the gamma radiation used, whereas $\tilde{s}_m$ and W are the mass stopping power ratio and the energy per ion pair for the associated secondary electrons. A is a correction for the attenuation of the gamma radiation by the wall of the ionization chamber.

The relation can be simplified in case the ionization chamber is filled with air and/or when the wall material and the gas are matched for the gamma radiation.

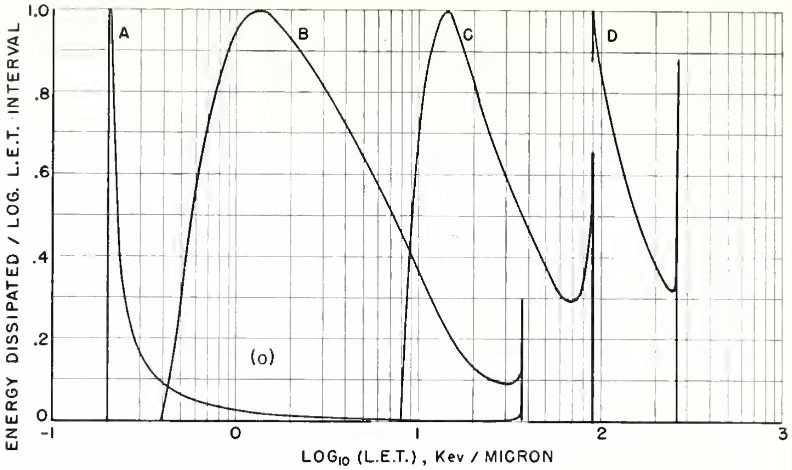


FIGURE 8.7a. *Linear energy transfer (LET) distribution of several ionizing radiations in water (Howard-Flanders, 1956).*

Energy transfers of all magnitudes (including δ-rays) have been attributed to the primary particle when calculating LET. The ordinate is proportional to the fraction of the total energy deposition per unit interval of log^{10} (LET). The proportionality factor is not the same for all radiations. Curve A is for Co^{60} γ-rays (Cormack and Johns, 1952); curve B for 200 kvp X-rays (referred to as radiation "b" by Cormack and Johns (1952); curve C for 5-Mev neutrons (Boag, 1954); and curve D for 5.5.-Mev α-rays (Gray, 1955; Howard-Flanders, 1956).

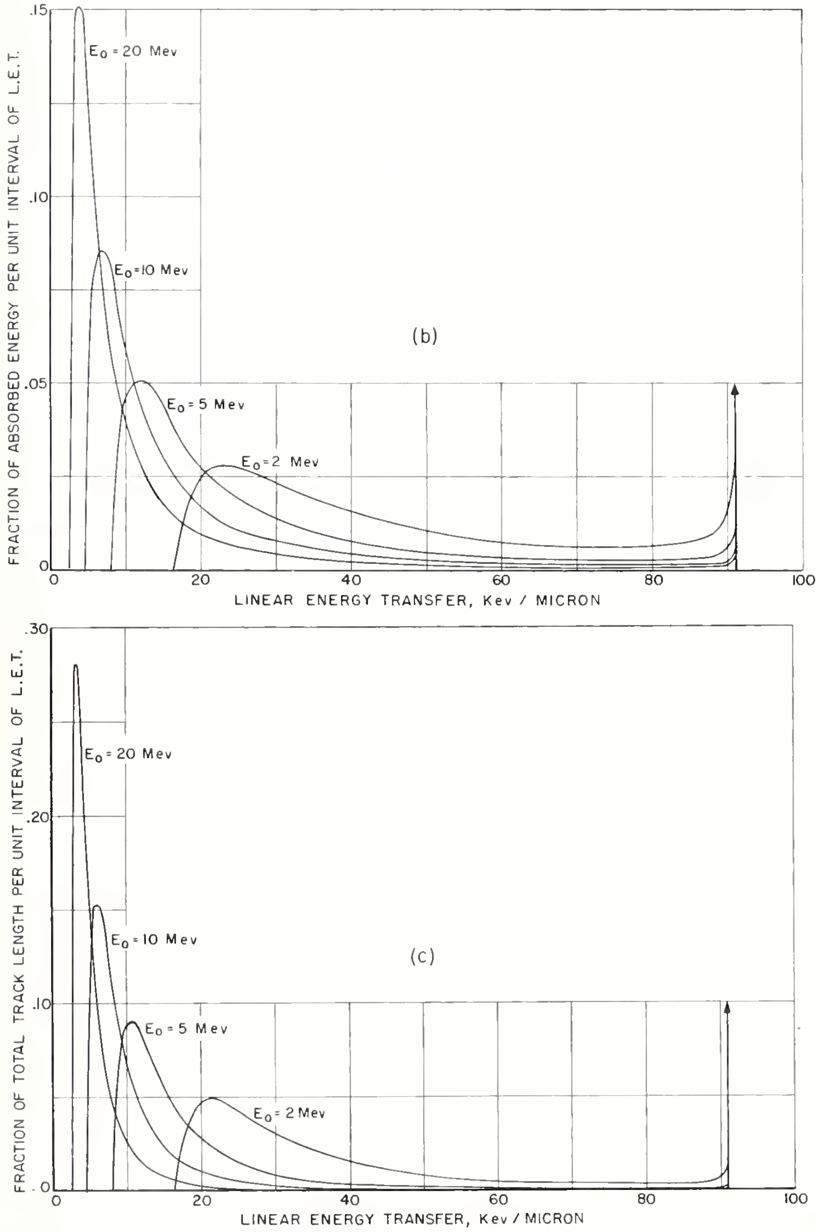


FIGURE 8.7b and c. *LET distribution for recoil protons produced in water by monoenergetic neutrons, when first collisions only are considered; i.e., when the irradiated object is thin compared with the mean free path of the neutrons in it.*

In figure 8.7b the ordinate gives the fraction of the total dose deposited per unit interval of LET and in figure 8.7c, the fraction of the total track length traversed per unit interval of LET. Additional LET distribution curves for mixed energy neutron beams and for total absorption of the neutrons in multiple collisions are given by Boag (1954).

8.9. *Other graphs and tables characterizing X-, gamma-, and neutron radiation and their associated secondary particles.*

(a) Typical primary X-ray spectra. Figures 8.5*a–o* present some measured typical primary X-ray spectra. They are by no means complete in their coverage of usual X-ray kilovoltage-filtration combinations, partially because of the limited number of spectral measurements now available. However, they do show that theoretical calculations can in some cases give a fairly useful estimate of an unknown spectrum. A brief bibliography of some of the literature of X-ray spectra is included in the references.

Very few data are now available on spectra of X-rays that have been degraded in energy by scattering in a phantom. A discussion of this problem together with some results are given by Cormack et al. (1957). As a first approximation, this energy degradation is usually ignored in computing energy absorption in tissue, but considerable underestimation can result from this procedure, particularly in the case of compact bone for X-ray energies where the photoelectric effect is important. Figure 8.5o indicates the change in the total spectrum resulting from scattered radiation at a depth of 10 cm in a water phantom, for 400-kvp X-rays. $\bar{f}_{\mathrm{bone}}$ is 1.39 for the total spectrum with a 400-cm^2 field, compared to 1.11 for the primary spectrum, a difference of 25 percent. $\bar{f}_{\mathrm{muscle}}$ is the same within 0.5 percent for these two spectra. It would be desirable for further experiments of this type to provide tables of conversion factors for various field sizes, depths, and primary X-ray qualities. These could then be included in summaries such as table 8.2 and figure 8.1 which were calculated in most cases for the primary spectra. From such data more accurate determinations of the absorbed dose in bone could be made.

(b) Typical distribution functions for initial Compton recoil electron energy. The curves shown in figures 8.3a–f represent the recoil-electron spectral-distributions produced by a number of monochromatic γ-ray energies. They can be combined with pair and photoelectron spectra (at energies where these are important) to give the total initial electron spectral distributions.

(c) Typical "slowing-down" electron distribution functions. The spectrum of secondary electrons produced in a medium, (i.e., those generated by the X- or gamma rays), under equilibrium conditions, by the continuous slowing down of initially monoenergetic electrons, is given approximately by the reciprocal of the stopping power of the medium. The production of tertiary electrons (δ-rays) by secondary electron collisions with atomic electrons in the medium, has been calculated approximately for a few initial energies. The resulting electron spectral distributions with and without δ-rays are given in table 8.7 for tissue, assuming the production of δ-rays to be the same as in graphite (Spencer and Fano, 1954; Spencer and Attix, 1955). More information is available in the report of McGinnies (1959).

A word of explanation concerning table 8.7 is in order. When a collision occurs between a secondary electron and one of the electrons of the medium, the most energetic electron after the collision is considered the one which was incident. Thus, all tertiary electrons will have an energy of $T_0/2$ or less. It will be seen that column (2) is identical to (3), (4), and (5) for energies greater than $T_0/2$ but for energies less than $T_0/2$ the tertiary electron flux is added to the secondary and the numbers in columns (3), (4), and (5) are greater than those in (2) where δ-rays are neglected.

The area under the curves, obtained by plotting the data of columns (3), (4), or (5) against the electron kinetic energy in column (1), gives the range of the electron of initial energy T_0 in g/cm^2. This range is approximately given in figure 8.4. For example, the area under the curve for $T_0 = 1.308$ Mev is 0.640 g/cm^2. The individual ordinate at energy E divided by this area gives the fraction of the range spent while the electron has energy E. For example, the fraction of the range spent while the electron has energies in the range 0.1 Mev to (0.1+0.01) Mev is

$$\frac{0.346\times 0.01}{0.640}=0.0054.$$

The table may be used to calculate the number of electrons per cm^2 per Mev at energy E produced at a point under electronic equilibrium conditions when one electron per cm^3 is set in motion with energy E_0. Thus, for example, if one electron with energy 1.308 Mev is set in motion per cm^3 of tissue, one would observe an electron flux of 0.346 electrons per cm^2 per Mev at energy 0.1 Mev, or 0.00346 electrons per cm^2 in the range 0.1 to (0.1+0.01) Mev. By interpolating the data of table 8.7, it is possible to obtain the electron spectra for other starting energies and so to determine the electron flux in tissue when a spectrum of electrons, as illustrated in figure 8.3, is set in motion.

(d) Typical neutron energy spectra are given in figures 8.6a–d for important radioactive neutron sources, accelerator neutron sources and locations in nuclear reactors.

8.10. *Distribution of the absorbed energy* (*Concept of LET*)

The absorbed dose specifies the total energy imparted by ionizing particles to unit mass of the irradiated material. The detailed spatial distribution of this dose depends primarily on the velocity and charge of the particles traversing the medium. Rather similar distribution patterns may be produced by electrons and protons of appropriate energies. Hence the "quality" of the radiation is best specified in terms of the type of tracks produced rather than in terms of the nature of the primary radiation (X-rays, neutrons, etc.)

or of the secondary ionizing particles (electrons, protons, etc.).

A full description of the type of track would require a knowledge of the number and spatial distribution of all the collision processes which occur and this is clearly impossible. A single parameter which has been found useful for characterizing the track is the LET, and this parameter has been found in general to correlate well with the biological effectiveness of the radiation.

Even for a homogeneous primary radiation there will usually be a wide variation of LET along the tracks of individual secondary particles and between tracks of different initial energy. The situation can, therefore, best be represented by giving the distribution of LET throughout the irradiated object. This can be done in two ways—in relation to dose, or to track length. For all chemical and many biological effects the chief interest lies in the proportion of the total dose which is deposited in different LET intervals. This is the "distribution-in-LET" of the absorbed dose, and typical spectra of this kind are presented in figures 8.7a and 8.7b. For certain biological effects depending upon the transit of secondary particles across small biological structures, the interest lies in the proportion of the total track length which falls into different LET intervals, i.e., the "distribution-in-LET" of the track length and a typical spectrum of this kind is given in figure 8.7c. Experimentally determined distributions are given for comparison in figure 8.8. For a fuller discussion of LET distribution see the original papers (Burch, 1957; Cormack, 1956; Cormack et al., 1957; and Cormack and Schneider, 1959).

When a mean value of LET is quoted, instead of a distribution, it is necessary to specify exactly how this mean has been taken, i.e., whether it refers to the LET of a particle of the mean energy of the secondaries, or to a mean value of LET with respect to dose, or with respect to path length. The fuller information conveyed by a distribution curve, however approximate, is much more valuable than a single mean value. It is important to note that figures 8.7 a, b, and c represent distribution with respect to *dE*/*dx* measured along the track of the secondary electrons which result from photon scattering and absorption, or of the recoil protons; and that the energy loss *dE* is obtained by including all energy transfers between the moving particle and the molecules encountered within a path length *dx*. Some of these energy transfers will be large enough to give rise to independent delta-ray tracks, and the energy communicated in such a transfer is not dissipated locally along *dx*, but along the delta ray. If delta-rays above an arbitrary initial energy are considered as independent tracks, a different LET distribution will be obtained which is more heavily weighted towards high LET, owing to the inclusion of these low energy tracks.

Calculations of the LET distributions for certain radiations have been made in which the primary ionizing particles and delta rays are treated separately (Burch, 1957). Some of these distributions are shown in figure 8.9 in which

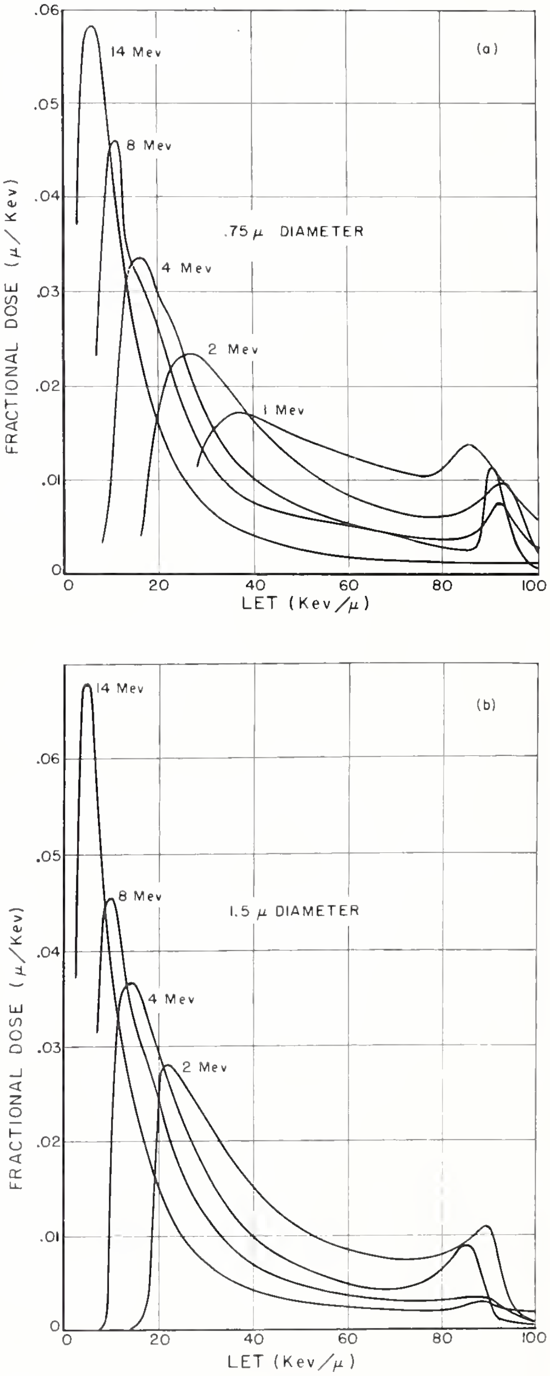


Figure 8.8a and b. *Experimentally determined LET distributions.*

These were determined for nearly monoenergetic neutrons of the indicated energies and for test spheres corresponding to diameters of 0.75 and 1.5μ in soft tissue (Rosenzweig and Rossi, 1959).

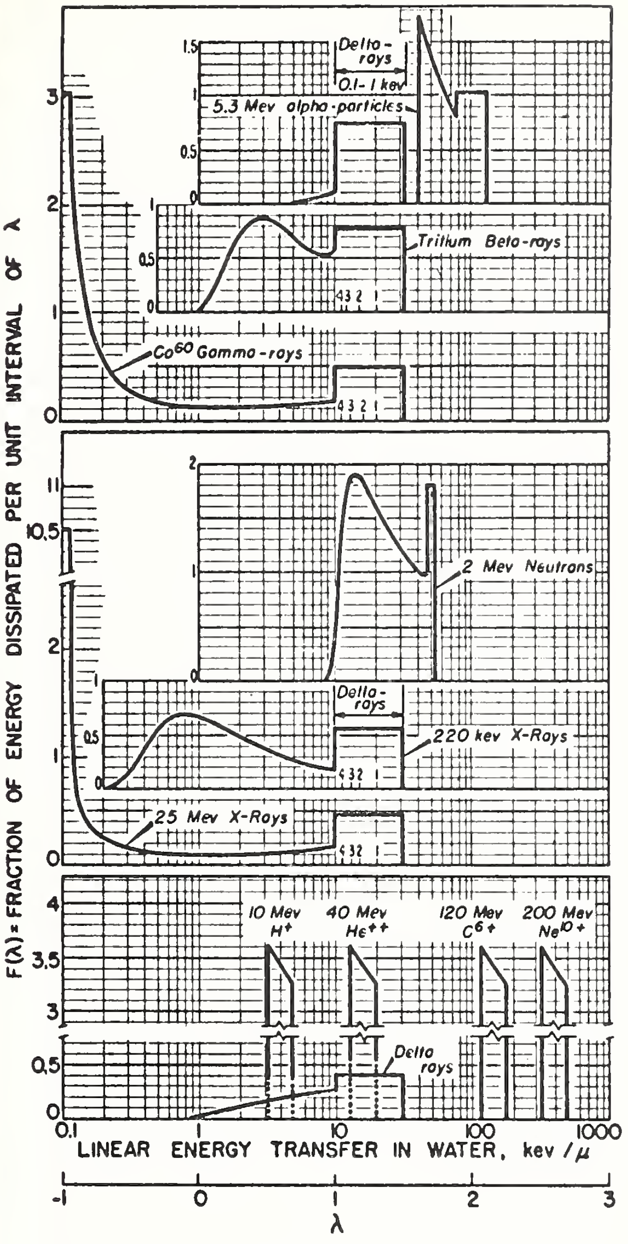


Figure 8.9. *The LET spectra in water for various commonly used ionizing radiations.*

$F(\lambda)$, the fraction of energy dissipated per unit interval of λ is plotted as a function of λ on a linear scale, where $\lambda = \log_{10}$ LET. For convenience, the horizontal scale is also marked directly in LET in kev/μ in water on a logarithmic scale. The fraction of energy dissipated between any two LET values is equal to the area under the curve between these limits. The rectangular areas represent the fraction of energy dissipated by δ-rays of energy 1000 ev and less. The LET values are for the core of the tracks with δ-rays of 100 ev and over considered separatedly. These spectra apply as long as the track segments t are shorter than 100 A. If values of t in excess of 100 A are used, the right-hand side of the δ-ray spectrum should be cut short at the points marked 1, 2, 3, 4. (The fraction of X-ray or β-ray energy dissipated by δ-ray tracks of length (along the truck) less than 100, 200, 500, or 1000 A, is represented by areas within the δ-ray rectangles to the right of the points marked 1, 2, 3, and 4, respectively. These areas may he deleted from the LET spectrum when treating data on materials for which it is thought that the short range δ-rays are without effect.) (Howard-Flanders, 1958.)

the fraction of energy dissipated per unit logarithmic LET interval is plotted as a function of the logarithm of the LET. The fraction of energy dissipated by delta rays with energies up to 1 kev is represented by the rectangular area between 10 and 30 kev/μ in water (Howard-Flanders, 1958). There is some uncertainty in the rate of energy loss in the slower electrons and therefore in the LET values to be ascribed to the delta rays. These distributions give the mean LET values in short lengths of track and require modification if averages are required in lengths of track greater than 50 angstroms in water.

8.11. *Saturation in ionization measurements.*

Failure to collect all the ions produced in an ionization chamber may be due to either initial or general recombination. The amount of initial recombination depends upon the LET of the ionizing particle and upon the strength of the collecting field and its direction in relation to the particle track. The best treatment of initial recombination is that by Jaffe (1929) with modifications by Zanstra (1935), and that by Kara-Michailova and Lea (1940). Initial recombination does not depend upon dose rate and is usually only troublesome for slow particles or high gas pressures.

General recombination can be treated theoretically in simple cases but it must be stressed that difficulty in achieving saturation is often due principally to inappropriate design of the ionization chamber. If there are regions in the chamber where the field strength is much lower than the maximum, it will be difficult to achieve saturation in these regions without exceeding, in other parts of the chamber, the field strength at which multiplication of ions by collision can begin.

For plane parallel ionization chambers uniformly irradiated at constant dose rate, a saturation curve can be drawn (Boag, 1956) as a function of the dimensionless variable $\xi = m(d^2\sqrt{q}/V,)$ where d is the spacing (cm), V the collecting voltage (volts), q the ionization intensity (esu/cm^3-sec), and m is a constant depending upon the type and density of the gas. This curve is illustrated in figure 8.10a, where $F(\xi)$ = collection efficiency, and it can be represented with adequate accuracy by the formula $F(\xi) = 2/(1+\sqrt{1+\xi^2})$. For air at 760 mm and 20° C, the constant m has the value 15.9.

For cylindrical or spherical geometry in the ionization chamber the foregoing saturation curve still applies, but instead of d, one must insert $K(a-b)$, where $(a-b)$ is the radial spacing of the electrodes, and K is as hape factor whose values are given in figure 8.10b.

The foregoing curve and formula for $F(\xi)$ apply to continuous radiation. In the case of instantaneous pulses of radiation (Boag, 1956) whose duration is short compared with the time required to collect the ions, the appropriate dimensionless variable is $u = \mu(d^2r/V)$, where r is the charge density per pulse (esu/cm^3). In this case the collection efficiency, $F(u), = (1/u) \log (1+u)$. For air at 760 mm and 20° C, $\mu = 1000$. $F(u)$ is given graphically in figure 8.10c. for plane geometry. The extension to cylindrical and spherical geometry again involves the shape factors K_c and K_s from figure 8.10b.

When an X-ray tube is run on a pulsating potential supply, the X-ray output occurs in pulses whose duration may be long compared with the collection time of the ions in the ionization chamber. In this case, the collection efficiency depends upon the average dose rate during the pulse, and, as in the case of instantaneous pulses, it is independent of the pulse repetition frequency. Saturation curves taken with a pulsating X-ray output agree well with those taken with continuous output when the average intensity *during the pulse* is the parameter used.

Chambers that include regions of plane, cylindrical, and spherical geometry can be dealt with by considering the collection efficiency in each of the regions separately and adding together the currents from all regions. These formulas and curves should be regarded as a guide to the variables which determine saturation but not as a substitute for accurate experimental checks on saturation conditions in a particular chamr. be

TABLE 8.9.—W$_{air}$ *for electrons*

Author	Radiation	W_{air}	Standard deviation	Stopping[a] power used
		ev	*ev*	
Emery (1956)	Survey	33.5	(1)	
Gross et al. (1957)	$S^{35}\ \beta$	33.6	0.3	Yes [b]
Bay et al. (1957)	$S^{35}\ \beta$	33.7	0.3	No
Barber (1955)[c]	1 to 35 Mev electrons	33.8	1.2	Yes
Jesse (1958)	$S^{35}\ \beta$	33.9	0.5	No
Jesse and Sadauskis (1955)	$H^3\ \beta$	33.9	(1)	No
Do	$Ni^{63}\ \beta$	34.0	(1)	No
Weiss and Bernstein (1956)	2 Mev X-rays	33.9	0.8	Yes
Bernier et al. (1956)[d]	Co^{60} γ-rays	33.0	0.3	Yes
Skarsgard et al. (1957)[d]	22 Mev X-rays	32.8	0.6	Yes
Ovadia et al. (1955)	9 to 17 Mev electrons	34.3	1	Yes
Goodwin (1959)	Cs^{137}	33.9	0.5	Yes

[1] Not given.

[a] As there are still some uncertainties in the stopping power ratios, this may be an important index of the reliability of the W measurements.

[b] The stopping power ratio was determined by a separate experiment.

[c] Barber did not determine W_{air}. This value is obtained from Barber's values for other gases and the ratio of W's obtained by other authors.

[d] There is some doubt about the reliability of these determinations because the current measurements were not obtained under saturation conditions.

9. Chemical Dosimetry

In recent years, chemical methods of dosimetry have assumed a new importance. The fields in which they are currently being applied may be grouped under four headings:

9.1. Laboratory standards of absorbed dose in the range 10^3 *to* 10^6 *rads.*[32]

Certain oxidation and reduction reactions, involving solutes in dilute aqueous solution, have been found to take place to an extent directly related to the absorbed dose, and measurable with current analytical techniques to within 1 percent. Absolute calibrations of the response of these systems to absorbed dose have been made by direct methods which avoid the assumptions inherent in ionization procedures.

a. *Ferrous sulfate solutions.* The chemical reaction which has been most extensively used as a secondary standard of dose is the oxidation of ferrous sulfate in 0.1 to 0.8N sulfuric acid solution. This was first studied by H. Fricke (1929), after whom the system has been called the "Fricke dosimeter." In recent years this has gained much favor, particularly in studies of the radiation chemistry of liquids (Miller, 1956). The reproducibility and ease of measurement are good. Disadvantages lie in the variation of the yield with the LET of the incident radiation, and in the requirement that the concentration of dissolved oxygen in the solution be kept above a certain limit throughout the exposure. The mechanism of the reaction is now well understood, even in detail (Allen and Rothschild, 1957; Allen, Hogan, and Rothschild, 1957). In the light of present knowledge of its behavior the following standard procedure can be recommended.

A solution is prepared, which is about 1 mM in ferrous sulfate or ferrous ammonium sulfate, 1 mM in sodium chloride (to neutralize the effect of possible organic impurities in the water (Dewhurst, 1952; Weiss, Allen, and Schwarz, 1956)), and 0.8N in sulfuric acid. The water is distilled beforehand from alkaline permanganate solution in all-glass system; further purification is unnecessary. Chemicals of analytical grade may be used without further purification. Using a solution initially in equilibrium with the atmosphere, a dose of between 5,000 and 50,000 rads of X-, γ-, or fast electron radiation may be measured but to avoid depletion of the oxygen content of the solution the upper limit should not be exceeded. Higher doses of X-, γ-, or fast electron radiation up to 200,000 rads may be measured if the solution is saturated beforehand with oxygen, care is taken not to introduce additional impurities during equilibration with this gas, and higher initial ferrous ion concentration (4 mM) is used. With radiations of higher LET, the consumption of oxygen is less rapid and somewhat higher doses may be given without oxygen depletion if the solution is well stirred during irradiation to maintain the oxygen concentration in the irradiated zone.

The temperature coefficient of the γ-ray-induced oxidation is 0.04±0.03 percent per °C between 0 and 70 °C (Schwarz, 1954), and the yield is independent of dose rate between 6 and 10^8 rads/min (Schuler and Allen, 1956). With radiations of high LET, a slight dependence of yield on ferrous ion concentration is observed, and is at present not fully explained. For example, with the recoiling radiations from the $B^{10}(n,\alpha)Li^6$ process, the ferric

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[32] Throughout section 7 the ranges given refer to the absorbed dose in the dosimetic material itself, essentially water.

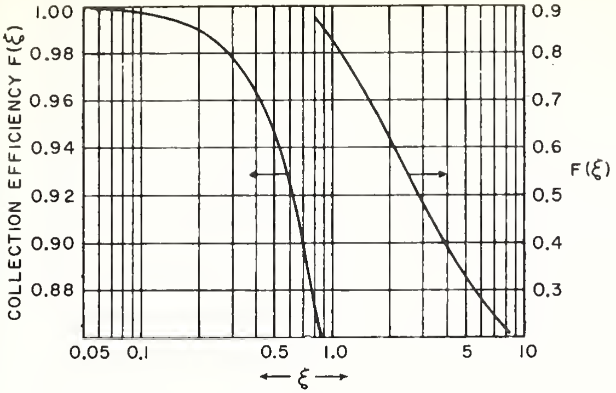

Figure 8.10a. *Collection efficiency in an ionization chamber exposed to continuous radiation.*

The curve is broken into two parts for greater accuracy in reading. The left-hand ordinate scale applies to the left-hand portion of the curve, and the right-hand ordinate scale to the remainder of the curve (Boag, 1956).

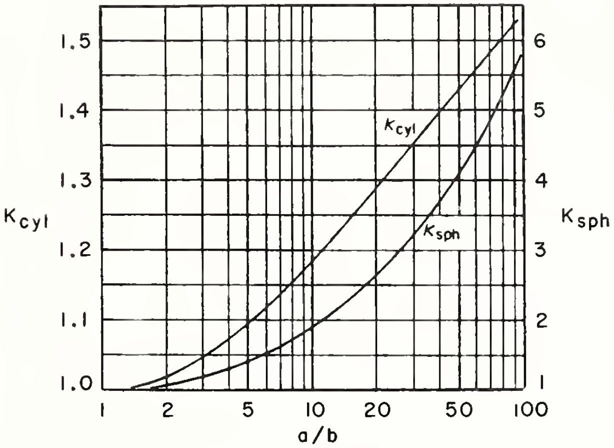

Figure 8.10b. *Factors for calculating the equivalent gap length in cylindrical and spherical ionization chambers.*

The a and b are external and internal radii of the chamber, respectively. Equivalent gap length=$k_{cyl}(a-b)$ for cylindrical geometry. Equivalent gap length=$k_{sph}(a-b)$ for spherical geometry (Boag, 1956).

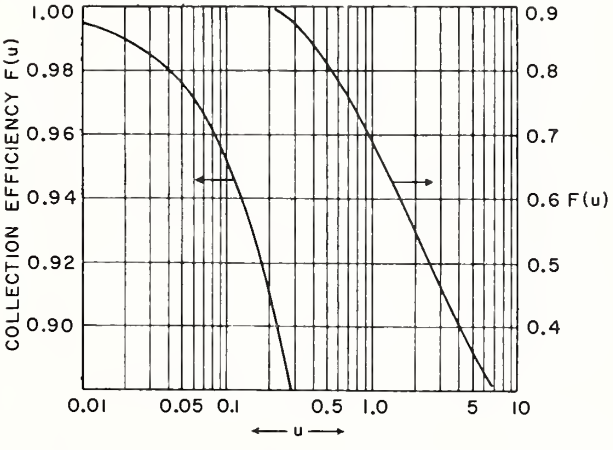

Figure 8.10c. *Collection efficiency of an ionization chamber exposed to pulsed radiation.*

The curve has been broken into two parts for accuracy in reading. Examples: For $u=0.2$, $F(u)=0.91$; for $u=2$, $F(u)=0.55$ (Boag, 1956).

yield in 10 mM ferrous solutions is about 4 percent higher than in 1 mM (Schuler and Barr, 1956); with deuterons, about 1.5 percent (Schuler and Allen, 1957); but with γ-rays, the difference is negligible.

In the absence of radiation, aerated ferrous sulfate solutions are slowly oxidized in a thermal reaction with dissolved oxygen. The rate of this oxidation is proportional to the square of the ferrous ion concentration and the first power of the oxygen concentration (Huffman and Davidson, 1956). In 10 mM ferrous solutions this spontaneous oxidation proceeds to the extent of about 2 μM/liter day at 25 °C, and sets a lower limit of dose rate at which the system can be used with accuracy. A correction for this oxidation may be obtained from an unirradiated sample of the same solution as a control.

The analysis of the ferric ion is most readily undertaken by direct spectrophotometry on the dosimeter solution at 304–5 mμ. The extinction coefficient varies somewhat with temperature (about 0.7 percent/°C (Dewhurst, 1953)), so that for high precision a constant temperature cell holder is advisable. Recent determinations of the molar extinction coefficient are in agreement within their stated limits of accuracy, bearing in mind the temperature coefficient just mentioned. The values are: 2174 ± 6 at 23.7 °C (Schuler and Allen, 1956), 2160 ± 10 at 22.5 °C (Lazo, Dewhurst, and Burton, 1954), and 2094 ± 20 at 20 °C (Haybittle, Saunders, and Swallow, 1956), all in 0.8*N* acid. If a spectrophotometer is not available, an alternative colorimetric method of determining ferrous ion as the complex with *o*-phenanthroline is simple and reliable (Miller, 1956). Very small quantities of ferric ion, corresponding to doses of up to 100 rads of γ-radiation, can be measured by using Fe^{59} tracer and extracting the ferric ion into isopropyl ether as its thiocyanate complex (Rudstam and Svedberg, 1953).

The ferrous sulfate system is less useful with electromagnetic radiation of such quality that photoelectric absorption or pair formation processes take place to an appreciable extent in aqueous solutions. With Co^{60} γ-rays, the ratio of the doses absorbed in 0.8*N* sulfuric acid and in water exposed to the same quantity of radiation can usually be taken as equal to the ratio of the electron densities of the two media, or 1.024; but with radiations of different quality this is, of course, no longer the case. The sulfuric acid concentration can be lowered to a minimum of 0.1*N* with only a 2 percent fall in the ferric yield observed with Co^{60} γ-rays (Allen, Hogan, and Rothschild, 1957), but at lower concentrations of acid the γ-ray yield falls, and the mechanism of the reaction ceases to be simple. It can be taken, therefore, that a concentration of 0.1*N* sulfuric acid is the lowest that can be used for reproducible dosimetry. From the analytical point of view the use of other acidities in this range introduces no complication in that the extinction coefficient of the ferric ion shows little variation with acidity in the region 0.1 to 0.8N (Haybittle, Saunders, and Swallow, 1956).

Even in 0.1N acid, however, the photoelectric absorption may in some conditions be difficult to correct for with precision; while the ferric yield at constant acidity falls with increasing LET (Back and Miller, 1957). Accurate dosimetry of radiations of 200 kev and lower is thus difficult to attain. Similar complications exist with very hard radiation, but few studies of the system with such radiation have been made (Laughlin, Zsula, and Liuzzi, 1957).

Curve A of figure 9.1 is a plot of some recent measurements of the number of ferric ions formed in these solutions per 100 ev released ($G(Fe^{3+})$) as a function of the initial LET of the particles concerned. The yields are integrated over the whole particle tracks, i.e., they are values of the ratio $\Delta M/E_i$, where ΔM is the total number of ferric ions formed by each ionizing particle, and E_i the mean initial energy of the particles in units of 100 ev. It is realized that no physically consistent relationship is to be anticipated between this quantity and the differential energy-loss parameter of the abscissa, but such a plot is considered to be more practical in the present context. More details of the results plotted in figure 9.1 are given in table 9.1. Where more recent work may be considered to have superseded earlier measurements, the latter are omitted from figure 9.1 and table 9.1. Others are omitted when it is difficult to quote a single representative value for the appropriate LET, e.g., with 200 kvp X-radiation. It will be noted that a consistent picture of the behavior of the system has emerged, though some anomalies remain, such as between the sets of results 6 and 7 (fig. 9.1), both obtained with deuterons from a cyclotron.

On many occasions a system is needed which can record higher doses than the $5 \cdot 10^4$ rads measurable with aerated ferrous sulfate solutions. Of a number of systems superior to ferrous sulfate in this respect, two will be described here. Further research is needed on both of these before they can be regarded as rigorous secondary standards of dose.

b. Ceric sulfate solutions. Solutions of ceric sulfate in 0.1 to 0.8N sulfuric acid are reduced to cerous ion with evolution of oxygen by all types of ionizing radiation. Analysis can be carried out spectrophotometrically using the ceric absorption at 320 mμ, at which wavelength the molar extinction coefficient is temperature-independent but varies somewhat with pH. Recent values are 5000 ±50 in 0.1N sulfuric acid (Johnson and Weiss, 1957), and 5580 (Hochanadel and Ghormley, 1953; Medalia and Byrne, 1951), 5600 ±50 (Johnson and Weiss, 1957) in 0.8N acid.

Higher standards of purity than those applying to the ferrous sulfate dosimeter are needed for this system to behave reproducibly. The water must be triply distilled, with the first two distillations from a reagent solution such as alkaline permanganate or acid ceric sulfate. All glassware must be very thoroughly cleaned, and before use allowed to stand for some days filled with a ceric sulfate solution. The dosimeter solution itself must stand for about a week; in making dilutions from a concentrated stock solution, it is desirable to use a very dilute (~50 μM) ceric sulfate solution rather than water which has not previously been standing with ceric ion. There is some evidence of a surface effect in the γ-ray induced reaction in that the cerous yield is considerably enhanced in vessels packed with glass beads. It is not certain that this is genuine as it may merely reflect the great difficulty in cleaning surfaces to the necessary extent. Increasing concentrations of cerous ion appear to reduce the cerous yield by scavenging OH radicals from the particle tracks (Sworski, 1956), so that the cerous yield should fall slightly with increasing dose (about 5 percent at $5 \cdot 10^5$ rads). No experiments of sufficient precision to test this prediction have yet been made.

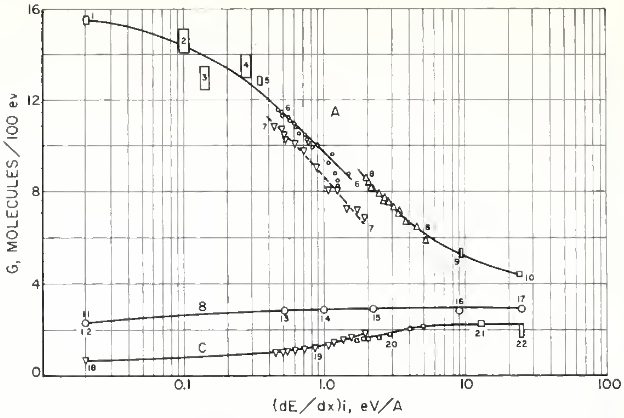

FIGURE 9.1. *Oxidation and reduction yields in dosimeter solutions as a function of the initial LET of the irradiating particles.*

Curve A: Ferric yields in aerated ferrous sulfate solutions in 0.8N sulfuric acid; B: Cerrous yields in aerated ceric sulfate solutions in 0.8N sulfuric acid; C: Ferric yields in aerated solutions 1 mM in ferrous sulfate and 10mM in cupric sulfate, in 0.01N sulfuric acid.

Details as to the points indicated by numbers are recorded in table 9.1.

Latest figures on the cerous yield as a function of initial LET of the radiation used are also plotted in figure 9.1 and recorded in table 9.1.

c. *Ferrous-cupric solutions.* In aerated solutions 0.01N in sulfuric acid and 1 mM in ferrous ion, containing 10 mM cupric sulfate, ferrous ions are oxidized without consumption of dissolved oxygen (Hart and Walsh, 1954). The yield is low ($G(Fe^{3+})=0.66$), and the system appears to be promising for the measurement of doses up to 10^6 rads. Analysis of the ferric ion may be made as usual by spectrophotometry at 304–5 mμ. The molar extinction coefficient of the ferric ion, which in the absence of cupric ion is some 10 percent lower in 0.01N sulfuric acid than in 0.8N acid, is raised by the addition of 10 mM cupric ion to the dilute acid solution until it is within 1 percent of that observed on 0.8N acid in the absence of cupric ion. However, the temperature

TABLE 9.1.—*Aqueous solutions for dosimetry*

Ref. on fig. 9.1	Type of radiation	Mode of energy measurement	G	Init. Fe^{2+} concentration	Footnote reference below
		Curve A. Ferrous sulfate solutions			
			Fe^{2+}	*mM*	
1	2 Mev electrons	Charge input	15.45±0.30	1–10	1
([a])	P^{32} β-particles	Abs. counting	15.4±0.8	1	2
([a])	Co^{60} γ-rays	Colorimetry	15.6±0.3		3
([a])	Co^{60} γ-rays	do	15.8±0.3		4
2	100 kvp X-rays	Ionization	14.7±0.5		5
3	60 kvp X-rays	do	13.1±0.5		5
4	8 kv X-rays	do	13.4±0.6		6
5	H^3-particles	Counting and gas density	12.9±0.2		7
6–6 (circles)	d's, 4–19 Mev	Charge input		10	8
7–7 (inverted triangles)	d's, 3.5–21 Mev	do		1	9
([a])	p's, 0.3–2 Mev	do		1	9
8–8 (triangles)	He ions 12–32 Mev	do		10	8
9	Po^{210} α-particles	Counting	5.3±0.2	10	10
10	$B^{10}(n,a)Li^7$ recoil radiation	Neutron flux measurement	4.38±0.08	10	11
		Curve B. Ceric sulfate solutions			
			Ce^{3+}		
11	Co^{60} γ-rays	Comparison with $FeSO_4$	2.45±0.08		12
12	2 Mev electrons	do	2.31		13
13	d's 18 Mev	do	2.84		13
14	d's 8 Mev	do	2.83		13
15	32 Mev He ions	do	2.92		13
16	Po^{210} α-particles	do	2.78		14
17	$B^{10}(n,a)Li^7$ recoil radiations	do	2.95		13
		Curve C. Ferrous-cupric solutions			
			Fe^{3+}		
18	Co^{60} γ-rays	Comparison with $FeSO_4$	0.66±0.02		15
19[b] (inverted triangles)	d's, 3–22 Mev	Charge input			9
20 (squares)	p's, 0.3-2 Mev	do			9
21[c]	3.4 Mev α-particles	Comparison with $FeSO_4$	2.3		16
22	$B^{10}(n,a)Li^7$ recoil radiations	do	2.0±0.2		15

[a] These results are not plotted in fig. 9.1.

[b] These results are plotted on the same basis of dosimetry as those for ferrous sulfate solutions in the same laboratory (7–7, fig. 9.1).

[c] This represents the results of a comparison with the ferrous sulfate dosimeter using Po^{210} α-particles which had passed through two mica absorbers. The appropriate ferrous sulfate yield has merely been interpolated from fig. 9.1.

1. Schuler and Allen, 1956.
2. Donaldson and Miller, 1955.
3. Hochanadel and Ghormley, 1953.
4. Lazo, Dewhurst, and Burton, 1954.
5. Haybittle, Saunders, and Swallow, 1956.
6. Cottin and Lefort, 1956.
7. McDonell and Hart, 1954.
8. Schuler and Allen, 1957.
9. Hart, Ramler, and Rocklin, 1956.
10. Trumbore.
11. Schuler and Barr, 1956.
12. Johnson and Weiss, 1957.
13. Barr, 1958.
14. Anta and Haissinsky, 1954.
15. Hart and Walsh, 1954.
16. Miller, unpublished work.

variation of the coefficient of extinction is larger in the 10 mM Cu^{2+}—0.01N H_2SO_4 solutions—1.1 percent as against 0.7 percent/°C in the 0.8N acid (Miller, unpublished).

Yields obtained with this dosimeter are likewise plotted in figure 9.1 and recorded in table 9.1.

9.2. *Field dosimeters in the range 10 to 10,000 rads* (see table 9.2).

A detailed review of a number of systems, which depend on the release of hydrochloric acid from chlorinated hydrocarbons and the consequent change of *p*H-sensitive dyes, has recently appeared (Taplin, 1956). Single-phase systems, essentially aqueous solutions containing the dye, which are saturated with the chlorinated hydrocarbon, and two-phase systems, of chlorinated hydrocarbon and aqueous dye in separate phases, have been used. The precautions necessary for their reproducible behavior with X- and γ-radiation are discussed in this review. They include the addition to the organic phase of stabilizers such as resorcinol or geraniol, the careful cleaning and sealing of the glassware, and the protection of the systems from light and heat to which they are sensitive.

These methods are applicable to research work involving doses less than 1000 rads, which cannot be measured satisfactorily with ferrous sulphate solutions. Their chief applications, however, lie

TABLE 9.2.—*Field dosimeters in the range 10–10,000 rads*

System	Mode of measurement	Dose range
		rads
Chlorinated hydrocarbonaqueous indicator, one phase.[a]	Color comparison	10 to 5,000.
Ditto, two-phase [a]	do	100 to 10,000.
KCl, contg., potassium hydride [b]	Spectrophotometry	10 to 1,000.
Solutions of dyes in $CHCl_3$ and CCl_4, and mixtures of these with EtOH.[c]	do	100 to 6,000.

[a] Taplin, 1956.

[b] Burns and Lockyer, 1955.

[c] Clark and Bierstedt, 1955; Wilkinson and Fitches, 1957.

in military and civil defense work, for which rapid dose estimation is called for without the need for high precision.

The application of fluorimetry to aqueous solutions allows determination of concentrations of appropriate substances as low as 5×10^{-11}M. Very sensitive dosimeters based on this method are under development (Barr and Stark, 1958). For example, a solution 0.1 mM in benzoic acid and 1 mM in NaOH yields hydroxybenzoates on irradiation which give a fluorescence at 400 mμ when excited by 310 mμ light. By means of this fluorescence, doses of 0.5 rad can easily be detected and a dose of 5 rads determined to ±5 percent precision. Radiation degradation of luminescence in quinine solutions (Barr and Stark, 1958) also offers a convenient and reproducible dosimetry method covering the range from 10 to 1000 rads.

9.3. *Gels for depth dose measurements.*

Rigid chemical dosimeters, which can be cut into sections after irradiation, offer obvious advantages in obtaining a picture of the depth dose distribution in a radiation field. If this distribution is to be related to that in irradiated tissue, the medium has to approach the chemical composition of tissue as nearly as possible. This requirement is met most closely by the use of gels of aqueous gelatin or agar, containing dyes, which are decolorized on irradiation. More work is needed on the linearity of response of these systems, their reproducibility, and dose rate dependence. Some systems of this type are indicated in table 9.3.

Table 9.3.—*Gels for depth dose measurements*

System	Mode of measurement	Dose range
		rads
0.001% Methylene blue in gelatin or agar gel.[a]	Colorimetry	10,000 to 50,000.
Methylene blue in agar gel [b]	Spectrophotometry	10,000 to 500,000.
Chloral hydrate-indicator agar gel [c]	Spectrophotometry and conductimetry.	2,000 to 8,000.

[a] Day and Stein, 1950.
[b] Goldblith, Proctor, and Hammerle, 1952.
[c] Andrews, Murphy, and LeBrun, 1957.

10. Solid State Dosimetry

10.1. *Dose-rate measurements by scintillation techniques.*

Lately considerable knowledge has been gained concerning the applicability of scintillation detectors for radiation dosimetry, and the subject has been reviewed by Rosman and Zimmer (1956, 1957). Though early hopes for the general usefulness of scintillation dosimeters in absolute measurements have not been realized, they have been found valuable in certain special dosimetric problems.

Scintillation dosimeters have high sensitivity and are much less dependent on temperature, pressure, and humidity than are ionization chambers; their use may therefore be convenient under difficult climatic conditions. The scintillator which forms the detecting element of such an instrument may be of small volume thus permitting detailed and exact studies of radiation dosage distribution and isodose curves (Belcher, 1953). Dose measurements with such instruments do not show any tendency to saturate at very high dosage rates (Rosman and Zimmer, 1957).

By choosing a scintillator or combination of scintillators of suitable composition, the response of such instruments may be made nearly independent of the energy of the measured radiation over a wide range of photon energies (Breitling and Glocker, 1952; Ittner and Ter-Pogossian, 1952; Cole et al., 1952; Belcher and Geilinger, 1957).

Specially constructed scintillators (Hornyak, 1952; Skjöldebrand, 1955) permit discrimination of fast neutrons against gamma rays. Such instruments are useful for protection measurements in mixed fields of fast neutrons and gamma radiation.

10.2. *Dose measurement by other solid state devices.*

Several new methods of dosimetry using solid materials have been developed in recent years. These methods are based upon measurement either of the luminescence of the material, or the transmission of light through it.

a. *Glasses.* A special silver-activated phosphate glass makes use of both techniques in order to cover a very wide range of doses (10 to 2×10^6 rad). For the range 10 to 10^4 r the U.V.-excited photoluminescence of the glass is measured by means of a fluorimeter (Schulman et al., 1951). The intensity of the luminescence is proportional to the total X-ray dose received by the glass prior to reading.

Several configurations of the glass have been used. A personnel casualty dosimeter (Schulman et al., 1953) has made use of ¾ in. square blocks of glass, $\frac{3}{16}$ in. thick. The response is linear with dose up to roughly 1000 r, where optical absorption processes begin to cause a saturation.

This glass has also been used in the form of rods 1 mm diameter by 6 mm in length (Schulman and Etzel, 1953; Riegert et al., 1956; Kondo, 1959; Hodara et al., 1959; Nucleonics, 1959; Blair, 1959; Peirson, 1958). These cover an exposure-dose range of about 30 to 10^4 r for Co^{60} rays, and are small enough for convenient implants *in vivo*.

The large energy dependence of the glass, relative to air or tissue, is a definite disadvantage, which can be reduced through the use of appropriate shielding for the rods, and these results should soon be available in the literature. (Note that because of this large energy dependence, the unshielded rods can detect exposure doses of 1 or 2 r for 200 kvp X-rays.)

This same glass can also be used to measure exposure doses in the range 10^3 to 2×10^6 r through observation of changes in its optical absorption

(Schulman, Klick, and Rabin, 1955; Rabin and Price, 1955).

Other inorganic glasses operating on the coloration principle have been developed, in particular a cobalt-activated silicate glass (Kreidl and Blair, 1956), which is claimed to have a higher useful dose range and less fading than the phosphate glass.

b. *Organics.* A number of plastics or plastic-plus-dye systems have been proposed for very high dose dosimetry, above about 10^4 r. These are listed in table 10.1, together with another method in which the degradation of U.V. photoluminescence in anthracene and *p*-quaterphenyl is used as a measure of absorbed dose. A more complete survey of dosimetry in the very high dose region is given by Nucleonics (1959).

c. *Thermoluminescence.* A thermoluminescent material stores energy given to it by a field of ionizing radiation, and later emits this energy as light upon heating of the sample. This method offers promise, particularly as a dosimeter for health physics monitoring, because of its potentially great sensitivity and wide useful dose range. A number of investigators have explored its possibilities (Daniels and Rieman, 1954; Kossel, Mayer, and Wolf, 1954; Burger, Lehmann, and Mayer, 1954; Patterson and Friedman, 1957).

TABLE 10.1.—*High dose measurement*

System	Mode of measurement	Dose range
		rads
Cellophane-dye [a]	Spectrophotometry.	10^5 to 10^7
Ag-activated phosphate glass [b, c]	do	10^4 to 10^6 10^4 to 5.10^5
Borosilicate glass containing 0.5% cobalt oxide.[d]	do	5.10^3 to 4.10^5
Pure polymethyl methacrylate [e]	U.V	5×10^4 to 4×10^6
Pure methylmethacrylate [f]	do	5.10^4 to 4.10^6
Anthracene [g, h, i]	Fluorimetry	5×10^5 to 10^8
p-quaterphenyl [h, i]	do	10^7 to 5×10^9

[a] Henley, 1954; Henley and Richman, 1956.
[b] Schulman, Klick, and Rabin, 1955.
[c] Davidson, Goldblith, and Proctor, 1956.
[d] Kreidl and Blair, 1956.
[e] Day and Stein, 1951.
[f] Boag, Dolphin, and Rotblat, 1958.
[g] Schulman, Etzel, and Allard, 1957.
[h] Attix, 1959.
[i] Nucleonics, 1959.

Most thermoluminescent materials (e.g., LiF, Al_2O_3, $CaSO_4$, Vycor) have been unsatisfactory either because of inadequate sensitivity or because of rapid "fading" at room temperature. A sensitive and stable thermoluminescent material (CaF_2:Mn) has been developed by Ginther and Kirk (1956). This has been incorporated into a dosimeter button (Schulman, Ginther, Kirk, and Goulart, 1959) which covers a range from 50 mr to at least several hundred r. An improved form of this dosimeter, particularly to lower doses, is now under development in the same laboratory.

11. Calorimetric Dosimetry

A calorimetric method was used by Curie and Laborde (1903) to determine the energy output of radioactive sources. A very sensitive null method which employs the Peltier cooling effect to balance the radiation heating has been utilized (Mann, 1954). Calorimetric methods also have been used to measure the intensity (i.e., the ergs/cm² sec) of high-energy photon beams by total absorption of the radiation in lead and measurement of the rate of rise of temperature of the absorber (Laughlin and Genna, 1956; McElhinney et al., 1957; Goodwin, 1959).

In principle, calorimetry would be the most direct method of measuring absorbed dose in rads, bearing in mind that most, but possibly not all, of the absorbed energy appears as heat. A few such measurements have been reported (Bernier et al., 1956; Skarsgard et al., 1957; Milvy et al., 1958; and Petree, 1958). In the present state of the art, it would appear that calorimetric measurement of absorbed dose is possible even for dose rates as low as $\sim$20 rads/minute in tissuelike material when the space surrounding an absorbing element can be evacuated. For large radiation intensities, where the use of ionization methods may be practically impossible owing to recombination of ions, the calorimetric technique becomes easier, as well as more direct. Sensitivity can be increased by using semiconductor thermocouples (Schwarz, 1952; Brown and Chasmar, 1953) giving some 1400 μvolts/°C.

12. Photographic Dosimetry

The photographic emulsion is one of the oldest detectors of ionizing radiation. Although the accuracy of photographic dosimetry is relatively low (5 to 10 percent, at best), the method is still widely used for personnel monitoring (Dudley, 1956; Ehrlich, 1954; Rotblatt, 1950). Photographic dosimetry is also used for measurements of total exposures up to 10^4 r and more, under conditions for which ruggedness and low cost are important factors, and for mapping of high-energy radiation fields in clinical applications. Recently, a method was developed by which the range of photographic dosimetry can be extended to 1×10^8 r through the use of conventional monitoring film without development (McLaughlin, 1959). Photographic dosimetry has been discussed in detail in the literature (Griffith, 1958). Only some of the more general features will be outlined here.

12.1. *Energy transfer to a photographic emulsion.* The energy transferred by ionizing radiation to the photographic emulsion initiates the reduction of the silver halide crystals (grains) of the emulsion to atomic silver. The microscopic silver specks formed in this way are referred to as latent image. Upon processing in special developing solutions, these silver specks then serve as nuclei for a reduction process, leading to the formation of massive silver aggregates which increase the opacity of the developed photographic emulsion. For very large radiation exposures, the reduction by the ionizing radiation alone is large enough to make any further processing unnecessary.

The increase in emulsion opacity (or in optical density, which is equal to the decadic logarithm of opacity) is usually measured by photoelectric means. By appropriate calibration procedures, optical density can be related to exposure dose, absorbed dose, or incident flux.

Charged particles transfer their energy to the silver halide grains mainly through collisions leading to atomic excitation and to ionization along the paths of the particles. The photographic effect of charged particles increases with the range of the particles in the emulsion, and—for a given range—with their specific ionization, until one single interaction with a silver halide grain is sufficient to make this grain developable. Any further increase in specific ionization leads to a decrease in the number of grains made developable for any given amount of energy dissipated within the emulsion. Photons, neutrons, and other uncharged particles lose their energy to the emulsion largely through the ionization produced by their charged secondaries.

12.2. *Response to X- and gamma radiation.*

The photographic latent image is produced by energy deposited in the grains by secondary and higher order electrons from X- and gamma-ray interactions with the material in the vicinity. Photoelectric absorption in the elements of relatively high atomic number contained in the emulsion causes the ratio of photographic effect to exposure dose to be up to 30 and more times higher for 40-kev photons than for 1-Mev photons. For dosimetry in the photon energy ranges up to about 200 kev, suitable arrangements of metallic filters aid in reducing this energy dependence, and facilitate the determination of the particular radiation energy (Langendorff, Spiegler, and Wachsmann, 1952). For higher energies, film response roughly parallels exposure dose, provided that electronic equilibrium is established at the film in air-equivalent materials.

No rate dependence effects have as yet been demonstrated for the film types usually employed in dosimetry when exposed to low-energy X-radiation (Ehrlich, 1956), unless the films were used in contact with scintillators (Ehrlich and McLaughlin, 1959). However, some rate dependence seems to be present for X-ray energies of around 1 Mev. Such rate dependence is particularly likely when the films are processed in weak developers which react mainly with the surface of the silver halide grains.

12.3. *Response to monoenergetic electrons and to beta rays.*

Photographic dosimetry of monoenergetic electrons and beta rays is complicated by the relatively high absorption of electrons both in the light-tight wrapping material usually surrounding commercial photographic film and in the emulsion proper. Actually, the ranges of electrons of energies below about 100 kev are smaller than the thickness of the average commercial photographic emulsion. As a consequence, the photographic effect increases with increasing energy up to about 100 kev and also depends strongly on the direction of electron incidence. At higher energies, the photographic effect changes only relatively little with radiation energy and direction of radiation incidence.

In some of the photographic emulsions used for electron investigations, a rate dependence has been found (Digby, Firth, and Hercock, 1953).

If elements are introduced over the film surface which become beta active upon bombardment with thermal neutrons (such as for instance rhodium), the beta-ray response of suitably calibrated films can also be utilized for thermal-neutron dosimetry.

12.4. *Response to heavy charged particles and to neutrons.*

Because of their large specific ionization and the associated waste in energy dissipation on already developable grains, the photographic effect of heavy charged particles is smaller by orders of magnitude than that of X-rays and electrons. Where the particle ranges are small compared to the emulsion thickness, the effect is also strongly dependent on the energy and the direction of the incident particles. Furthermore, in some of the emulsions, the photographic effect is rate dependent. For these reasons, track analysis on nuclear plates, which provides for a detailed study of individual events, is the preferred method of photographic dosimetry of heavy charged particles and also of fast neutrons, whose photographic action is largely due to proton recoils produced in the hydrogenous gelatine. Track analysis for personnel dosimetry of fast neutrons can be simplified by the use of a nuclear-track plate surrounded by essentially tissue-equivalent materials of such thickness that the number of tracks produced in the emulsion per unit neutron dose is essentially independent of neutron energy (Cheka, 1954). In the 30-micron Eastman NTA emulsion, this method yields about 5 or 6 tracks in a field of 2×10^{-4} cm^2 after the maximum permissible exposure for a period of 13 weeks, i.e., an exposure of 3 rems.

Analysis of the proton tracks produced by the reaction of thermal neutrons on the nitrogen of the gelatine, or of alpha tracks in lithium- or boron-loaded emulsions, can be used for thermal-neutron dosimetry. The protons produced by the reaction of thermal neutrons on nitrogen produce a slightly larger number of tracks per field than the proton recoils from elastic scattering of fast neutrons. The sensitivity of the lithium- or boron-loaded emulsions is about 200 times greater than ordinary emulsions for thermal neutrons.

13. On Measurement of Neutron Radiation Fields

The ICRU has not as yet recommended a quantity that may be utilized to describe a field of neutron radiation in a manner analogous to the characterization of an X- or gamma ray field by exposure dose. In the following discussion, and until other recommendations are made for use in the field of neutron biology, the concepts and terminologies of neutron physics are employed. A complete description of a neutron radiation field is in terms of neutron flux, neutron spectrum, and angular distribution (we can ignore polarization for this discussion). The strengths of neutron sources, especially small sources such as radioactive neutron sources and accelerator targets, are customarily expressed as total neutron emission rate in neutrons per second. The object of this report is to discuss methods of measurement of these quantities, to review the status of international intercomparisons of measurements, and to make recommendations regarding (1) preferred methods of measurement, (2) possible improvements in measurement techniques, (3) specific areas where further work is needed, and (4) encouragement of international comparisons where needed. A more complete discussion of some of these questions will be found in the report on "Measurement of Neutron Flux and Spectra for Physical and Biological Applications" of the National Committee on Radiation Protection and Measurements in the United States (NCRP-NBS Handbook 72).

13.1. *Definitions.*

General description of a neutron radiation field at a given point of space. The number of neutrons per cubic centimeter as a function of energy and direction.

Neutron flux density (usually called neutron flux). The number of neutrons per unit time entering an infinitesimally small sphere divided by the cross-sectional area of the sphere, or the number of neutrons per cubic centimeter multiplied by their mean velocity.

Neutron spectrum, or neutron energy spectrum. A description of a neutron radiation field in terms of the number of neutrons per unit energy interval versus energy. If neutron direction is important, this also should be specified.

13.2. *Neutron source strength.*

Desirable properties in a neutron source which is to serve as a standard include:

(1) neutron yield independent of time
(2) reproducible manufacture and neutron yield
(3) monoenergetic neutrons of known energy
(4) negligible gamma ray yield
(5) large neutron yield
(6) transportable
(7) small
(8) negligible self absorption and scattering of neutrons
(9) isotropic emission of neutrons
(10) ease of preparation.

No source has all of these properties, but several, which have been used as primary or secondary standards or which offer outstanding promise as source standards, are given in table 13.1.

Recent series of *direct* international intercomparisons of neutron source strengths have been carried out by Richmond and Gardner (1957) and by Larsson (1958). These intercomparisons, together with some information from the earlier comparisons of de Troyer and Tavernier (1954) and von Planta and Huber (1956) are summarized in table 13.2 (adapted from Caswell, Mosburg, and Chin, 1958). The absolute measurements listed involve all techniques in current use. Many kinds of sources are also compared: Ra-Be (α,n); Ra-Be (γ,n); Pu^{240} spontaneous fission; and, indirectly, neutron yields from Li^7 (p,n) and H^3 (d,n) reactions.

In intercomparisons of national standard radioactive neutron sources, agreement is within 2 to 3 percent of the mean value. Part of this error may be attributed to errors in the intercomparison methods which are seldom better than ±1 percent. Some of the errors which may contribute

TABLE 13.1.—*Characteristics of neutron sources important for standardization*

Source	Chief advantages	Chief disadvantages	Important data
Ra-Be(α,n)	Large neutron yield, small, most widely measured standard source.	Too much gamma radiation, unknown spectrum, not reproducible except for $RaBeF^4$, strength changes with time due to Po growth.	$T_{1/2}=1622$ yr, $\sim 1.5\times 10^7$ n/sec curie.
$PuBe_{13}(\alpha,n)$	Few gamma rays, long half life	Not monoenergetic, significant self-absorption, large bulk.	$T_{1/2}=24{,}400$ yr, $\sim 1.5\times 10^6$ n/sec/curie.
Ra-Be(γ,n)	Reproducible, isotropic	Too many gamma rays, polyenergetic spectrum below 0.7 Mev, lower neutron yield than the (α,n) source.	$T_{1/2}=1622$ yr, 1.2×10^6 n/sec curie.
Pu^{240} (Spontaneous fission).	Long half-life (6600 yr,) fission spectrum, few gamma rays.	Expensive, low neutron yield	$\sim 7\times 10^2$ n/sec/g.
$Li^7(p,n)$	Monoenergetic neutrons, few gamma rays	Requires accelerator, knowledge of flux is not too good, anisotropic, not transportable.	$\sim 10^7$ n/microcoulomb at 2 Mev; threshold energy 1.881 Mev.
$H^3(p,n)$	Same as $Li^7(p,n)$	Same as $Li^7(p,n)$	$\sim 2\times 10^6$ n/microcoulomb at 2 Mev; threshold energy 1.02 Mev.
$H^2(d,n)$	Lowest gamma ray contamination, monoenergetic neutrons.	Requires accelerator, anisotropic, associated particle counting possible, but requires knowledge of $H^2(d,n)$ and $H^2(d,p)$ relative cross sections and angular distributions.	$\sim 2\times 10^6$ n/microcoulomb at 2 Mev (thick target).
$H^3(d,n)$	Low gamma-ray field, large neutron yield for low deuteron energies, monoenergetic, rather isotropic.	Requires accelerator, excellent for associated particle counting for flux determination.	Neutron energies roughly 10-20 Mev, $\sim 10^9$ n/microcoulomb at 2 Mev (thick target).

TABLE 13.2. *International intercomparisons of standard radioactive neutron sources*

Country	Source designation	Source type	Absolutely measured source strength $\times 10^6$ n/sec	Quoted error (%)	Reference date	Relative comparison ratio, R[a]					
						Direct	Literature ref	Through Sweden[b]	Through Basel and Sweden[c]	Through Belgium and Sweden[d]	Through Oxford or NRC[l]
Belgium	Union Miniere	Ra-Be(α,n)	7.87	2	Dec 1952	.992	(e)	1.011	1.004		
Canada	NRC No. 200-1	Ra-Be(α,n)	3.22$_0$	1.6	July 1958	1.013	(f)				
France	Paris, 37 C.E.A	Ra-Be(α,n)	5.43	2.5	Oct 1950					1.029	
Germany	(Freiburg, i.B.)	Ra-Be(α,n)	.753	3.5	Apr 1955			1.077			
	P.T.B	Ra-Be(α,n)	1.949	2	Aug 1958			0.988		0.984[h]	
Italy	(Milano)	Ra-Be(α,n)	3.27	5	June 1954			.996[i]		(.920)	
Japan	E.T.L	Ra-Be(α,n)	14.8$_8$	3	Jan 1958					.997[j]	
Sweden	Stockholm	Ra-Be(α,n)	2.65	2	Oct 1954	.984	(b)			.999	
Switzerland	Basel B-2	Ra-Be(α,n)	1.518	2.8	Apr 1955	.971	(e)	.985			
Switzerland	Lausanne		.717$_7$	5	Apr 1955			.982			
U.K	(Harwell)	Ra-Be(α,n)	9.66	4.5	Mar 1951	(1.069)	(e)	(.973)[k]	(.997)[k]		
U.K	Oxford	RdTh-$D_2O(\gamma,n)$	.0618	1.6	Jan 1955	.989	(e)				
U.K	NPL3B	Ra-Be(α,n)	.161$_5$	2	1959						.981
U.S.A	NBS I	Ra-Be(γ,n)	1.264	1.7	July 1957	1.000					
U.S.A	(NBS II)	Ra-Be(γ,n)		3	July 1957	(1.000)					
U.S.A	Los Alamos, 44	Ra-Be(α,n)	6.06$_8$	5	Aug 1944	(1.016)	(e)			1.019	
U.S.A	(Argonne, 38)	Ra-Be(α,n)	5.5	7	Oct 1944	(1.049)	(e)			(1.040)	
U.S.S.R	1 II-22	Ra-Be(α,n)	5.96	3	June 1951			.979			
U.S.S.R	(II [H-26])	Ra-Be(α,n)	0.486	7	Mar 1952			(.976)			

Values in parentheses are for sources which, it is believed, are not regarded as national standards.

[a] $$R=\frac{\text{ratio of the absolutely determined source strength to the absolute strength of NBS 1}}{\text{ratio of source strength to NBS 1 from direct intercomparison (or product of such intercomparison ratios).}}$$

The value of R indicates the relative value that would be assigned to source NBS 1 by comparisons with the various other national standards.

[b] Larsson, K. E., 1958.
[c] von Planta, C., and Huber, P., 1956.
[d] Tavernier, G. C., and de Troyer, A., 1954 and in Littler, 1957.
[e] Richmond, R., and Gardner, B., 1957.
[f] Geiger, K. W., and Whyte, G. N., 1959.
[h] Through Belgium (comparison at Brussels and Braunschweig).
[i] Larsson, K. E., 1959.
[j] Michikawa, T., Teranishi, E., Tomimasu, T., and Inoue, Y., 1959.
[k] These values result from a new value for the strength of the Harwell source using the Oxford source as a national standard.
[l] Axton, E. J., 1959.

significantly to this value are (1) neutron leakage out of the water or $MnSO_4$ bath usually used to moderate the neutrons, (2) oxygen capture of neutrons during slowing down (which introduces a large correction when comparing an (α,n) source to a (γ,n) source), (3) absorption in the source of slowed-down neutrons reflected back from the bath, and (4) counting statistics (especially important for weak sources).

Recommendations:

(1) In view of the relatively good international agreement and large number of intercomparisons, work in this field is less urgent than in some others to be reviewed later.

(2) Intercomparisons in the future should emphasize *direct* intercomparisons such as measurements on many sources in one laboratory or wide circulation of a single source to many laboratories. A Ra-Be(γ,n) source, known as NBS II, is available from the National Bureau of Standards (U.S.) for the purpose of intercomparisons.

(3) New methods of absolute calibration of radioactive neutron sources should be sought which offer promise of accuracies significantly better than 1 percent. Improved methods for absolute calibration need to be essentially independent of neutron cross sections. One of the most promising of these is the measurement of helium production from $Be^9(\gamma,n)2He^4$. This work has been started in England by Martin and Martin (see Littler, 1957). Photoneutron sources now being standardized by the manganese sulfate method at the National Physical Laboratory will be calibrated by this method, the helium determination being made by Martin at Durham University. A second possibility is absolute calibration of an Sb-Be(γ,n) source in a heavy water manganese sulfate bath by a method similar to that of DeJuren and Chin (1955), and then calibration of a standard neutron source in terms of this source, by relative methods. The absolute calibration would be independent of the hydrogen cross section since there would be deuterium rather than hydrogen in the bath.

(4) The polonium growth correction factor for Ra-Be(α,n) sources of 0.17 be abandoned in favor of the 0.14 factor given in NCRP–NBS Handbook 72.

(5) To improve accuracy of relative intercomparisons of sources, measurements of the $O^{16}(n,\alpha)$ cross section and of source energy spectra are needed. Efficient methods for counting $MnSO_4$ bath activity are needed to extend this intercomparison method to weaker sources (see Richmond and Gardner, 1957; Mosburg, 1959).

(6) Spherically symmetric photoneutron sources or stoichiometric compound (α,n) sources such as $RaBeF_4$ or $PuBe_{13}$ are recommended as national standard radioactive neutron sources. In the use of $PuBe_{13}$ sources as standards, isotopically pure plutonium-239 ($\geq$99.9 percent purity) is necessary. Impurities from plutonium-238, 240, 241

may lead to growth of neutron emission. For a 94 percent Pu^{239} source, the growth in neutron emission rate was found to be 1.40_3 percent/yr, mainly from the growth of Am^{241} (Geiger, 1959). Recommendations relative to the accelerator neutron sources will be made under the discussion of neutron flux.

13.3. Thermal neutron flux.

Absolute measurement of thermal neutron flux is usually carried out either by observing the reaction rate of an element of known cross section or by basing the determination on thermalization of neutrons from a standard neutron source. For experimental comparisons, cadmium difference methods are used. In this field, therefore, knowledge of thermal neutron cross sections of elements such as boron, manganese, and gold may serve as standards. With Au detectors the cadmium difference method is not sufficient to allow the calculation by the " $\frac{1}{v}$ law" of the neutron flux from the thermal neutron cross section of gold; it is necessary to subtract the low but appreciable (1 percent at 0.025 ev) contribution of the resonance cross section. The boron cross section is for the $B^{10}(n,\alpha)$ reaction, and the latter two are activation cross sections (for Mn^{56} and Au^{198}). To utilize the boron reaction, one must know the isotopic composition accurately (i.e., the amounts of B^{10} and B^{11} in the sample).

Few international comparisons (and few others as well) of thermal neutron fluxes are known. In table 13.3. are summarized the results of these comparisons. Comparisons of thermal neutron fluxes using gold foils are planned between the Physikalisch-Technische Bundesanstalt (Braunschweig, Germany) and the National Bureau of Standards (Washington, D.C., U.S.A.). Comparisons are in progress between Los Alamos Scientific Laboratory (New Mexico, U.S.A.) and the National Bureau of Standards (Washington, D.C., U.S.A.).

Recommendations:

(1) Comparisons of standard thermal neutron fluxes between national laboratories using Au or other foils should be encouraged on a much wider basis than has occurred heretofore.

(2) Investigation should be made to determine the best method for direct intercomparisons of intense thermal neutron fluxes found in reactors without having to scale down and scale up again in flux as is currently done in flux intercomparisons.

13.4. Intermediate Energy Neutron Flux (0.5 ev to 10 kev).

Flux measurements in this region and in the fast neutron region up to 100 kev are in need of improvement. Furthermore monoenergetic neutron sources existing in this energy region have not been measured with good accuracy. No known international intercomparisons of flux have been made except indirectly through cross section measurements. In the absence of flux standards in this region, the U^{235} cross section is often assumed to be a secondary standard. This cross section is known to no better than about 3 percent above 100 kev, has a 10 percent discrepancy for thermal neutrons, and is known very poorly between 1 kev and 50 kev.

Recommendations:

(1) Development of an accurate method of flux measurement in the intermediate energy region is most needed before it will be worthwhile to intercompare measurements. Possible methods are (a) thick layer boron ("black" detectors) using the gamma rays following most of the (n,α) reactions in boron for neutron detection, and (b) foils exhibiting a neutron capture resonance in the region of interest (and covered with cadmium to exclude detection of thermal neutrons). The latter method appears very attractive in view of its

TABLE 13.3.—*Intercomparisons of thermal neutron flux measurements*

Laboratory No. 1	Laboratory No. 2	No. 1 Absolute method and quoted error	No. 2 Absolute method and quoted error	Intercomparison method	Ratio, R[a] of intercomparison	Uncertainty of intercomparison	Intercomparison reference
Aktiebolaget Atomenergi, Stockholm, Sweden (Larsson).	JENER, Kjeller, Norway (Grimeland).	Activation of Au foils in thermal column, absolute activity by β-γ coincidence method, ±2%.	Activation of NaI crystals and 4π-scintillation counting of induced activity.	Activation of Au foils from Sweden together with NaI crystals at Kjeller.	0.989	±2%	Unpublished, 1953.
AERE, Harwell, England (Littler, Lockett).	C.E.A., France (Cohen).	Beta counting of Na^{24}, Mn^{56} and Co^{60}, ±3%.	Beta activity of Mn^{56} by 4π-counting ±4%.	Counting of Cu foils at Chatillon.	1.06	±3%	Littler and Lockett, 1952.
NBS, Washington, D.C., U.S.A. (DeJuren).	Oak Ridge, Tenn, U.S.A. (Klema, Ritchie).	Absolute counting of $B^{10}(n,\alpha)$ reaction, ±2%.	β-γ coincidence counting of Au foils activated in standard graphite pile.	End-window beta counting of Au foils.	0.98	±2.5%	DeJuren and Rosenwasser, 1954.
NBS, Washington, D.C., U.S.A. (DeJuren).	C.E.A., France (Cohen).	-----do---------------	Beta activity of Mn^{56} by 4π-counting, ±4%.	4π-counting of thick Cu foils.	0.968	±1.5%	Unpublished, 1954.

[a] $R=\frac{\text{value determined for given arbitrary thermal neutron flux by laboratory No. 1}}{\text{value determined for given arbitrary thermal neutron flux by laboratory No. 2}}$, each value based on the absolute flux determination of the particular laboratory.

possible use inside reactors. A scintillation detector using a compound of ZnS and boron in a thick layer, should also be useful. The alpha particle produces scintillations in ZnS.

13.5. Fast neutron flux (10 kev to 20 Mev).[33]

Two methods seem to offer the possibility of highest accuracy in fast neutron flux measurements. These methods are proton recoil counters (based on knowledge of the hydrogen cross section) and associated particle counting. The latter method is limited to certain accelerator neutron sources, but with care should be capable of accuracies approaching 1 percent. Greater effort has been directed toward development of the proton recoil methods for which accuracies of the order of 2–5 percent are claimed.

Truly energy independent measurements of fast and intermediate neutron flux can be made in a geometry where *all* neutrons are thermalized and then detected. This implies that the method can only be used where it is possible to surround the source of neutrons by a moderator. The moderator should have a high resonance escape probability so as not to lose any fast neutrons. Moderators such as oil, water, graphite, paraffin, etc., have been used. The thermal neutron detector should have a cross section varying as 1/v; otherwise corrections are necessary. Integration to find the total neutron emission rate may be done mathematically or physically (by stirring the solution in the case of $MnSO_4$ in H_2O for example). Sensitivity and convenience are poor, accuracy is good. If activation of a dissolved element is used, sensitivity can be improved by using the Szilard-Chalmers reaction, but it is difficult to get the extraction reproducible to better than 1 or 2 percent (Edge, 1956; Littler, 1957).

Generally, fast neutron flux intercomparisons have been indirect, and frequently are referred to the fission cross section of U^{235} which serves as a secondary standard in this range. This cross section is known to about 5 percent from 50 to 450 kev and to about 3 percent between 450 kev and 1500 kev. Errors in cross section intercomparisons may enter from sources other than the neutron flux measurement. A program of absolute cross section measurements of U^{235} is of great interest; it would be desirable to make direct intercomparisons as well. The intercomparisons of standard radioactive neutron source strengths provide information on fast neutron flux measurements, but require knowledge of source spectra and detector energy sensitivity for precise interpretation. A possible direct way of comparing fast neutron flux measurements would be to use threshold reactions such as $Al^{27}(n,\alpha)Na^{24}$. Foil activations could be determined in one laboratory for exposures of the same or identical foils in known fast neutron fluxes of several laboratories. In this way, fast neutron fluxes in reactors could be related to fast neutron fluxes from accelerators (determined by associated particle counting and proton recoil counting) and radioactive neutron sources (determined by moderation methods).

Results on the intercomparison of fast neutron flux measurements are summarized in table 13.4.

One of the most widely used neutron detectors for flux determinations is the "long counter" whose energy sensitivity is rather independent of energy over a wide range of neutron energy (see for example NCRP–NBS Handbook 72). This instrument usually consists of a boron trifluoride counter in a special paraffin moderating geometry. Characteristics vary somewhat from counter to counter. Another problem is that the effective counting center moves along the counter axis with neutron energy. An attempt to design a standard, reproducible long counter from materials such as polyethylene is now under way by DePangher at the Hanford Laboratories (U.S.A.). This counter will contain provision for calibration in a standard geometry with a Pu-Be source. It is believed that intercomparisons between laboratories may be

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[33] Some authors set an upper limit of 10 Mev here.

TABLE 13.4.—*Intercomparisons of fast neutron flux measurements*

Laboratory No. 1	Laboratory No. 2	No. 1 Absolute method	No. 2 Absolute method	Intercomparison method	Ratio, R[a], of intercomparison	Uncertainty of intercomparison	Intercomparison reference
AERE, Harwell, England (Allen and Ferguson, 1955, 1956).	Aktiebolaget Atomenergi, Stockholm, Sweden (Larsson, 1955).	Proton recoil proportional counter.	Associated particle counting of alphas from $H^3(d,n)He^4$.	Intercomparison of Oxford and Stockholm standard radioactive neutron sources.	0.989	$\pm 2N$	Larsson, 1958.
AERE Harwell, England (Allen and Ferguson, 1957).	Los Alamos, New Mexico, U.S.A. (Diven, 1957).	Proton recoil proportional counter.	Proton recoils from thin radiator.	U^{235} cross section from 0.4–1.6 Mev[b].	1.0	$\pm 5N$	Henkel, 1957.
AERE, Harwell, England (Allen and Ferguson, 1957).	Los Alamos, New Mexico, U.S.A. (Bame *et al*, 1957).	Proton recoil proportional counter.	Proton recoil counter telescope.	U^{235} and U^{238} cross section 1.5–3 Mev[b].	1.13	?	Henkel, 1957.

[a] $R=\frac{\text{value determined for given arbitrary fast neutron flux by laboratory No. 1}}{\text{value determined for given arbitrary fast neutron flux by laboratory No. 2}}$, each value based on the absolute flux determination of the particular laboratory.

[b] Fluctuations in results may be caused by errors other than in the evaluation of the neutron flux, e.g., detection of the fission reactions.

improved if many laboratories build their instruments to this standard design (when available).

13.6. Neutron spectra.

Neutron spectroscopy has been a field which has undergone rather slow development due to serious technical difficulties. The most promising approaches to this field appear to be:

(1) Proton recoil analysis in photographic emulsions (0.5 to 15 Mev). Both neutron flux and spectrum can be determined for a monodirectional neutron radiation field. The emulsion should be electron insensitive; the most practical thickness is 200 μ. The main disadvantage is the laborious analysis of the proton tracks.

(2) Time-of-flight method, where applicable, offers high resolution but is of low efficiency.

(3) Observation of neutron-induced reactions. These require no collimation of the neutrons and may be used in reactors. The cross section of the element used should be well known within the range of interest and free of resonances. The method of threshold detectors is an example of endothermic reactions. Such reactions as $He^3(n, p)H^3$ in a gaseous proportional counter and $Li^6(n, \alpha)H^3$ in a cooled scintillation crystal are most promising.

Important problems in neutron spectroscopy which need investigation include:

(1) Neutron spectra of radioactive neutron sources. There are few measurements, and poor agreement among the measurements that do exist. Especially needed are Po-Be; Po-B; Pu-Be; and, perhaps most of all, Ra-Be(α,n), where the spectrum is essentially unknown. For Ra-Be(α,n), a method applicable in the presence of high gamma-ray intensities needs to be developed. Spectra which have been measured for these sources are given in the NCRP-NBS Handbook 72.

(2) Reactor neutron spectra. Measurements of the neutron spectra inside reactors present a very important but difficult problem which may be approached in two ways: (a) use of isotropic detectors which may be placed inside the reactor, or (b) measuring the spectrum with a spectrometer placed outside the reactor shield at the beam port. The problem of minimizing the effect of the walls of the duct through which the neutrons are led out is considerable. It is apparently fairly simple to extract a representative beam when the source point flux is isotropic, but it may be very difficult if the flux is seriously anisotropic (Eggler et al., 1956).

Part IV. Appendixes

Appendix I. Report of Committee I, Standards and Measurement of Radioactivity for Radiological Use [34]

The Committee has considered standards and measurements of radioactivity under the headings:

1. Absolute and relative measurements.
2. The availability of radioactivity standards.
3. The techniques of measurement of radioactivity in hospitals and biological laboratories.
4. Comparisons of radium standards.

14. Absolute and Relative Measurements

14.1. *Introduction.*

In this report two types of measurement are considered. Firstly, the preparation by absolute measurement, of so called "absolute standards" of radioactive nuclides. Secondly, the preparation, by comparative methods of measurement, of standards which may be called either reference standards or working standards; these standards will frequently be prepared by the use of a "standard instrument," calibrated by means of an absolute standard.

An absolute method is one in which a fundamental quantity such as rate of disintegration or rate of emission of energy is measured. A relative method is one which depends on a direct standardization. Terms such as "primary" and "secondary" have been used in the past to describe such standards but the present report is based on a system of nomenclature which has recently been proposed.[35]

14.2. *Absolute measurements.*

In view of the comprehensive review articles and detailed technical papers [36] which are available on absolute methods of measurement, a brief survey only is deemed necessary for the purpose of this report. A list of the methods applicable to nuclides used for radiological purposes is given in table 14.1.

The absolute measurement of the activity of a radioactive nuclide involves either the direct measurement of the rate R at which the radioactive nuclei in the sample are disintegrating or of the determination of the number N of unstable nuclei present in the sample and the decay constant of the nuclide. R and N are related by the radioactive decay law $R=\lambda N$. When $R=3.700 \times 10^{10}$ per second, the quantity of radionuclide present in the sample is 1 curie.

The most common modes of decay of radioactive nuclei are those involving the emission of an α or a β particle with or without associated gamma radiation, or capture of an electron from one of the atomic electron shells with emission of X-rays characteristic of the daughter nucleus, again with or without associated gamma-ray emission from the daughter nucleus.

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[34] Prepared by Committee I.

[35] A bibliography is appended.

[36] A bibliography is appended.

TABLE 14.1. *Absolute methods for the preparation of standards of radioactive nuclides*

Method	Nuclides to which method is applicable	Activity of measured source	Special corrections, limitations, etc.	Accuracy under most favorable conditions*
1. 4π *Counting*				
(a) Geiger-Müller counter	α emitters, β emitters, with or without γ rays.	$0.01\mu c$ or less	Accuracy usually limited by uncertainty in corrections for loss of β particles in source material and source mount, particularly for β particle energies <0.5 Mev.	±2%
(b) Proportional counter	do	do	do	±2%
(c) High pressure proportional counter.	Electron-capturing nuclides	do	Method assumes knowledge of K-fluorescence yield and K/L capture ratio.	±5%
(d) Scintillation counting; solid phosphor.	β emitters, with or without γ rays.	do	Suitable for medium and high energy β emitters; low energy cut-off limits accuracy for low energy β emitters.	±2%
(e) Scintillation counting; liquid phosphor.	β emitters, with or without γ rays, α emitters.	do	Suitable for medium and high energy β emitters; low energy cut-off limits accuracy for low energy β emitters.	±2%
2. *Defined Solid Angle*				
(a) End window Geiger-Müller counter.	β emitters	$0.1\mu c$ or more	Complex corrections for absorption and scattering effects	±5%
(b) Proportional counter	α emitters	do		±0. 5%
	β emitters	do		±5%
(c) Zinc sulphide screen	α emitters	do		±0. 5%
3. *Coincidence Counting*				
(a) $\beta\gamma$	$\beta+\gamma$ emitters having simple decay schemes.	$1\mu c$ or less	Uncertainty in the correction for the gamma sensitivity of the β detector usually limits accuracy.	±2%
(b) $4\pi\beta\gamma$	Many $\beta+\gamma$ emitters	$0.1\mu c$ or less	Corrections, in general, very small and hence method is more accurate than $\beta\gamma$ method.	±1%
(c) $\gamma\gamma$	So far chiefly used for Co^{60}.		Requires careful adjustment of efficiencies of counters, and allowance for angular correlation of γ rays.	±1%
(d) $X\gamma$ and $4\pi X\gamma$	Electron-capturing nuclides	$0.5\mu c$ or more		±5%
4. *Internal gas counting*	Low-energy β emitters Electron-capturing nuclides.	$0.01\mu c$ or less	Limited to nuclide samples having suitable gaseous form	±2%
5. *Measurement of loss of charge*	β emitters	0.1mc	Application limited by experimental difficulties	±5%
6. *Calorimetry*	α emitters	1mc or more		±0. 5%
	β emitters	1mc or more	Involves a knowledge of the mean energy of the β particles	±2%
	$\beta+\gamma$ emitters	1mc or more	Involves a knowledge of the mean energy of the β particles, the quantum energies of the γ rays, the number of γ quanta of each energy per disintegration and the fraction of the total γ radiation absorbed in the calorimeter.	±5%
7. *Ionization measurement of dose rate.*	γ emitters having relatively simple γ-ray spectra.	1mc or more	Accuracy of disintegration rate determination limited by uncertainty in W (energy per ion pair) for air and in some cases by incomplete knowledge of γ-ray spectra.	±4%
	β emitters		Accuracy limited by uncertainty in W and in average energy of β particles.	±5%
8. *Weighing*	Radium and long-lived radionuclides.	10 mg	High chemical and radio-chemical purity of nuclide or parent essential.	±0. 2%†
9. *Mass spectrometer*	Has been used for H^3			±1% (routinely)

*The figures given in this column are estimates of the extreme limits of total error in the disintegration-rate value, and the errors which contribute to this total depend on various factors which are not the same for each nuclide; the limits given indicate the accuracy believed to be obtainable under the most favorable conditions presently available.

†Radium standards.

14.2.a. *Alpha-particle standards.* Standard α-particle sources have been prepared from Po^{210}, uranium oxide, Pu^{239} and Am^{241} and thin sources of these can be satisfactorily calibrated in a $2\pi\alpha$ proportional flow counter. Sources of Po^{210} can also be calibrated in terms of α-particle emission by measuring their rate of energy emission microcalorimetrically or by finding their total γ-ray emission. The α-particle emission is then obtained using the energy of the α-particles or the branching ratio respectively.

14.2.b. *4π Geiger-Müller or gas-flow proportional counters.* For the absolute measurement of the activities of β emitters the $4\pi\beta$ method, using GM or gas-flow proportional counters has been adopted for routine use by most standardizing organizations. The relatively simple techniques for preparing thin conducting films have almost eliminated the need for a correction for loss of β particles by absorption in the source mount.

Current experiments indicate that losses due to absorption in the source material can be reduced to a minimum by employing certain techniques in the preparation of solutions and sources, e.g., in the measurement of the activity of S^{35}, a weak β emitter ($E_{max}=0.167$ Mev), carrier-free solutions are used and sources are redissolved and dried in an ammonia atmosphere.

The activities of samples of electron-capturing nuclides can be measured by counting the X-photons in a 4π-gas-proportional counter. The pressure in the counter is varied over a range from one-half to several atmospheres, and the rate corresponding to infinite pressure (i.e., complete absorption of the X-rays) is derived by extrapolation: alternatively, the counter is operated at a pressure sufficiently high to ensure practically complete absorption. The derivation of disintegration rate from the counting rate involves a knowledge of the fluorescence yield of the daughter nucleus and a correction for the absorption of the X-rays in the foil used to absorb the Auger electrons. Agreement between this method and the coincidence method referred to later is good.

14.2.c. *4π Scintillation counters.* Scintillation counting of β-particles in a 4π geometry using either solid or liquid phosphors has been shown to give good agreement with gas-proportional counting, and has also been used for α-particle emitters. However, this technique is not suitable for the absolute measurement of the activities of low energy β emitters because the detection efficiency is very low for β particles having energies below a few kilovolts, and these form an appreciable fraction of the total. Moreover the background of thermal noise pulses from the electron-multiplier phototube reduces the accuracy of measurement for β particles below 10 kev. This background can be reduced by cooling and by using two or more phototubes in coincidence, but the latter method also reduces the sensitivity. The method may however be applicable to electron-capturing nuclides of sufficiently high atomic number because of the great efficiency of phosphors for the detection of X-rays.

14.2.d. *Low-geometry defined-solid-angle counting.* End-window counters subtending a well-defined solid angle at the source have been used to measure a known fraction of the β particles emitted by a source. Although it is possible with care to reduce the errors associated with the corrections for absorption and scattering, the method is now seldom used for the direct standardization of β emitters.

Low-geometry defined-solid-angle systems in which the detector is a zinc sulphide screen and phototube have been shown to be capable of high precision in measurements of α-particle emitters.

14.2.e. *Coincidence counting.* The βγ coincidence method provides an alternative, or an auxiliary, to 4πβ counting of nuclides for which the β radiation is accompanied by γ rays, and for which the decay schemes are relatively simple and well-known, e.g., Co^{60}. Two detectors are required, one for β particles and one for γ photons, and either GM counters, proportional counters or scintillation counters are used. The method is generally slow unless scintillation counters are used in one or both channels; short resolution times (of the order of 10^{-8} second) are desirable in the coincidence circuits to reduce the random-coincidence rate when fast counting rates are employed. The corrections involved in the coincidence method are very much reduced in magnitude when the β detector is a 4π counter as in the so-called 4πβγ coincidence technique. Restriction to simple decay schemes is then less stringent.

The γγ coincidence method is of limited application but has been successfully used for Co^{60}. Xγ coincidence counting is used for the absolute measurement of the activities of electron-capturing nuclides for which the decay is accompanied by γ radiation, and has been used to check the validity of the extrapolation method described above.

14.2.f. *Internal gas counting, loss-of-charge method, calorimetry.* The absolute measurement of the activities of low-energy β emitters and electron-capturing nuclides has been effected by internal gas counting; the more general use of this method is desirable, but limitations are imposed by the difficulty of preparation of suitable gases.

The loss-of-charge method, in which the number of charged particles emitted by a sample is determined by measuring the total charge lost by the sample, is complicated in practice by secondary effects, and is not generally a suitable alternative to the direct counting methods.

Calorimetry has been used for the intercomparison of national radium standards and for the standardization of α emitters and of β emitters for which the average energy of β decay is known. It is also used in conjunction with other measurements to determine the mean energy of β emission, especially when this is low. Calorimetry of γ-ray activity involves a knowledge of the γ-ray spectra and of the absorption of the γ radiation in the calorimeter, and is restricted in application and accuracy.

14.2.g. *Dosimetry:* In the measurement of activities by γ-ray dosimetry the experimental part consists of the determination of the exposure-dose rate at a convenient distance from the sample; the disintegration rate is then obtained by dividing the dose rate at 1 cm, derived from this measurement, by the calculated value of the specific γ-ray emission (r-cm^2/mc-h) of the nuclide (table 14.6.). The accuracy is limited by uncertainties in the values of a number of quantities on which the calculated value of the specific γ-ray emission depends, e.g., the energy required to produce an ion pair in air, and any uncertainty in the γ-ray spectrum of the nuclide. (For this reason, the measurement of disintegration rate by one or more alternative methods is often combined with exposure-dose-rate measurements in order to obtain experimental values of specific γ-ray emission, thereby providing a reliable basis for the preparation of reference or working standards, particularly of nuclides having complex γ-ray spectra.)

For β emitters, disintegration rates have been determined from measurements of the ionization current in a cavity chamber lined with a medium containing the radioactive material, or in a parallel-plate extrapolation chamber, one plate of which is formed by a solution of the material. The conversion from ionization current to disintegration rate involves the mean energy of β decay (table 14.7.) and the average energy required to produce an ion pair. (Conversely, one of these two quantities can be determined if the other is known and a sample of known disintegration rate is used.)

14.2.h. *Radiochemical purity.* Because of possible changes in radiochemical composition, it is preferable not to use short-lived standards for long periods of time, even though the half-life of the nuclide in question may be accurately known.

Even in freshly-prepared standards unknown amounts of other isotopes may be present, e.g., P^{33} in P^{32}. It would be desirable for known radioactive contaminants to be specified by the supplier.

14.3. *International comparisons of standards.*

A list of the national laboratories participating in comparisons of measurements of standards is given in table 14.2. together with the methods used. Standards have been distributed for this purpose (column 2 table 14.2.) by the NBS [37] (C^{14}, Na^{24}, P^{32}, K^{42}, Co^{60}, Zn^{65}, Sr^{90}+Y^{90}, I^{131}, Au^{198}, Radium-226); the NPL (Na^{24}, P^{32}, Co^{60}, Sr^{90}+Y^{90}, I^{131}, Au^{198}); the AECL (H^{3}, S^{35}, P^{32}) and the CEA (S^{35}). The standards were ampoules containing a few milliliters of solution of the nuclide; the activities ranged from a few microcuries to a few tens of microcuries per milliliter of solution.

TABLE 14.2.—*National laboratories participating in comparisons of standards, and methods of measurement used*

Laboratory	Laboratories from which standards were received	Methods used by the laboratory (column 1) in comparisons of measurements of standards
Atomic Energy of Canada Limited, Chalk River (AECL).	NBS, NPL	$4\pi\beta$(PC), $4\pi\beta$(PC)γ
Atomic Energy Research Establishment, Harwell (AERE).	AECL, NBS	$4\pi\beta$(GM) $\beta\gamma$(GM) $4\pi\beta$(GM)γ(LS) $4\pi\beta$(PC), $4\pi X\gamma$
Institut für Angewandte Radioaktivität, Leipzig (IAR).	AECL, NPL	$4\pi\beta$(GM), 4π(PC) $\beta\gamma$(GM)
Commissariat a l'Énergie Atomique, Saclay (CEA).	NBS, NPL	$4\pi\beta$(GM), $4\pi\beta$(PC), $\beta\gamma$, DSA
Fondation Curie, Paris (FC)	NBS	$4\pi\beta$(PC)
Kernreaktor Isotopen Laboratorium Karlsruhe (KIL).	NBS	$4\pi\beta$(PC)
National Bureau of Standards, Washington (NBS).	AECL, NPL	$4\pi\beta$(PC), $\beta\gamma$, $\gamma\gamma$, LS calorimetric
National Physical Laboratory, Teddington (NPL).	AECL, CEA, NBS.	$4\pi\beta$(PC), $4\pi\beta$(PC), $\beta\gamma$ $4\pi\beta$(SS)
National Physical Research Laboratory, Pretoria (NPRL).	AECL, NBS, NPL.	$4\pi\beta$(PC), $4\pi\beta$(LS), $4\pi\beta$(PC)γ, $4\pi\beta$(LS)γ
National Research Council, Ottawa (NRC).	AECL, NBS	$4\pi\beta$(PC), $4\pi\beta$(PC)γ
Max Planck Institute, Göttingen (MPI).	CEA, NBS, NPL	$4\pi\beta$(GM), $4\pi\beta$(PC), $\beta\gamma$
Physikalisch-Technische Bundesanstalt, Braunschweig (PTB).	NBS, NPL	$4\pi\beta$(GM), $4\pi\beta$(PC), $\beta\gamma$ $4\pi\beta$(PC)γ

The symbol (GM) denotes Geiger-Müller, (PC) proportional gas-flow, (LS) liquid scintillation and (SS) solid scintillation counter. DSA denotes defined-solid-angle.

The results of measurements of the standards by the different organizations generally agree within ±2 percent; in some cases there is a spread of 5 percent in the results, and only occasionally are there wider differences. Brief statements on the comparisons are given below, and several are summarized in table 14.3.

Hydrogen-3

Samples of H^{3} distributed by AECL were measured at a number of research laboratories in North America, including AECL and NBS and also by the University of Glasgow in 1953. Agreement to within about ±2.5 percent was obtained.

TABLE 14.3.—*International comparisons of standards*

Nuclide	Dates of comparisons	Laboratory issuing standard	Participating laboratories	Spread in values
Na^{24}	1955–57	NBS	NBS, NPL	% 2
	1957	NBS	AECL, NBS, NPL	3
	1956–57	NPL	NPL, MPI	2
	1957	NPL	MPI, PTB, NPL, NPRL	3
	1958	NPL	AECL, IAR, NBS, NPL, PTB	3
P^{32}	1955–56	NBS	NBS, NPL	2
	1957	NPL	AECL, CEA, MPI, NPL, NPRL, PTB.	3
	1958	NPL	IAR	1
S^{35}	1955–56	AECL	AECL, AERE, NBS, NPL	4
	1958	CEA	CEA, MPI	2
K^{42}	1956–57	NBS	AECL, NBS, NPL	2
Co^{60}	1957	NPL	MPI, NPL	2.5
	1958	NPL	IAR, NPL	1.8
	1958	NPL	NPL, PTB	1
Zn^{65}	1958	NBS	AERE, NBS	1.5
Sr^{90}+Y^{90}	1957	NBS	NBS, NPL	0.6
	1957	NPL	NBS, NPL, MPI	2
	1958	NPL	NPL, IAR	1
	1958	NPL	NPL, PTB	1
I^{131}	1955–57	NBS	AECL, NBS, NPL	3
	1955	NPL	CEA, MPI, NPL	3.5
	1956–57	NPL	NPL, MPI	2
	1957	NPL	MPI, NPL, NPRL, PTB	4
	1958	NPL	AECL, NBS, MPI, NPL, PTB	3
Au^{198}	1955–56	NBS	AECL, NBS, NPL, NPRL	4
	1957	NPL	AECL, CEA, NPL, MPI, PTB	4
	1958	NPL	NPL, PTB	2
	1958	NPL	NPL, IAR	5
	1958	NBS	AECL, CEA, FC, KIL, NBS, NPL, NPRL, NRC, PTB.	5

Sodium-24

The NPL measured NBS standards on three occasions during 1955–57; NBS $4\pi\beta$ values were on the average about 1 percent greater than NPL values. In an intercomparison between NBS, AECL and NPL in September, 1957 the spread of the individual values was about 3 percent.

The MPI values for NPL Na^{24} standards on four occasions during 1956–57 are on the average 1 percent greater than NPL values. NPL standards were measured by the MPI, PTB and NPRL in 1957, the spread in the results (excluding one value) being 3 percent. In 1958 the spread in the results of measurements of NPL standards by AECL, NBS, NPL PTB and IAR was also 3 percent.

Phosphorus-32

NPL measurements of three NBS P^{32} standards during 1955–56 agreed with NBS measurements to within ±1 percent, and measurements of NPL standards by the AECL, CEA, MPI, NPRL and PTB in 1957 agreed to within ±1.5 percent.

Sulfur-35

S^{35} standards were distributed by AECL for intercomparison in 1955. Results obtained by AERE, NBS, NPL, and the Royal Cancer Hospital agreed to within about ±2 percent of the

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[37] Please see table 14.2. for listing of laboratory abbreviations.

AECL value. Results obtained by McGill University by $4\pi\beta$ counting were, however, some 10 percent higher than the AECL value. Measurements of CEA S^{35} standards by CEA and MPI in 1957 agreed to within 2 percent.

Potassium-42

Agreement within ±1 percent was obtained between measurements of NBS standards in 1956 and 1957 by NBS, AECL, and NPL.

Cobalt-60

A sample of a Co^{60} solution prepared by the NPL in 1955 and standardized by the $4\pi\beta\gamma$ coincidence method was measured by the MPI in 1957 by the $\beta\gamma$ coincidence method: the MPI value, after correction for decay, was 2.4 percent greater than the NPL value. A solution, also standardized at the NPL by the $4\pi\beta\gamma$ coincidence method in 1957 was measured by the IAR in 1958 by the $\beta\gamma$ coincidence method using GM counters; the IAR value was 1.8 percent greater than the NPL value. The PTB values by $4\pi\beta\gamma$ coincidence counting in 1958 were up to 1 percent greater than the NPL values.

Zinc-65

Measurements by AERE of NBS Zn^{65} standards in 1957 agreed with NBS measurements to within ±1.5 percent.

Strontium-Yttrium-90

Measurements by the NPL of NBS standards in 1957 agreed with NBS values to within ±0.3 percent. Measurements of NPL standards by the MPI and the NBS in 1957 and by the IAR and PTB in 1958 agreed with NPL values to within ±1 percent.

Iodine-131

The average of values obtained by the NPL by $4\pi\beta$ measurements of three NBS standards during 1955–57 was 2 percent less than the NBS average (which includes a correction of 2 percent for self-absorption); the AECL values in 1957 by the $4\pi\beta\gamma$ coincidence method were on the average 1.6 percent less than the NBS $4\pi\beta$ value (corrected for self-absorption).

In measurements of NPL standards during 1955–58, NBS $4\pi\beta$ values, with added absorption correction, and AECL $4\pi\beta\gamma$ values exceeded NPL values by about 2 percent, NPRL values by $4\pi\beta\gamma$ counting (using both liquid scintillation and proportional $4\pi\beta$ counters) exceeded NPL values by 1.5 percent, and CEA $4\pi\beta$ and $\beta\gamma$ values exceeded NPL values by about 3.5 percent. MPI and PTB values have agreed with NPL values to within ±1 percent.

A remarkably close gamma-ray analog for iodine-131, whose principal gamma rays have energies of 335 and 608 kev, has been produced by mixing the correct proportions of barium-133 (82-, 300-, and 375-kev gamma rays) and cesium-barium-137 m (662-kev gamma rays). When such a simulated standard is used to calibrate a true iodine-131 sample, its effect as measured by any kind of gamma-sensitive equipment will be approximately the same as for iodine-131. It may thus have uses for calibration to within say ±3 percent when standards of short-lived iodine-131 itself do not happen to be available. A set of such "mock-iodine" standards consisting of two nominal 10-microcurie and two 50-microcurie ampoules was calibrated in the summer of 1955 at the NBS, MPI, and the NPL. The results, obtained by ionization chambers, Geiger-Müller and scintillation counters at these three laboratories did not differ by more than ±4 percent from the mean value.

Gold-198

Except for results which are in doubt due to self-absorption, measurements during 1955–56 of NBS standards by AECL and NPL on three occasions, and by NPRL on one, agreed to within ±2 percent. Measurements of NPL standards by AECL, CEA, MPI, and PTB in 1957–58 also agreed to within about the same limits, as did measurements of NBS standards by AECL, CEA, FC, KIL, NPL, NPRL, NRC, and PTB.

Radium-226

New radium solution standards for radon analysis containing 10^{-9} and 10^{-11}g of radium-226 in 100 ml of carrier solution have been recently prepared at the National Bureau of Standards. Measurements of the 10^{-9}g standards at PTB agreed to within 0.3 percent with the NBS certified values. NBS measurements of PTB standards containing 4×10^{-9}g of radium-226 in 10 ml of solution agreed to within 0.5 percent of the PTB certified values.

14.4. *Relative measurements: preparation of reference or working standards.*

The preparation of a reference or working standard involves the comparison of the activity of a sample of a radioactive nuclide with that of a sample of the same nuclide, the activity of the latter sample having been established by one or more absolute methods. The comparison may either be made directly, using any suitable measuring system or through the medium of a calibrated standard instrument. Identity of conditions (form, geometry, etc.) for sample and standard or a knowledge of the corrections allowing for departure from identity is necessary. In principle any equipment which detects ionizing radiations can be used for relative measurements. For all such instruments it is necessary to use a suitable, long-lived, reference source in order to check the stability of the instrument over long periods of time. Counting systems have a wide application, especially for low activities, but they generally require frequent re-calibration. Halogen counters, when used under reproducible geometrical conditions (e.g., annular-type liquid counters for β emitters, well-type counters for γ emitters, etc.) maintain their calibration to within ±1 percent over long periods (see also 15.3). Other types are more variable and all have limited operational life. Comparisons of low-energy β emitters can sometimes be carried out using liquid-scintillation, organic-solution, or radioactive-gas counting.

The long-term reproducibility and simple operational characteristics of ionization chambers and current-measuring apparatus has led to a preference for this type of equipment for relative measurements. An optical-bench system using a cavity chamber provides the simplest conditions for precise direct comparisons of γ-ray sources when it is important to allow accurately for differences in their dimensions and form; for such systems activities of at least 10 μc are required. Chambers of the re-entrant cylinder (or well) type, known colloquially as $4\pi\gamma$ chambers, have proved very satisfactory for samples emitting gamma-rays (table 14.4.) and chambers having 2π geometry,

TABLE 14.4.—*Particulars of "$4\pi\gamma$" chambers used at national laboratories for the relative standardization of samples of radioactive nuclides*

Laboratory	Main dimensions of chamber			Size of cavity		Background current (inclusive of leakage)	Activity equivalent to background	Current-measuring apparatus	Remarks
	Ht	Diam	Vol	Depth	Diam				
	cm	*cm*	*liter*	*cm*	*cm*	*amp*			
National Bureau of Standards (USA).	30	25	15	23	2.5	3.6 x 10^{-14}	0.3 μc Co^{60}	Lindemann-Ryerson Electrometer (automatic null method with kilocycle time measurement).	Response not critically dependent on source position and length.
National Physical Laboratory (UK):									
A	18	16.5	4	13	6.8	2 x 10^{-14}	0.3 μc Co^{60} 0.5 μg Ra	Lindemann Electrometer (null method).	Response not critically dependent on source length or diameter.
B[a]	18	16.5	4	12	6.6	1–2 x 10^{-14}	0.3 μc Co^{60} 0.5 μg Ra	do Also D.C. amplifier	Available commercially (with calibration data).
Commissariat à l'Énergic Atomique:									
A	34	13	2.3	34	7.5	5 x 10^{-14}	0.5 μc Co^{60}	D.C. Amplifier	Argon; 3 atmospheres. Range 2μc—1c Ra.
B	36	13	2	30	8	5 x 10^{-14}	0.05 μg Ra	do	Argon; 20 atmospheres. Range 0.2 μc to 1c Ra.
Atomic Energy of Canada Limited.	28.5	17.1	13	32	7.9	7 x 10^{-14}	0.02 μc Co^{60}	Vibrating-reed electrometer and strip-chart recorder.	Argon; 20 atmospheres. Response not critically dependent on source position.
Physikalisch-Technische Bundesanstalt.	30	25	14.5	19	3.4	4 x 10^{-14}	0.33 μg Ra	D.C. Amplifier	Response not critically dependent on source position and length.

[a] Combined $\beta+\gamma$ chamber designed by National Physical Laboratory Advisory Committee on Radioactive Standards.

TABLE 14.5.—*Particulars of beta-ray ionization chambers used at national laboratories for secondary standardization of samples of radioactive nuclides*

Laboratory	Geometry	Form	Source	Background current (inclusive of leakage)	Activity equivalent to background	Current measuring apparatus
				amp		
National Bureau of Standards (USA).	2π internal source	Sphere 15-cm diam	Dried deposit on disk.	5 x 10^{-15}	0.001 μc	Lindemann-Ryerson electrometer (automatic null method with kilocycle time measurements).
National Physical Laboratory (UK):						
A	2π (approx.) external source under wire mesh window.	Cylinder 7.5-cm diam. Cylinder 3.7-cm deep. Alkathene walls lined with graphite. Axial central electrode.	do	10^{-14}	0.003 μc	Lindemann electrometer (null method).
B[a]	$<2\pi$ external source under dural foil window.	Parallel plate	1 ml liquid in plastic dish.	10^{-14}	0.04 μc P^{32}	Lindemann electrometer (null method); D.C. amplifier.
Commissariat à l'Énergie Atomique[b].	$<2\pi$ external source	do	1 ml liquid in cylindrical vessel.	10^{-14}	1 μc P^{32} 10 μc Co^{60}	D.C. amplifier.

[a] Combined $\beta+\gamma$ chamber designed by National Physical Laboratory Advisory Committee on Radioactive Standards.
[b] Combined $\beta+\gamma$ chamber.

or an approximation thereto, for relative measurement of β activity (table 14.5.). Chambers providing "$4\pi\gamma$" and "$2\pi\beta$" facilities in separate or in single units have also been used. Forty commercially available composite $2\pi\beta$–$4\pi\gamma$ chambers, based on a design of the NPL Advisory Committee on Radioactive Standards, were tested at the NPL and found to have calibration factors differing by not more than ±1 percent from the averages for standard γ-ray sources, and by not more than ±3 percent from the averages for β-emitting sources.

14.5. *Specific gamma-ray emission.*

According to the ICRU (1956) the specific gamma-ray emission is the exposure dose rate produced by the unfiltered gamma rays from a point source of a particular nuclide at a defined distance, and the unit for this quantity is roentgen per millicurie hour at 1 cm; this unit has dimensions r cm²/mc-h.

Theoretical and experimental values of the specific γ-ray emission (r/mch at 1 cm, or r-cm²/mc-h) are given in table 14.6. It should be noted that the calculated values do not include any contribution to the exposure-dose rate made by X-rays, e.g., bremsstrahlung or X-rays following internal conversion or electron capture.

The experimental values given in the final column have been derived from measurements of the disintegration rates and exposure dose rates for samples of the nuclides; the dose rates were measured using chambers calibrated with radium sources taking the specific γ-ray emission, Γ_{Ra}, for a source filtered with 0.5 mm platinum as 8.25 r/mgh at 1 cm. (The values should be multiplied by $\Gamma'_{Ra}/8.25$ if Γ_{Ra} is changed from 8.25 to Γ'_{Ra} r/mgh at 1 cm.) Consideration of the methods used in measuring exposure-dose rates indicates that reasonable agreement may be expected between theoretical and experimental values for nuclides such as Co^{60} for which the exposure-dose rate is produced almost entirely by high energy γ radiation. However, for nuclides such as Tm^{170} for which the γ radiation is of low energy and the exposure-dose rate due to X-rays may be appreciable and depend markedly on the form of the source, calculated values of the specific γ-ray emission must be used with caution in any radiological application. For such applications of these nuclides it is advisable to make a direct determination of the exposure-dose rate for a particular source.

14.6. *Average β-particle energies.*

Values of the average β-particle energy per disintegration ($\overline{E}_\beta$) are given in table 14.7. for some radionuclides which are of radiological interest. The calculated values, with two exceptions, are those given by R. Loevinger, and include the energy of Auger and conversion electrons and of the characteristic X-rays following electron capture and internal conversion.

TABLE 14.6.—*Specific gamma-ray emission*

Nuclide	Quantum energy in Mev and probability of emission		Specific gamma-ray emission, Γ (r-cm²/mc-h)	
	Annihilation	Nuclear gamma radiation	Theoretical [a]	Experimental [b]
Na^{22}	0.511(1.8)	1.28 (1)	11.9_1	
Na^{24}		1.37 (1)		18.7 [c]
		2.75 (1)	18.2_3	
K^{42}		1.53 (0.2)	1.3_6	1.4 [c]
Mn^{52}	.511(0.66)	0.73 (1)	18.5_2	
		0.94 (1)		
		1.45 (1)		
Mn^{54}		0.84 (1)	4.7_0	
Co^{58}	.511(0.3)	0.81 (1)	5.4_4	
Fe^{59}		0.191 (0.03) [d]	6.2_4	6.8 [e]
		1.098 (0.57)		
		1.289 (0.43)		
Co^{60}		1.173 (1)	12.9_6	13.1 [c]
		1.332 (1)		
Cu^{64}	.511(0.38)	1.34 (0.01)	1.1_9	
Zn^{65}	.511(0.034)	1.11 (0.49)	2.9_9	
Br^{82}		0.25 (0.06) [f]	14.6_4	15 [c]
		0.35 (0.03)		
		0.554 (0.75)		
		0.619 (0.42)		
		0.698 (0.28)		
		0.777 (0.83)		
		0.827 (0.25)		
		1.044 (0.28)		
		1.317 (0.27)		
		1.475 (0.17)		
Rb^{86}		1.08 (0.085)[g]	0.4_9	0.50 [h]
Mo^{99}		Many lines		1.29 [i]
Sb^{124}		0.60 (0.98)	9.8_2	
		0.64 (0.07)		
		0.72 (0.14)		
		0.97 (0.03)		
		1.05 (0.02)		
		1.33 (0.02)		
		1.37 (0.11)		
		1.45 (0.02)		
		1.61 (0.46)		
		2.11 (0.07)		
I^{130}		0.41 (0.23)	12.1_6	
		0.53 (1.0)		
		0.66 (1.0)		
		0.74 (0.69)		
		1.15 (0.31)		
I^{131}		0.08 (0.02)	2.2_0	2.23[c]
		0.28 (0.05)		
		0.36 (0.80)		
		0.51 (0.01)		
		0.64 (0.09)		
		0.72 (0.03)		
I^{132}		2.2 (0.02)	12.3_3	12.0 [j]
		1.96 (0.04)		
		1.40 (0.13)		
		1.16 (0.10)		
		0.96 (0.23)		
		0.78 (0.18)		
		0.67 (1)		
		0.62 (0.06)		
		0.53 (0.18)		
$Cs^{137}+Ba^{137m}$		0.66 (0.81)[d]	3.0_8	3.4[k]
Tm^{170}		0.084 (0.03)[g]	0.01_1	0.04[i]
Ir^{192}		Many lines		5.0[e]
Au^{198}		0.41 (0.95)	2.2_7	2.32[c]
		0.68 (0.01)		
Ra^{226}+decay products (0.5 mm Pt filter).		Many lines		8.25[l]

[a] For all the nuclides, with the exception of those based on reference (d), (f) and (g), the theoretical values are based on nuclear data supplied by the U.S. National Research Council Nuclear Data Group to the U.S. National Committee on Radiation Protection and Measurements for publication in the NCRP Manual on Radioactivity Procedures. The values of Γ have been calculated using a dose-rate versus photon-energy curve calculated by K. W. Geiger from the data given in table 8.1. for a value of $W=34.0$ ev per ion pair. The initial preparation of this table was carried out by N. G. Trott and Miss F. S. Lawson and the final calculation of Γ by W. B. Mann.

[b] The references given do not cover all published work.

[c] Perry, W. E. and Dale, J. W. G. (National Physical Laboratory) unpublished. The values are based on the specific gamma-ray emission of radium which has been taken to be 8.25 r-cm²/mc-h.

[d] Hine, G. J., and Brownell, G. L. Radiation Dosimetry Appendix "Tables of Data Useful for Dose Calculations" p. 897 (Academic Press, New York. 1956).

[e] National Physical Laboratory Advisory Committee on Radioactive Standards. (Unpublished).

[f] Aten, A. H. W., Private Communication.

[g] Allen, R. A., Millett, R. J., and Smith, D. B., "Radioisotope Data". A.E.R.E.—R.2938. (H.M.S.O., London, 1959).

[h] Emery, E. W., Bradley, J. E. S., and Veall, N., Nature **175**, 34 (1955).

[i] Sinclair, W. K., Trott, N. G., and Belcher, E. M., Brit. J. Radiol. **27**, 565 (1954).

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[l] Garrett, C., Can. J. Phys. **36**, 149 (1958). The unit for this number is r-cm²/mg-h. Wyckoff, H. O., NBS Tech. News Bull. **41**, 198 (1957).

TABLE 14.7.—*Average β-particle energies*

1	2	3	4	5	6	7	8	9
Nuclide	Radiation emitted	Component max. energy (Mev)	Component frequency (fract/dis)	Conversion electrons (kev/dis)	Auger electrons (kev/dis)	Characteristic radiation (kev/dis)	Calculated average energies $\bar{E}_b$ in Mev/dis	References for columns 3–7
H^3	β^-	0.018	1	------	------	------	0.0055	SW
C^{14}	β^-	.156	1	------	------	------	.050	SW
Na^{22}	$\beta^+ \gamma$	.544	.898	<1.3				
	E.C	[.0009]	.102	------	0.1	0.0	.193	SW
Na^{24}	$\beta^- \gamma$	1.392	1.00	------	------	------	.56	SW
P^{32}	β^-	1.710	1	------	------	------	.70	SW
S^{35}	β^-	.1673	1	------	------	------	.0492	SW
Cl^{36}	β^-	.714	.983					
	E.C	[.0023]	.017	------	.0	.0	.295	SW
K^{42}	$\beta^- \gamma$	3.54	.80					
		1.98	.20	------	------	------	1.42	SW
Ca^{45}	β^-	.256	1	------	------	------	.077	SW
Cr^{51}	E.C. γ	[.0049]	1	<.3	4.4	1.0	.0057	SW
Fe^{55}	E.C	[.0059]	1	------	4.5	1.5	.0060	SHS
Fe^{59}	$\beta^- \gamma$	.462	.53					
		.271	.46					
		1.561	.003	.1	.0	.0	.118	SW
Co^{60}	$\beta^- \gamma$	.314	1.00	.4	.0	.0	.093	SW
Cu^{64}	β^-	.57	.39					
	β^+	.65	.19					
	E.C. γ	[.0075]	.42	.0	2.0	1.6	.130	SHS
Zn^{65}	β^+	.325	.017					
	E.C. γ	[.0082]	.983	.1	5.0	3.0	.0105	SW
Br^{82}	$\beta^- \gamma$	.45	1	.5	.0	.0	.142	SHS
Rb^{86}	$\beta^- \gamma$	1.776	.90					
		.698	.10	------	------	------	.68	SW
Sr^{90}	β^-	.544	1	------	------	------	.205	SW
Y^{90}	β^-	2.25	1.00	.4	.0	.0	.93	SW
$Sr^{90}+Y^{90}$	β^-	.544	.50					
		2.25	.50	------	------	------	.57	SW
I^{131}	$\beta^- \gamma$	.608	.872					
		.335	.093					
		.250	.028					
		.812	.007	6.7	.3	.0	.188	SW
I^{132}	$\beta^- \gamma$	1.53	.24					
		1.16	.23					
		.97	.20					
		2.12	.18					
		.73	.15	------	------	------	.483	SHS&LJB
$Cs^{137}+Ba^{137m}$	$\beta^- \gamma$	.52	.92					
		1.18	.08	62.4	.3	.0	.242	SHS
Au^{198}	$\beta^- \gamma$	0.959	.99					
		.282	.01	16.2	.1	.1	.328	SW
Tl^{204}	β^-	.764	.98					
	E.C	[.071]	.02	------	.1	.1	.234	SHS

TABLE 14.7. Notes

Column 1: Chemical symbol and atomic weight of nuclide.

2: The radiations are shown by the following symbols:

β^-=negatron (negative electron)
β^+=positron (positive electron)
γ=gamma radiation
E.C.=characteristic X-rays from electron capture

The symbol for gamma radiation follows the symbol for the transition with which it is associated.

3: The maximum energy of the β spectrum, and the energy of the K_α X-ray line of the product nucleus in square brackets. Beta components which make a negligible contribution to $\bar{E}_\beta$ have been omitted.

4: The component frequency is given to the number of significant figures to which it is known, except that the notation "1" means that there are no other known components.

5, 6, 7: Conversion and Auger electrons and characteristic X-radiation have been computed, using data from the reference of column 10 and using the theory given in Reference SW.

Since these radiations are absent from pure beta emitters, no entry occurs in these columns for such nuclides. Where no entry is given for a β-γ emitter, no information is available.

7: The energy of the characteristic X-radiation includes K and L radiation for entries down to and including Zn^{65}. For all entries below that (i.e., for greater atomic numbers), only L radiation is included.

8: The value of $\bar{E}_\beta$ has been calculated from the graphs of Reference L, using the data of columns 3 and 4, except for the following nuclides:

Cl^{36}: $\bar{E}_\beta$ computed by graphical integration in Reference SW.

Rb^{86}, Sr^{90}, Y^{90}, and Tl^{204}: Information on the distortion of $\bar{E}_\beta$ by the forbidden shape of the β spectrum has been taken from Reference LJB, page 703.

The tabulated values of $\bar{E}_\beta$ include the energies given in columns 5, 6, and 7. See Reference LHH, page 842 ff.

Experimental values of $\bar{E}_\beta$ are given, for a number of nuclides, in References SW and LJB.

9: References for the data in columns 3 through 7.

References

SW: Slack L. and Way K. *Radiations from Radioactive Atoms in Frequent Use.* U.S.A.E.C. publication, February, 1959. U.S. Government Printing Office, Washington, D.C.

SHS: Strominger D., Hollander J. M. and Seaborg G. T. "Table of Isotopes", *Rev. Mod. Physics* **30**, 585, 1958.

LJB: Loevinger R., Japha E. M., and Brownell G. L. "Discrete Radioisotope Sources", in *Radiation Dosimetry*, edited by Hine G. J. and Brownell G. L. (Academic Press, New York, 1956).

LHH: Loevinger R., Holt, J. G., and Hine G. J. "Internally Administered Radioisotopes", in *Radiation Dosimetry*, edited by Hine G. J. and Brownell G. L. (Academic Press, New York, 1956).

L: Loevinger R. "Average Energies of Allowed Beta Spectra", *Physics in Medicine and Biology* **1**, 330, 1957.

Table 14.7 has been kindly checked and revised by R. Loevinger.

15. Availability of Radioactivity Standards.

15.1. *Radioactivity standards.*

Certain laboratories in several countries undertake to supply radioactivity standards calibrated by the absolute or relative measurements described in 14.2. and 14.4. and also to calibrate standards against their standards. Whereas in the last

ICRU Report it was possible to list such standards in detail, it has now become impracticable on account of the great increase in the number available. For up-to-date information reference should be made to such publications as the Isotope Index.

The laboratories particularly concerned in this work include:

Atomic Energy Research Establishment, Harwell, England (AERE);

Commissariat à l'Énergie Atomique, Centre d'Étude Nucléaires de Saclay (France) (CEA);

Kernreaktor Isotopen Laboratorium, Karlsruhe (W.Germany) (KIL);

National Bureau of Standards, Washington 25, D.C. (NBS);

National Physical Laboratory, Teddington, England (NPL).

Details of standards of short-lived nuclides, and their availability, are obtainable on request from the above laboratories.

The National Bureau of Standards now only maintains national standards of a number of radionuclides as reference standards, based on these, are in most cases now available commercially in the United States of America.

The Atomic Energy Research Establishment issues absolutely measured solution standards of radionuclides if the corresponding standards are not conveniently available from the National Physical Laboratory.

Absolutely measured standards of many nuclides can also be obtained on request from Atomic Energy of Canada Limited, Chalk River, Ontario, Canada; the Atomic Energy Research Establishment, Harwell; the Commissariat à l'Énergie Atomique; the Kernreaktor Isotopen Laboratorium; the National Physical Research Laboratory, Pretoria, South Africa, and the National Research Council, Ottawa, Ontario. For the majority of these nuclides no systematic international comparisons of standards have yet been made.

The Committee considered listing nuclides for which standards were required, but decided that such a list would be of little value in view of possible rapid changes in the situation.

15.2. *Radioactivity standards in solid form.*

The Committee has given some consideration to the question of standard gamma-ray sources in the millicurie region, in the form of wires, cylinders, grains, etc., of Co^{60}, $Cs^{137}+ Ba^{137m}$, Ta^{182}, Ir^{192}, and Au^{198}. Such standards, of known disintegration rate, are generally produced by comparing the gamma-ray emission of a number of solid sources having identical dimensions, dissolving one of the sources and assaying the solution. Facilities for the calibration of Co^{60} sources by comparison with standards which have been set up in this way are available at the AERE, the CEA, the NBS, and the NRC. No steps have been taken to specify the construction and dimensions of standard sources of artificially produced nuclides in solid form.

The Committee is of the opinion that there is no very urgent need at present for standard sources in the multi-curie range. This problem should be considered, however, during the next three years.

15.3. *Maintenance of standards.*

A serious problem in connection with standards of short-lived nuclides is the maintenance of the calibration of measuring equipment between the dates of issue of the standards. For β- and γ-emitting nuclides ionization chambers and Geiger-Müller or proportional counters seem at present to be the most satisfactory instruments.

For such equipment it is essential to use reference sources for checking stability, and a list of radioactive nuclides suitable for this purpose is given in table 15.1. Each of the nuclides listed has a half-life [38] greater than 5 years, and reasonably reliable corrections for decay can be made. It may, however, be desirable to take into account the possibility of error in such corrections for reference sources of the shorter-lived nuclides in table 14.7., depending on the lengths of the periods during which they are used. In this connection also the possibility of radiochemical impurity in the source should be borne in mind.

TABLE 15.1.—*Radionuclides*[a] *for checking instrument stability*

Nuclide	Useful radiation	Percentage decay (approx) in a year
H^3	β (Tritiated plastic foil)	5.4
C^{14}	β (Thick layers)[b]	0.01
Cl^{36}	β	Negligible
K^{40} (natural)	β	Do.
Co^{60}	γ and β	12.4
$Sr^{90}+Y^{90}$	β and bremsstrahlung	2.4
Tc^{99}	β	Negligible
$Cs^{137}+Ba^{137m}$	β and γ	2.5
Ra^{226} (+decay products)	γ and Compton recoils	0.04
$Pb^{210}+Bi^{210}$ (Ra(D+E))	β	3
$U^{238}+U^{235}+UX+UY$	α (thick layers) and β	Negligible
Pu^{239}	α (thin layers)	Do.

[a] Reference sources of some of these nuclides are available commercially.

[b] Exchange of C^{14} with atmospheric carbon may occur if the source is not properly constructed.

15.4. *Radiochemical standards.*

A need has arisen to have available radioactive samples which can be used to check the yield of chemical procedures. Such problems arise particularly in investigations of radioactivity of biological materials and the sources from which they derive.

A first step in this direction has been taken by the United Nations Scientific Committee on the Effects of Atomic Radiation, which has distributed $Sr^{90}+Y^{90}$ samples of the following composition:

(i) milk ash;

(ii) animal bone ash;

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[38] The Committee has not been able to examine critically the literature on determinations of half-lives and has therefore omitted half-life values from the tables in this report. The approximate decay in a year, expressed as a percentage of the activity, is given in the final column of table 15.1.

(iii) artificial human bone preparation: A preparation of animal bone diluted with non-active calcium phosphate so as to have an activity comparable to that of human bone (about ten times lower than animal);

(iv) calcium phosphate (control sample);

(v) vegetation ash;

(vi) aqueous solution of mixed fission products;

(vii) soil;

(viii) $Sr^{90}+Y^{90}$ standard solution.

Committee I considers that attention should be given to the question of whether appropriate corresponding samples of Sr^{89}, $Cs^{137}+Ba^{137m}$, Pu^{239} and possibly of other nuclides should be made available.

16. Techniques for Measuring Radioactivity in Hospitals and Biological Laboratories

Techniques and instruments for the measurement of radioactivity for the diagnosis and treatment of human disease and the investigation of biological processes are comparatively new and the field is still developing quite rapidly. Some of the instruments and techniques have now been in use for sufficiently long periods, however, to permit some evaluation of their respective advantages and disadvantages for particular purposes. Since the choice of an instrument or technique for each purpose is so often a compromise between sensitivity and reliability, and between accuracy and convenience, any recommendation as to the best type of instrument or technique to be used must necessarily be arbitrary. The Committee has endeavored to consider the various clinical and biological techniques in use at the present time and to suggest preferred methods of measurement for each of these, based on the experience of various laboratories in a number of different countries.

The techniques of absolute measurement are usually of little value in the practical application of radionuclides to medical and biological problems. Furthermore, standards of most of the nuclides of interest are available (section 15). The choice of technique or instrument must therefore be concerned chiefly with the sensitivity, reliability, convenience and accuracy necessary for the desired purpose.

The various technical procedures to be considered divide themselves into three main classes:

I. Measurement of samples before administration to patients or biological specimens;

II. Measurement of activity in the patient or animal, both quantity and distribution;

III. Measurement of samples taken from the patient or biological specimen.

In general, the amounts of radioactivity involved in each sample are greatest with I, less with II and least with III. Consequently, since the highest sensitivity is not required, the most reliable instrument, the ionization chamber, can often be used for I but is impractical for II and III. Since the amount of material necessary for a clinical procedure will be determined by the sensitivity of the equipment used in classes II and III, the highest sensitivity consistent with reliability should be the objective in order to keep the radiation hazard to a minimum. In the case of γ radiation this almost invariably necessitates a scintillation counter, preferably with a high sensitivity inorganic phosphor such as sodium iodide. In the case of β radiation, the Geiger-Müller counter is usually at least the equal of the scintillation counter in sensitivity and usually is superior in reliability. The scintillation counter is frequently less stable and requires, therefore, much more careful testing than a Geiger-Müller counter system. However, provided adequate care is taken, the sensitivity is sufficient to justify the use of the instrument in most circumstances in classes II and III.

In addition to these three categories there are additional ancillary measurements that may have to be made, such as the checking of samples for disposal and laboratory monitoring, which have been designated as class IV. In general, rather simple systems of comparatively low accuracy and usually not of the highest sensitivity are adequate for this purpose.

The details of the type of instrument preferred for a particular procedure are contained in table 16.1. The sensitivities quoted are very approximate figures for the systems in common use.

It is evident that reliability is a very important consideration in most clinical and biological procedures with radionuclides. The reliability of equipment can best be tested over long periods by the use of reference standards (see section 15.2.). A standard of the nuclide whose activity is to be measured must be used to calibrate the equipment, but, since most of these are short-lived, reference sources of long-lived nuclides (table 15.1.) may subsequently be used in their stead. It is good practice to use such a source to check the performance of the equipment before every series of measurements and at least once daily. It must be remembered, however, that such standards do not check all aspects of the behavior of the measuring system for other nuclides. Scintillation counters, for example, may require occasional additional testing with a pulse-height analyser (if not already equipped with such a device) to ascertain that the performance has not altered.

Methods for measuring samples to be used in therapy, in such a manner that the dose rate from the surface or at a distance is obtained rather than the activity, are included in class I. Such measurements require special measuring techniques. These usually involve ionization-chamber systems equipped with suitably calibrated current-measuring devices for the measurement of both β and γ rays. Such techniques are not often used for the measurement of samples in class II or III.

TABLE 16.1.—*Preferred methods of measurement of radionuclides for clinical and biological use*

Class	Purpose	Radiation	Type	Approximate sensitivity	Remarks
I	Calibration of therapy samples or bulk tracer shipments.	γ, bremsstrahlung and β	Calibrated well (re-entrant "$4\pi\gamma$") ionization chamber.	1 μg radium up	One standard design available is that of NPL (calibrated).
	Checking of aliquots for injection.	γ and bremsstrahlung	Simple fixed geometry Geiger-Müller or scintillation system.	0.1 μc up	If previous measurements are made by supplier, accuracy required is not high.
	Measurement of discrete source output.	γ	Ionization chamber and fixed geometry.	Usually of order of 1 mc up.	Various systems exist. Necessary for therapeutic situations where dose rate rather than activity is required.
		β	Extrapolation ion chamber	do	
II	"In vivo" counting Quantitative.	γ	Fixed geometry scintillation phosphor (inorganic).	at 25 cm 1 μc I^{131}	Large phosphor may increase sensitivity further.
	Distribution	γ	"Scintiscanner" (inorganic phosphor and mechanical scan).	5μc I^{131} up	Do.
		β	End window or needle Geiger-Müller system.		
III	Excreta samples (total volume).	γ and bremsstrahlung	Scintillation phosphor (inorganic).	5 mμc Co^{60} up	
			Ring of large Geiger tubes	50 mμc Co^{60} up	
	Blood and other small samples.	γ and bremsstrahlung	Well scintillation counter	0.05 mμc Co^{60}	Sensitivity to I^{131} and Co^{60} very similar.
		β	Cylindrical jacketed counter	0.05 mμc/ml P^{32}	Sensitivity depends on β energy.
		Low energy β	Gas flow Geiger-Müller counter.	0.2 mμc C^{14}	
	Chromatographic and autoradiographic samples, etc.	β	End window Geiger-Müller system, or gas flow Geiger-Müller system.	0.2 mμc/cm^2 C^{14}	Sensitivity varies according to nuclide studied.
IV	Routine laboratory monitoring	β and γ	Geiger-Müller counter	0.05 mμc	
			do	2 mμc	
	Disposal	γ and bremsstrahlung	Simple fixed geometry Geiger-Müller or scintillation system. or calibrated ion chamber	0.1 μc to 1 mc 1 mc up	

The techniques referred to in table 16.1. are not necessarily the only ones which may be used for the purposes mentioned. However, any others that are likely to be employed will in general have similar characteristics.

17. Comparisons of Radium Standards

A great many careful and precise relative measurements on national radium standards have been made in recent years, and the results of these measurements have also been subjected to careful statistical analysis. In order that it may be more generally available Committee I has been requested to collect and collate the information relating to these measurements.

17.1. *Comparisons of 1934 Hönigschmid radium standards.*

In table 17.1. is given as comprehensive a list as possible of comparative measurements that have been carried out in recent years using various pairs of the radium standards that had been prepared by Hönigschmid in 1934. As there has been some difficulty in collecting all the individual laboratory estimates of errors these have not been included. A 95 percent confidence limit has, however, ultimately been derived from the variations in the measured values shown in table 17.1.

The standards have been designated as follows:

"A" U.S. National Radium Standard No. 5437 (38.23 mg of radium element as of June 2, 1934)

"B" British National Radium Standard No. 5432 (15.60 mg of radium element as of June 2, 1934)

"C" Canadian National Radium Standard No. 5425 (24.15 mg of radium element as of June 2, 1934)

"D" U.S. National Radium Standard No. 5440 (20.45 mg of radium element as of June 2, 1934)

"G" German National Radium Standard No. 5426 (14.61 mg of radium element as of June 2, 1934)

"P" Paris International Radium Standard No. 5430 (16.92 mg of radium element as of June 2, 1934)

"V" Vienna International Radium Standard No. 5428 (23.41 mg of radium element as of June 2, 1934)

The ratios given in table 17.1. have been obtained for pairs of Hönigschmid standards at the designated laboratories during the past five years.

Since the results in table 17.1. were listed, two further microcalorimetric values were obtained at the National Bureau of Standards for A/D of 1.871 and 1.872. These would not, however, influence the final result for A/D and were not included.

These results have been analyzed for consistency by Dr. W. S. Connor, Jr., and Dr. W. J. Youden of the National Bureau of Standards Statistical Engineering Section. In the first instance a least-squares adjustment was made using only the experimentally determined ratios and ignoring Hönigschmid's ratios. By normalizing so that the product ABCDGPV was equal to unity the following numerical values for the radium standards A,B,C,D,G,P, and V were obtained, with cor-

TABLE 17.1.—*Summary of results of measurements on Hönigschmid standards*

Place	Method	Standards	Ratios	Hönigschmid ratios
NBS	Electroscope	A/B	2. 441	2. 450
NBS	Calorimeter	A/B	2. 450	2. 450
NBS	Electroscope	A/D	1. 870	1. 870
NBS	Calorimeter	A/D	1. 873	1. 870
NBS	Electroscope	A/D	1. 870	1. 870
NBS	Calorimeter	A/D	1. 869	1. 870
NBS	Electroscope	A/D	1. 870	1. 870
NBS	Calorimeter	A/D	1. 875	1. 870
NBS	Calorimeter	A/D	1. 874	1. 870
NBS	Electroscope	D/B	1. 305	1. 310
NBS	Calorimeter	D/B	1. 308	1. 310
NBS	GM counter	D/B	1. 306	1. 310
NPL	Electrometer	D/B	1. 304	1. 310
NPL	Electroscope	D/B	1. 306	1. 310
NPL	GM counter	D/B	1. 304	1. 310
NBS	Electroscope	A/G	2. 608	2. 617
NBS	Calorimeter	A/G	2. 612	2. 617
NBS	Electroscope	D/G	1. 395	1. 400
NBS	Calorimeter	D/G	1. 398	1. 400
NBS	Electroscope	A/C	1. 578	1. 583
NBS	Calorimeter	A/C	1. 583	1. 583
NBS	Electroscope	C/D	1. 185	1. 181
NBS	Calorimeter	C/D	1. 184	1. 181
NRC	Ionization chamber	C/D	1. 185$_5$	1. 181
NRC	Ionization chamber	C/G	1. 652$_7$	1. 653$_5$
PTB	Electroscope	C/G	1. 651$_6$	1. 653$_5$
Institut für Radiumforschung.	Electroscope	C/V	1. 030$_6$	1. 031$_9$
Union Minière	Radon ionization chamber.	C/P	1. 425$_7$	1. 427$_4$
NPL	Electroscope	C/B	1. 546	1. 547$_8$
NPL	Electrometer	C/B	1. 545	1. 547$_8$
NPL	Scintillation counter	C/B	1. 547	1. 547$_8$

responding 95 percent confidence limits:

A: 1. 832 ±0. 002
B: 0. 7496± . 001
C: 1. 159 ± . 0005
D: 0. 9788± . 0004
G: . 7015± . 0006
P: . 8132± . 002
V: 1. 125 ± . 003

From these values, obtained by "least-squares" treatment of the experimental data, were derived the experimental ratios given in table 17.2, which we may compare with those of Hönigschmid.

Hönigschmid's estimate of his own weighings was that they were accurate to about 0.02 mg per weighing, or to 0.1 percent on 20.00 mg. His values are therefore as precise as those reported from the various standardizing laboratories. Dr. Connor therefore made another analysis of the results incorporating those of Hönigschmid, however, after the manner of Connor and Youden (Journal of Research NBS **53**, 273, 1954) giving equal weight to both the experimental and Hönigschmid ratios (Connor and Youden case of k=1 ibid.) and obtained the following "best estimates" for each standard:

Standard	Best estimate in milligrams	Hönigschmid mass of radium (June, 1934) in milligrams	Difference in milligrams
A	38. 208	38. 227	−0. 019
B	15. 615	15. 605 (15. 604$_6$)	. 010
C	24. 159	24. 153	. 006
D	20. 425	20. 447	−. 022
G	14. 614	14. 607	. 007
P	16. 935	16. 921	. 014
V	23. 428	23. 407	. 021

TABLE 17.2.—*Comparison of weighed and measured values*

Standards	Experimental ratios	Hönigschmid ratios	Percentage difference
A/B	2. 444	2. 450	0. 24
A/D	1. 872	1. 870	. 11
D/B	1. 306	1. 310	. 31
A/G	2. 612	2. 617	. 19
D/G	1. 395	1. 400	. 36
A/C	1. 581	1. 583	. 13
C/D	1. 184	1. 181	. 25
C/G	1. 652	1. 654	. 12
C/V	1. 030	1. 032	. 19
C/P	1. 425	1. 427	. 14
C/B	1. 546	1. 548	. 13

TABLE 17.3.—*Comparisons of 1934 Hönigschmid radium standards with earlier standards*

Standards compared	Activity ratio Weight ratio
1934 Hönigschmid Paris (5430) 1911 Paris	1. 0014
1934 Hönigschmid Vienna (5428) 1911 Paris	1. 002
1934 Hönigschmid USA (5437) 1911 Paris	0. 9974
1934 Hönigschmid USA (5437) 1911 Vienna	. 9984
1934 Hönigschmid USA (5440) 1911 Paris	. 9966
1934 Hönigschmid USA (5440) 1911 Vienna	. 9975
1934 Hönigschmid Belgium (5431) 1911 Paris	1. 0012
1934 Hönigschmid United Kingdom (5432) 1912 United Kingdom (secondary standard)	*1. 003
1934 Hönigschmid Germany (5426) 1912 Germany (secondary standard)	*1. 0025
1934 Hönigschmid Canada (5425) 1932 Canada (secondary standard)	0. 9994
1934 Hönigschmid Canada (5425) 1911 Vienna	1. 004
1934 Hönigschmid Canada (5425) 1911 Paris	1. 001
Mean (omitting*)	0. 9999

TABLE 17.4.—*Comparisons of "secondary" international radium standards with the 1911 Paris and Vienna standards*

	Standard	Vienna measurement mg $RaCl_2$	Paris measurement mg $RaCl_2$	Difference percent
I	Paris (for Mme Curie) 1912	22. 47	22. 42	+0. 22
II	Germany (1912)	19. 73	19. 74	−0. 05
III	England (1912)	21. 10	21. 16	−0. 28
IV	USA (1913)	20. 29	20. 28	+0. 05
V	Sweden (1913)	9. 74	9. 71	+0. 31
VI	Japan (1913)	9. 80	9. 80	0. 00
VII	Portugal (1913)	9. 07	9. 11	−0. 44
VIII	Belgium (1924)	28. 46	28. 40	+0. 21
IX	Czechoslovakia (1924)	33. 09	33. 03	+0. 18
X	Leningrad (1927)	39. 10	39. 06	+0. 11
XI	Leningrad (1927)	18. 98	19. 00	+0. 13
XII	Australia (1928)	26. 78	26. 75	+0. 11
XIII	Canada (1932)	31. 90	31. 81	+0. 28

References: (a) Report No. 452a of the Institute für Radiumforschung "On the radium-standard-preparations" by Stefan Meyer; Anzeiger der Akademie der Wissenschaften in Wien. Vol. 82 Nos. 8 and 9, pp. 25–30 (1945).
(b) The Work of the International Radium Standards Commission. Berta Karlik. Report to the Joint Commission on Radioactivity, Stockholm meeting 1953.

In no case are the observed differences greater than Hönigschmid's stated accuracy of 0.02 mg per weighing. In view of the many precise subsequent intercomparisons that have been carried out, however, it may seem reasonable to adopt the "best estimates" given above, although for all practical purposes adherence to Hönigschmid's original weights, in "Hönigschmid milligrams",[39] is probably more than adequate.

17.2. *Earlier comparisons of radium standards.* It seems desirable to refer to the comparisons which have been made between the 1934 Hönigschmid standards and earlier standards, in particular comparisons between the 1934 Hönigschmid standards and the 1911 Paris and the 1911 Vienna standards. These comparisons provide the link between the Hönigschmid standards and the 1911 Paris and 1911 Vienna standards, and thus relate existing "secondary" international standards to the 1934 Hönigschmid standards.

The "secondary" international standards in question were issued by the International Radium Standards Commission; their values are based solely on comparisons by means of ionization chambers with the 1911 Paris and 1911 Vienna standards. It seems that radium measurements in Sweden, Japan, Portugal, Czechoslovakia, and Australia may still be based on such "secondary" international standards.

Table 17.3. gives the results of comparisons by means of ionization chambers of the 1934 Hönigschmid standards with earlier standards, and table 17.4. gives the results of comparisons of "secondary" international standards with the 1911 Paris and 1911 Vienna standards.

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[39] Although Professor Hönigschmid calibrated his weights for internal consistency he did not compare them with the standard kilogram. It is doubtful, however, whether the 100-g weight in his set, which is the weight with which the other would probably be compared, would have been very greatly in error.

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Appendix II. Report of ICRU Subcommittee IV.1 on Focal Spots

The task of this Subcommittee has been considerably simplified by the instruction given to it in the report of ICRU IV dated August 14, 1958, as follows:

"Only the tools for making an accurate measurement under any desired operating conditions should be specified. The specification of tolerances allowable from nominal focal sizes, of the tube current and voltage conditions should be considered to be outside the scope of ICRU IV."

Consistent with this instruction, the Subcommittee has given special attention to methods of measurement of focal spot size in X-ray tubes for diagnosis. In this report we submit our recommendations for a measurement method and procedure suitable for such tubes, particularly of the line focus variety, when operated at voltages not exceeding 150 kvp.

At the ICRU meeting held in Geneva, September 1958, it was suggested that Subcommittee IV.1 should also include in its task the method of measurement of uniformity of focal spots and that the studies should not be limited to diagnostic tubes but should include therapy tubes and possibly radioactive sources. The Subcommittee has not as yet been able to evaluate this suggestion. It is of the opinion, however, that a method applicable specifically for diagnostic tubes can and should be recommended at this time, and that therapy tubes and radioactive sources should be the subject of further deliberation.

The following recommendations represent a composite of the positions taken by the individual members of the Subcommittee with respect to the various points involved rather than the unanimous opinion of all. However, a majority are in agreement that the method proposed will give a substantially accurate measure of the effective focal size of an X-ray tube under the operating conditions obtained at the time the measurement is made.

Recommendations

Method of Focal Spot Image Formation and Measurement (for diagnostic tubes at voltages up to 150 kvp)

1. *Basic method*
   Pinhole radiograph of focal spot, produced by pinhole accurately alined with respect to "central ray" of X-ray beam.
2. *Details of pinhole camera*
   a. Pinhole diameter:
      0.030 mm for focal sizes below 1.0 mm
      .075 mm for focal sizes 1.0 mm to 2.5 mm
      .100 mm for focal sizes above 2.5 mm
   b. Diaphragm thickness, material, and pinhole profile: The pinhole diaphragm shall be manufactured from a 90/10 gold-platinum alloy, 1.5 mm thick. The cylindrical part of the pinhold shall be five times as long as its diameter and followed by a conical part with a divergent angle of 8° (see fig. II.1 below).
   c. Focal spot to pinhole distance: 10 cm minimum (where focus location is not precisely known or directly measurable, precise location stereoscopically with dual pinholes is recommended.)

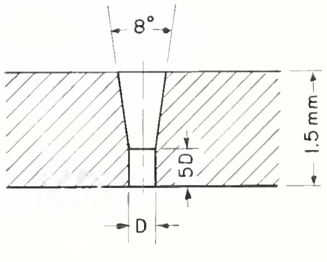

FIGURE II.1. *Cross-sectional view of pinhole diaphragm.*

d. Enlargement factor:
   2.0 minimum, for focal sizes up to 2.5 mm
   1.0 minimum, for focal sizes above 2.5 mm

3. *Photographic technique*
   a. Type of film:
      Any commonly used dental film (single coating)
   b. Image density:
      0.6 to 1.0, as measured in the most dense portion of the image. Density to be controlled by means of standard developing technique and proper exposure time at tube current and voltage factors employed. (This element involves a point of special difficulty, because of variations in structure of respective focal spots. If lines in focal spot are superimposed, greater maximum density results than in cases where lines are not superimposed, for equal exposure values. Hence a wide latitude in allowable maximum density becomes necessary.)
4. *Measurement of image*
   a. Lighting:
      Back-lighted at approximately 20 foot-candles.
   b. Measurement procedure:
      Use scaled magnifier with 0.1 mm scale divisions, 5 to 10 times magnification. Measure length and width, respectively, including all perceptible portions of the image, to nearest 0.1 mm. For irregularly shaped (nonrectangular) focal spots, measurement should be taken of all significant dimensions.
   c. Correction of image measurements:
      All image dimensions shall be divided by the enlargement factor, accurately determined (see 2d).
5. *Statement of dimensions*
   a. Width of focal spot shall be stated as the width of the image divided by the enlargement factor.
   b. Length of focal spot shall be stated as the length of the image divided by the enlargement factor, further corrected by a multiplier of 0.7.[41]
   c. For nonrectangular focal spots, all significant dimensions, including maximum diameter, shall be stated as indicated by actual measurements divided by the enlargement factor.[42]

Appendix II.A

The proposal of a fractional multiplier for correction of the measured image length arises from the long-recognized fact that the lengthwise distribution of energy in the focal spot of a line-focus tube tends to be peaked at the center and diminishes gradually to zero at the ends. Hence, the effective length, based on its effect on either radiographic definition or loading capacity, cannot be stated as equal to the measured image length as initially corrected. Some further correction is necessary. This point has been exposed, and to some extent explored in the papers heretofore presented by Polansky and O'Connor (1954), Rogers (1956), Fenner and Jochim (1957), and Servais (1958). However, we do not feel that work done to date has been sufficiently comprehensive to establish a suitable correction factor, and the suggestion above of a multiplier of 0.7 is given primarily by way of example or as an approximate value rather than as a final recommendation. We would like to recommend that ICRU IV enlist the good offices of a suitable research organization to explore this matter quantitatively over the entire range of diagnostic tubes commercially produced, with the objective of establishing a proper formula for focal spot length.

Appendix II.B. Off-Focus Radiation

It must be recognized that the effect of any off-focus radiation on definition is not taken into account by these measurements, and unless off-focus radiation is suppressed, the definition to be expected from the tube may be influenced by it.

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Appendix III. Estimation of Integral Absorbed Dose

Though integral absorbed dose may not be regarded as of great importance in clinical radiotherapy, there are occasions when a knowledge of it may be of considerable value. Therefore it is useful to have a simple and generally accepted method for its determination. Of the methods described in the radiological literature (Happey, 1940; Mayneord, 1940; Grimmett, 1942; Meredith and Neary, 1944; Mayneord and Clarkson, 1944; Boag, 1945), those based on calculation seem to be of the most general and the most straightforward application. Furthermore, much of the information required for the calculations may be tabulated as standard constants.

As the report has outlined, various integral dose values may be considered but the most informative appear to be Σ_B, the integral absorbed dose within the confines of the beam, and

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[41] See appendix II.A for explanation.

[42] See appendix II.B for explanation.

Σ_R, the integral absorbed dose to the rest of the patient.

Meredith and Neary (1944) give formulas for the calculation of integral absorbed dose (per 100 r "in air") under a variety of conditions and one of these formulae can be used for the determination of both Σ_B and Σ_R. For a medium irradiated by a field of surface radius "a", the formula enables one to calculate the integral absorbed dose within a coaxial, confocal cone of surface radius "ka". If $k=1$, the formula gives Σ_B directly. The evaluation of $\Sigma_R(=\Sigma_K-\Sigma_B)$ is more difficult because the variation in size and shape of the body makes it difficult to decide on the value of k which should be chosen.

An extreme would be to calculate Σ_K for $k=\infty$ which would give an upper limit of Σ_R though a rather unrealistic value since a patient can seldom be regarded as an infinite phantom even for the smallest field. A better approach would be to formalize the patient as being a cylinder of 20-cm radius coaxial with the beam axis for all trunk treatments and possibly of 10-cm radius for all head treatments. Σ_K could then be calculated out to this distance. The value of k would, of course, vary for each field size but this should not introduce any difficulty in calculation or tabulation.

The general formula for $k \geq 1$ is

$$\Sigma_K \text{ to a depth } z'=A\int_0^{z'}\left(\frac{d+z}{d}\right)^2 \{P+M[1-2kI_1(\lambda^*a)K_1(k\lambda^*a)]\}dz. \quad \text{(III.1)}$$

When $k=1$ $\Sigma_{K=1}$ is the same as Σ_B

$$\Sigma_B=A\int_0^{z'}\left(\frac{d+z}{d}\right)^2 \{P+M[1-2I_1(\lambda^*a)K_1(\lambda^*a)]\}dz. \quad \text{(III.2)}$$

If $k=\infty$, i.e., we consider an infinite phantom,

$$\Sigma_\infty=A\int_0^{z'}\left(\frac{d+z}{d}\right)^2 (P+M)\,dz. \quad \text{(III.3)}$$

In these formulas the symbols used stand for the following:

A = Field surface area in sq cm
a = Field surface radius in cm
d = Focus skin distance in cm
z = Depth in cm
z' = Depth of phantom
P = Primary percentage depth dose at depth z
M and λ = Meredith-Neary constants for the radiation conditions being used.

$$\lambda^*=\lambda\,\frac{d+z}{d}$$

$I_1(x)$ and $K_1(x)$ are respectively the modified Bessel functions of the first and second kinds of order one.

It will be noted that each of the above formulas may be expressed in the form

$$\Sigma_K=A\Phi \quad \text{(III.4)}$$

where Φ is the integral of eq (III.1).

Thus Φ may be tabulated for different sets of conditions and the evaluation of the integral absorbed dose for any treatment reduces to the multiplication of the surface area by the value of Φ appropriately selected.

An example of such a table is given below (table III.1), together with an example of its use. In this connection, it must be stressed that the quoted factors allow the calculation of the integral absorbed dose per 100 r delivered "in air." To calculate the values for 100 r "with backscatter," the above calculated values of I_K must be multiplied by $(100/100+S)$, where S is the percentage surface backscatter for the field in question.

Example. To find the values of Σ_B and Σ_∞ to 15 cm deep for a 10-cm diameter (80 cm²) field delivering 4000 r with backscatter.

(a) *To evaluate Σ_B and Σ_∞ per 100* **r** *"in air."* From the above table, by interpolation where necessary

for $k=1$ $\Phi=1068$
and for $k=\infty$ $\Phi=1800$

Hence $\Sigma_B=80\times1068 = \underline{85{,}440}$ and $\Sigma_\infty=80\times1800 = \underline{144{,}000}$

(b) *To allow for backscatter.*
Now the percentage backscatter for this field is approximately 25.

$\therefore \Sigma_B$ and Σ_∞ per 100 r with backscatter will be:

$$\Sigma_B=85{,}440\times\frac{100}{100+25} \qquad \Sigma_\infty=144{,}000\times\frac{100}{100+25}$$

$$=\underline{68{,}400} \qquad =\underline{115{,}000}$$

Table III.1.—*Typical table of Φ*

2.0 mm Cu HVL. 50 cm F.S.D.

z'	$k=1$							$k=\infty$
	Field area in sq cm							All areas
	20	50	100	150	200	300	400	
10	680	800	890	940	980	1030	1070	1300
15	810	990	1120	1200	1250	1330	1390	1800
20	890	1090	1260	1350	1420	1530	1610	2200

For 4000 r with backscatter

$\Sigma_B=2.74$ megagram rads

and $\Sigma_\infty=4.6$ megagram rads.

Hence an estimate of $\Sigma_R = \Sigma_\infty - \Sigma_B = 1.86$ *Mg rads.*

The formulas presented can only be applied, of course, where values of M and λ exist; i.e., to those radiation conditions to which the Meredith-Neary formula has been found to fit. For some published data this basic formula does not provide good agreement and is inappropriate to the calculation of central depth dose data and isodose curves. The accuracy, however, required for integral absorbed dose calculations is considerably less than for percentage depth dose data, and therefore constants which give only a moderate agreement between experimental and calculated data could be used satisfactorily in this work.

A test for the accuracy of the method being proposed is not easy to apply since there is comparatively little comparable information in the literature. However, Mayneord (1940), using the equivalent wavelength, calculated that the incident flux per 1 r in air and per sq cm of field surface when an infinite phantom is irradiated by a 10-cm diameter field at 1.5 mm Cu HVL would be 3200 ergs. The Meredith-Neary calculation of the total energy absorbed for such circumstances gives an answer of 3290 ergs. This agreement to within 3 percent must be regarded as most satisfactory, especially when one remembers that Mayneord's computational method for similar circumstances only accounts for about 50 percent of the incident energy.

Though the formulas specifically deal with the integral doses for treatments with circular fields, they may easily be applied to rectangular fields by the use in the formulas of the radius and area of the "equivalent circles," as evaluated by Jones (1949) and by Day (1950).

In conclusion, therefore, it may be claimed that the method which is most suited for general acceptance as the method for integral absorbed dose determinations is that of Meredith and Neary, because it is probably the most flexible yet described. Further, the necessary constants can readily be tabulated and with them the computations are extremely simple.

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Appendix IV. ICRU Position Plan

(Adopted 14 September 1958, Geneva, Switzerland)

I. The need: Continued study and development of definitions and basic concepts in the general field of units and measurements of ionizing radiation, with special emphasis on radiological applications.

(Radiation is defined as gamma-rays and X-rays, alpha and beta particles, high-speed electrons, neutrons, protons, and other nuclear particles; but not sound or radio waves or visible, infrared, or ultraviolet.),

The ICRU began work in this field in 1925 and has continued its organization and the study of the problems continuously since then with the exception of the war years. (Some of the members of the original Commission are still active.) While the ICRU initially had its roots mainly in the radiological profession, it has since 1950 included representation from most other areas of science where there are requirements for radiation units, standards, and measurements.

It is proposed that the ICRU continue to operate as it has in the past, maintaining its responsibilities in the development of radiation units, standards, and measurements.

II. The need: At some international focal point, to keep informed on and evaluate the measurement standards and the intercomparison of such standards required in the radiation field.

Beginning in 1927, the ICRU developed preliminary ideas relative to X-ray standards and through its encouragement brought about an intercomparison (Behnken) of the then existing standards at the national laboratories of Germany, the United States, and the United Kingdom. Because of the uncertain nature of the preliminary intercomparisons, further intercomparisons were made in 1931 (Taylor), as a result of which the ICRU agreed on the basic principles of X-ray standardization. As a further result of the Commission's recommendations, agreements were reached between the national laboratories of the United States, United Kingdom, and Germany on the basic requirements for X-ray standards. Intercomparisons between X-ray standards have continued from time to time since then.

Again upon the recommendation of the Commission, new intercomparisons beginning in 1953 were carried out between Sweden, Germany, United Kingdom, Holland, France, and the United States (Allisy, Aston, Jaeger, Oosterkamp, Thoraeus, Wyckoff). The results of these intercomparisons have been evaluated by the ICRU, and the overall agreement is generally within ±0.5 percent in the energy range of about 60 to 250 kv.

Similar intercomparisons have been carried out for the standards of radium involving the following countries: United States, United Kingdom, France. Canada, Austria, and Germany. For the radium standards best values are to be discussed at the September 1958 meeting of the ICRU.

As a result of discussions at the Tripartite Conferences beginning in 1947 between the United States, United Kingdom, and Canada, at which the ICRU was represented, a program was agreed upon for the interchange and intercomparison of calibrated sources of artificial radionuclides. This program has been carried forward with the co-operation of the national laboratories of these and other countries and the results co-ordinated through the ICRU.

While the ICRU itself has not carried out the specific programs outlined above, it is largely due to its influence that the programs have been carried out and the results suitably correlated.

It is proposed that the ICRU continue its central co-ordinating activity in this general field, at least until such time as some other suitable international scientific organization is in a position to take over and support the activities. It is believed that the International Bureau of Weights and Measures (BIPM) is the type of permanent standards organization that, having no regulatory authority, should undertake these responsibilities. It could either perform the services itself or appoint a committee to review the intercomparisons and evaluate the results.

It is recognized that at the present time the BIPM lacks authority and facilities to carry out this work. On the other hand, it is the type of permanent organization of an intergovernmental nature based on treaty that should handle such standards problems along with other standards. It is understood that the BIPM is actively investigating the possibility of performing this service and obtaining the necessary changes in its treaty authority to perform such work.

Until the BIPM is in a position to promote a radiation standards program, the ICRU should continue to perform the correlating service that it has in the past, and collaborate with the BIPM in providing them with the results of its studies and evaluations for their future use.

It is not possible at present for the ICRU itself to be in a position to develop or maintain the necessary physical facilities or standards. On the other hand through the close relationship of the ICRU with the various national laboratories, it can appropriately continue to correlate and co-ordinate the standards programs between the countries involved.

During the interim period prior to the establishment of a radiation standards program by the BIPM, the ICRU should make itself available, on a formal or informal basis, to serve in an advisory capacity to the BIPM on matters of radiation standards.

III. The need: To study and make recommendations relative to the practical measurements of radiation.

While properly equipped national laboratories can establish and reach agreement on radiation standards, there are frequently substantial gaps between the existence of the standards and the ability to properly perform radiation measurements under the widely variable conditions encountered in medicine, biology, industry, and research generally. For example, there is need to continue a study on the proper measurement of source output for X-rays, gamma-rays, electrons, and neutrons. The ICRU has had these problems under study and consideration for many years, with the result that there is relatively good international agreement on the techniques and procedures of measurement. The complete specification of radiation dose in animate or inanimate material presents new difficulties as new radiation applications are developed. The ICRU has kept abreast of most of these and has, when necessary, introduced new units, concepts, and procedures as required.

It is proposed, because of the broad base of interest of the members of the ICRU, that this activity continue.

IV. The need: To continue the development of the principles and practices of radiation dosimetry.

This program includes the development of the principles and practices of the measurement of absorbed dose, exposure dose, RBE dose, etc. Because some of these quantities cannot be measured directly, it is necessary to employ supplemental data and information for the interpretation of the physical measurements. Since its introduction of the absorbed dose concept in 1953, the ICRU has been actively engaged in developing the data necessarily associated with radiation dosimetry. In addition to evaluating data contained in existing literature, the ICRU has encouraged the development of programs to produce new data that may be needed and has itself carried out extensive computations and evaluations to make these data useful in dosimetry practice. (See 1956 report of the ICRU.)

It is proposed that the ICRU continue its activities in this field and that it keep under continuous review all existing or new information relative to auxiliary data required and associated with dosimetry.

V. The need: Primarily for medical radiology, to establish some guidance and international agreement on the methods of testing the basic characteristics of certain essential component radiological equipment.

Numerous factors involving radiological equipment components contribute importantly to the end results of a clinical procedure. This is particularly so where diagnostic procedures may be importantly influenced by factors such as focal spot sizes, grids, intensifying screens, or photographic emulsions. Agreement is needed on the ways to describe and test basic characteristics of such items and a committee was established in 1956 to study the problem.

The aim is not to develop specifications of a performance or other type, nor to develop standards. On the other hand, the recommendations of the Committee may be used later as a basis for such developments by other suitable groups such as the International Standardization Organization.

At present, there is no international group other than the ICRU that deals with the equipment aspects of medical radiology. The continuance of this program, which has been on an exploratory basis for the past two years, will depend upon a review of the situation in 1959.

VI. The need: A collaborative and working relationship at some central focal point between various international organizations having special needs or interests in the field of radiation units, standards, and measurements.

Since its formation, the ICRU has served in this capacity in what is believed to be a very valuable way. Because of its independent nature, it has adequate freedom from governmental influence, but because of its technical competence its opinion would be sought by the governmental groups. By remaining an independent organization, the ICRU is able to devote its total effort to scientific matters in such a way that its scientific judgment and recommendations are accepted on the basis of merit rather than on national or international policy. Because of its desire to maintain this high level of scientific competence, the individuals drawn into the ICRU are selected primarily on the basis of technical ability and achievement without the influence of national or organizational pressures.

Recognizing that various other international organizations have certain specific interests or needs in the field of radiation units, standards, or measurements, the ICRU should collaborate with any and all such organizations to the mutual benefit of all concerned. The ICRU should invite other organizations to present problems for consideration by the Commission; it should evaluate the technical aspects of the problems and the manpower available in the Commission; it should determine that proposed requirements are not already covered by past recommendations of the ICRU; and it should, in collaboration with the other organizations, reach a decision as to how the work should be best carried forward. In some instances this would result in the Commission undertaking the work; in other cases the Commission would collaborate with some other organization to undertake a specific program.

In the recent rapid development of the uses of radiation, numerous new committees or commissions have been established and some older ones have been revitalized. Where these groups have been concerned with problems of radiation units, standards, and measurements, they have almost without exception called upon Commission members as individuals to work on their committees. From the point of view of cross-fertilization of ideas, this has its desirable aspects. On the other hand, the number of experts in the field of radiation units, standards, and measurements is somewhat limited with the result that some individuals find themselves grossly over-burdened by activities of different international committees reworking (usually in relatively minor ways) material already developed by the ICRU. This has been wasteful of manpower and has resulted in the development of a complex of basically similar recommendations being promulgated by a variety of organizations.

To avoid these difficulties it is proposed that where other international committees have particular needs, they work through the ICRU when assistance is needed, rather than draw upon Commission members as individuals to work on other committees.

It is recognized that certain international organizations will require certain committee activities for their own needs. But it is believed that their interests can be best served by such committees concerning themselves with the applicational problems of their parent organizations rather than with problems of broad basic interest to all organizations. To provide the other organizations with adequate liaison to the proceedings of the ICRU, it is recommended that particular effort be made to draw individuals associated with other organizations into the appropriate working committees or subcommittees of the ICRU. By this means cross-fertilization of ideas can be brought about.

It is proposed that the ICRU establish working relationships with other international organizations having interests and needs in the field of radiation units, standards, and measurements to the extent that there is complete interchange of information on pertinent study programs and technical recommendations. By advance review of all such programs, the various international bodies including the ICRU can minimize wasteful duplication of efforts.

It is suggested that the ICRU invite other collaborating international bodies to provide lists of their members together with their scientific or technical backgrounds, who might be qualified to participate in the work of the Commission or its committees. By the same token the ICRU could provide similar lists which might serve as a basis for collaboration.

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