ICRU REPORT 13

Neutron Fluence, Neutron Spectra and Kerma

INTERNATIONAL COMMISSION
ON RADIATION UNITS
AND MEASUREMENTS

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INTERNATIONAL COMMISSION ON RADIATION UNITS AND MEASUREMENTS

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Preface

Scope of ICRU Activities

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

(1) Quantities and units of radiation and radioactivity,

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

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

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

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.

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

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

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

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

Current Program

In 1962 the Commission laid the basis for the development of the ICRU program over the next several years. At that time it defined three broad areas of concern to the Commission:

I. The Measurement of Radioactivity
II. The Measurement of Radiation
III. Problems of Joint Interest to the ICRU and the International Commission on Radiological Protection (ICRP)

The Commission divided these three areas into nine subareas with which it expected to be primarily concerned during the next decade. The division of work agreed upon is as follows:

I. Radioactivity
   A. Fundamental Physical Parameters and Measurement Techniques
   B. Medical and Biological Applications
II. Radiation
   A. Fundamental Physical Parameters
   B. X Rays, Gamma Rays and Electrons
   C. Heavy Particles
   D. Medical and Biological Applications (Therapy)
   E. Medical and Biological Applications (Diagnosis)
   F. Neutron Fluence and Kerma
III. Problems of Joint Interest to the ICRU and the ICRP
   A. Radiation Protection Instrumentation and its Application

The Commission established a separate planning board to guide ICRU activities in each of the subareas. The planning boards, after examining the needs of their respective technical areas with some care, recommended, and the Commission subsequently approved, the constitution of task groups to initiate the preparation of reports. The substructure which resulted from these actions is given below.

Planning Board I.A. Radioactivity—Fundamental Physical Parameters and Measurement Techniques
- Task Group 1. Measurement of Low-Level Radioactivity
- Task Group 2. Specification of Accuracy in Certificates of Activity of Sources for Calibration Purposes
- Task Group 3. Specification of High Activity Gamma-Ray Sources (Joint with P.B. II.B)

Planning Board I.B. Radioactivity—Medical and Biological Applications
- Task Group 1. In Vivo Measurements of Radioactivity
- Task Group 2. Scanning
- Task Group 3. Tracer Kinetics
- Task Group 4. Methods of Assessment of Dose in Tracer Investigations

Planning Board II.A. Radiation—Fundamental Physical Parameters

Planning Board II.B. Radiation—X Rays, Gamma Rays and Electrons
- Task Group 1. Radiation Dosimetry; X Rays from 5 to 150 kV
- Task Group 2. Radiation Dosimetry; X and Gamma Rays from 0.6 to 100 MV
- Task Group 3. Electron Beam Dosimetry

Planning Board II.C. Radiation—Heavy Particles
- Task Group 1. Dose As a Function of LET
- Task Group 2. High Energy and Space Radiation Dosimetry

Planning Board II.D. Radiation—Medical and Biological Applications (Therapy)
- Task Group 1. Measurement of Absorbed Dose at a Point in a Standard Phantom (Absorbed Dose Determination)
- Task Group 2. Methods of Arriving at the Absorbed Dose at any Point in a Patient (In Vivo Dosimetry)
- Task Group 3. Methods of Compensating for Body Shape and Inhomogeneity and of Beam Modification for Special Purposes (Beam Modification)
- Task Group 4. Statement of the Dose Achieved (Dosage Specification)

Planning Board II.E. Radiation—Medical and Biological Applications (Diagnosis)
- Task Group 1. Photographic Materials and Screens
- Task Group 2. Image Intensifier Radiography
- Task Group 3. TV Systems

Planning Board II.F. Radiation—Neutron Fluence and Kerma
- Task Group 1. Neutron Fluence, Energy Fluence, Neutron Spectra and Kerma

Planning Board III.A. Radiation Protection Instrumentation and its Application
- Task Group 1. Radiation Protection Instrumentation Handbook—Part I
- Task Group 2. Neutron Instrumentation and its Application to Radiation Protection

Because the Commission's basic recommendations on radiation quantities and units relate to the work of all of the planning boards, the Commission decided to establish a separate committee with membership drawn largely from the Commission itself to initiate the revision of ICRU Report 10a, *Radiation Quantities and Units*. Thus, the Committee on Fundamental Quantities and Units was added to the above substructure.

In 1962 the Commission decided to abandon its past practice of holding a meeting together with all of its sub-units every three years. Instead, it was decided that the Commission would receive reports from the subgroups at the time of their completion rather than at fixed deadlines. Meetings of the Commission and of the subgroups are held as needed.

The adoption of the new substructure and mode of operation was intended to alleviate some of the problems associated with the expanded program required in recent years. In the past, the Commission's attempt to administer and review the work of each of the working groups imposed a very considerable burden on the Commission itself. The need to concern itself with each detail, which was inherent in such a scheme of operation, when coupled with the procedure of completing all reports at one time, subjected the Commission members to an intolerable work load if rigorous standards were to be maintained. The new substructure and mode of operation is now beginning to produce results in the form of reports drafted by the task groups and reviewed by the planning boards. Present evidence indicates that the substructure and mode of operation, while not perfect, has to a substantial extent succeeded in alleviating the problems previously experienced.

ICRU Reports

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

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

These reports were published, as had been many of the

previous reports of the Commission, by the United States Government Printing Office as Handbooks of the National Bureau of Standards.

In 1967 the Commission determined that in the future the recommendations formulated by the ICRU would be published by the Commission itself. This is the second report to be published under this new policy. With the exception of ICRU Report 10a, which was superseded by ICRU Report 11, the other reports of the "10" series have continuing validity and, since none of the reports now in preparation are designed to specifically supersede them, will remain available until the material is essentially obsolete. All future reports of the Commission, however, will be published under the ICRU's own auspices. Information about the availability of ICRU Reports is given on page 53.

ICRU Relationships With Other Organizations

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

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

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

Operating Funds

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

Prior to 1959, the principal financial assistance to the ICRU had been provided by the Rockefeller Foundation which supplied some $11,000 to make possible various meetings. In 1959 the International Society of Radiology increased its contribution to the Commission providing $3,000 for the period 1959–1962. For the period 1962–1965 this was again increased, the Society providing $5,000. In 1960 the Rockefeller Foundation supplied an additional sum of some $4,000 making possible a meeting of the Quantity and Units Committee in 1960.

In 1960 and 1961 the World Health Organization made available the sum of $3,000 each year. This was increased to $4,000 in 1962 and this amount has been made available annually since then. It is expected that this sum will be allocated annually, at least for the next several years.

In connection with the Commission's Joint Studies with the ICRP, the United Nations allocated the sum of $10,000 for the joint use of the two Commissions.

The most substantial contribution to the work of the ICRU has come from the Ford Foundation. In December 1960, the Ford Foundation made available to the Commission the sum of $37,000 per year for a period of five years. This grant was to provide for such items as travel expenses to meetings, for secretarial services and other operating expenses. In 1965 the Foundation agreed to a time extension of this grant making available for the period 1966–1970 the unused portion of the original grant. To a large extent, it is because of this grant that the Commission has been able to move forward actively with its program.

In 1963 International Atomic Energy Agency al-

located the sum of $6,000 per year for use by the ICRU. This was increased to $9,000 in 1967. It is expected that this sum will be allocated annually at least for the next several years.

From 1934 through 1964 valuable indirect contributions were made by the U.S. National Bureau of Standards where the Secretariat resided. The Bureau provided substantial secretarial services, publication services and travel costs in the amount of several thousands of dollars.

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

Composition of the ICRU

It is of interest to note that the membership of the Commission and its subgroups totals 140 persons drawn from 16 countries. This gives some indication of the extent to which the ICRU has achieved international breadth of membership within its basic selection requirement of high technical competence of individual participants.

The membership of the Commission during the preparation of this report was as follows:

L. S. Taylor, *Chairman*
M. Tubiana, *Vice Chairman*
H. O. Wyckoff, *Secretary*
A. Allisy
J. W. Boag (1965–1966)
R. H. Chamberlain
F. P. Cowan
F. Ellis (1965)
J. F. Fowler
H. Fränz (1965)
F. Gauwerky
J. R. Greening
H. E. Johns (1965–1966)
K. Lidén
R. H. Morgan
V. A. Petrov (1965)
H. H. Rossi
A. Tsuya

Composition of ICRU Subgroups Responsible for the Initial Drafting of this Report

Serving on the Task Group on Neutron Fluence, Energy Fluence, Neutron Spectra and Kerma during the preparation of this report were:

N. Starfelt, *Chairman*
A. H. W. Aten, Jr.
K. W. Geiger
W. C. Roesch
J. W. Weale

Serving on the Planning Board on Radiation—Neutron Fluence and Kerma during that time were:

A. H. W. Aten, Jr., *Chairman*
R. S. Caswell
K. W. Geiger
F. Netter

The Task Group and Planning Board were assisted by two consultants:

E. J. Axton
D. Nachtigall

Serving as Commission Sponsors for the Planning Board were H. H. Rossi (1962–65) and J. F. Fowler (1965–69).

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

Lauriston S. Taylor
Chairman, ICRU

Washington, D. C.
July 1, 1969

Contents

Neutron Fluence, Neutron Spectra and Kerma

I. Introduction

1. Scope of the Report

This report deals with the description and measurement of neutron radiation fields. Its purpose is to bring up to date the information previously published on these topics and to make recommendations for further work. Frequent reference will be made to the previously published ICRU Report 10b (1964). The following quantities are considered: fluence and flux density, energy fluence and energy flux density, kerma and kerma rate. Some consideration is given to absorbed dose in connection with the discussion of kerma. However, dose equivalent, which requires modifying factors such as the quality factor, QF, is not considered here. An additional report will deal with high energy and space radiation and the present report is, therefore, limited to neutrons below approximately 20 MeV. For basic studies in the field of the present report we consider it necessary to work towards a good accuracy and have taken the requirement to be 5%, with a possible improvement to 1%.

Recommendations arising from each chapter are presented in Chapter V of the present report.

2. Characteristics of Neutron Radiation Fields

The particle fluence or fluence, Φ, of particles is the quotient of ΔN by Δa, where ΔN is the number of particles which enter a sphere of cross-sectional area Δa.

$$\Phi = \frac{\Delta N}{\Delta a} \tag{I.1}$$

The particle flux density or flux density, ϕ, of particles is the quotient of $\Delta\Phi$ by Δt where $\Delta\Phi$ is the particle fluence in time Δt.

$$\phi = \frac{\Delta\Phi}{\Delta t} \tag{I.2}$$

The definitions as given make no reference to the energy or speed of the particles. The flux density due to particles of a specific energy is given in terms of the differential flux density, which may be denoted by $\mathrm{d}\phi(E)/\mathrm{d}E$ which is such that $[\mathrm{d}\phi(E)/\mathrm{d}E]\,\mathrm{d}E$ is the flux density of particles with energies between E and $E + \mathrm{d}E$. Often the word spectrum is used to mean a distribution of differential flux density, and will be used so here.

The neutron radiation field is determined by the sources and by the position of absorbing and scattering materials present. The source neutrons are described by giving their energy spectrum and angular distribution. Source neutrons come from nuclear reactions. These reactions may occur in a nuclear reactor (fission), in an accelerator beam or in radioactive neutron sources. The source neutrons are slowed down and scattered by the material around them and the neutron spectrum, therefore, differs from that of the source neutrons. Particularly, large amounts of light elements (moderators and shields) reduce the energies to those characteristic of thermal motions in the scatterer. Ultimately, a "thermal" (Maxwellian) spectrum results if neutron absorption is small. The shape of the spectrum between the source energy and the thermal region is theoretically predictable (Amaldi, 1959); the slowing-down spectrum approaches a $1/E$ distribution provided certain limiting conditions are met. This is illustrated in Figure I.1 where curve number 1 shows a spectrum typical of that occurring in a moderated reactor or in a moderating medium surrounding an accelerator or a radioactive neutron source. The relative magnitudes of the source peak, the thermal peak and the $1/E$ portion depend on the nature of the moderator and on the position of measurement. Curve 2, on the other hand, illustrates the type of spectrum that arises in an environment where little slowing down is taking place, for example, in the core of a fast reactor. The spectrum is not greatly changed from that of the source neutrons. In Figure I.1 the ordinate is the quantity $E[\mathrm{d}\phi(E)/\mathrm{d}E]$, where $\mathrm{d}\phi(E)/\mathrm{d}E$ is the distribution of the flux density with respect to the neutron energy, E. The area under a curve of $\mathrm{d}\phi(E)/\mathrm{d}E$ versus E, between two energies E_1

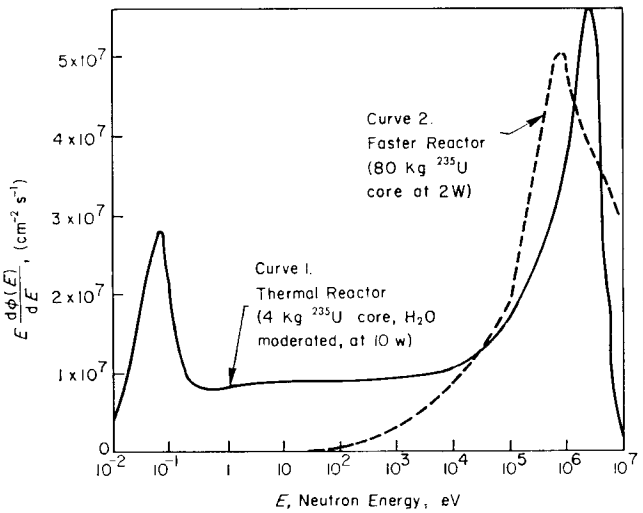

Fig. I.1. Comparison of neutron spectra in fast and thermal reactors.

and E_2, equals the flux density of neutrons in this energy range. Similarly, the area under a curve of $E[d\phi(E)/dE]$ versus the logarithm of E, (i.e. the neutron energy expressed in units of lethargy) between the same two energies, also equals this part of the flux density. (Lethargy is defined as $\ln(E_0/E)$, where E_0 is an arbitrary energy.) Thus, plotting $E[d\phi(E)/dE]$ versus the logarithm of E allows the spectrum to be displayed over a large range of energies in a way that preserves area representation of flux densities and gives a clear indication of the relative contributions of source neutrons, slowing-down neutrons, and thermal neutrons.

In general, it is neither possible nor practical to measure a neutron spectrum accurately throughout its entire range. Often, only a limited part of the spectrum is of interest. Also, available instrumentation covers only limited ranges. A cadmium absorber prevents neutrons below approximately 0.5 eV from reaching a detector and thus allows these neutrons to be distinguished from the rest. They are called "thermal" neutrons. It is not possible, however, to separate the source region from the slowing down spectrum since the dividing line depends on the energies of the source neutrons. Based on limitations of instruments, it has been customary to establish a demarcation line between "intermediate" and "fast" neutrons somewhere between 10 keV and 1 MeV. Above this energy proton recoil detectors have been considered to become useful. For the present report, no such demarcation has been adopted, since such detectors may be useful for source neutrons as well as for slowing-down neutrons. A useful practical summary of a spectrum can be obtained as follows. The thermal flux density can be obtained by a single measurement. If the spectrum between thermal and the lowest source energy can be assumed to be $1/E$, this part can also be determined in a single measurement. The spectrum in the source-energy region can often be characterised by a few simple measurements, depending on how much the spectrum is degraded from the source spectrum. If greater detail is required, the possibility of an adequate result being obtained by calculation should not be ignored. Large computers and the necessary programmes for the calculation of neutron spectra are now available in many laboratories and may be used with a few simple experimental checks instead of undertaking detailed spectrum measurements. The calculation methods are not generally discussed in this report but some account of kerma calculations is given. The experimental techniques described include methods suitable for simple representation or for spot checks on a calculated spectrum as well as methods intended to provide a detailed experimental coverage.

II. The Measurement of Thermal Neutron Flux Density (Neutron Fluence Rate)

It was indicated in the introduction, that thermal neutrons are obtained when neutrons are being slowed down in a moderator and reach thermal energies. For the present purpose, neutrons are called "thermal" when they have energies below the effective cadmium cut-off which is in the neighbourhood of 0.5 eV, depending on the cadmium thickness and angle of incidence. The thermal spectrum in a moderator will normally consist of a Maxwellian component characterized by a temperature, T, closely related to that of the actual moderator temperature, and a much weaker slowing down component ideally proportional to $1/E$, where E is the neutron energy. The low energy cut-off for the slowing down component is assumed to be at about $E = 4kT$ which is normally well below Cd cut-off (for details see Westcott et al., 1958). Therefore, a measurement of thermal neutron flux density by a cadmium difference method not only includes the Maxwellian component but also part of the slowing down spectrum which may contribute several percent to the thermal flux density.

1. Definitions Relevant to Thermal Neutron Fluxes

Flux density is defined by the number of neutrons which enter a sphere of unit cross sectional area per unit time (for a more rigorous definition see ICRU, 1968). Using the concept of neutron number density, the total flux density can be expressed as follows:

$$\phi = \int_{v=0}^{\infty} \frac{\mathrm{d}n(v)}{\mathrm{d}v} v\mathrm{d}v = \bar{v} \int_0^{\infty} \frac{\mathrm{d}n(v)}{\mathrm{d}v} \mathrm{d}v = n\bar{v} \quad \text{(II.1)}$$

where $\mathrm{d}n(v)/\mathrm{d}v$ is the neutron density (quotient of number of neutrons by volume) per unit velocity interval,

v—is the neutron velocity ($\bar{v}$ average velocity)

$$n = \int_0^{\infty} \frac{\mathrm{d}n(v)}{\mathrm{d}v} \mathrm{d}v$$

is the total density and includes neutrons of all energies.

Since the neutron spectrum is usually not well known, $\bar{v}$ cannot be established and it is not practical (see section II.2) to express the neutron flux density in the form of equation II.1. However, the relation between thermal neutron flux density and reaction rate is the most important one; also the cross section, σ, of many elements follows the $1/v$ — law, at least in the thermal region. For a detector sufficiently thin to avoid neutron self-shielding or flux depression, $\sigma(v) = \sigma_0 v_0/v$ and the reaction rate per atom becomes

$$R = \int_0^{\infty} \frac{\mathrm{d}n(v)}{\mathrm{d}v} \sigma(v) v\mathrm{d}v = \sigma_0 v_0 \int_0^{\infty} \frac{\mathrm{d}n(v)}{\mathrm{d}v} \mathrm{d}v = \sigma_0 n v_0 \quad \text{(II.2)}$$

The reaction rate is proportional only to the total neutron density and it is therefore convenient to express the total flux density by nv_0. Here σ_0 is the cross section for an arbitrarily chosen neutron velocity of $v_0 = 2200$ ms^{-1} ($T_0 = 293.6$°K, $kT_0 = E_0 = 0.0253$ eV). The neutron flux density obtained in this way is referred to as the "conventional flux density" or the "2200 meters per second flux density". The simple equation II.2 therefore applies for $1/v$ detectors in a moderated neutron spectrum; slight deviations from the $1/v$ — law can sometimes be neglected (see discussion below). For reaction rate measurements in fast fluxes (unmoderated spectra, use of threshold detectors etc.) the nv_0 convention is unsuitable since the $1/v$ — law for the cross section does not apply. The first integral of equation II.2 would then have to be evaluated to obtain the reaction rate.

For measurements in moderators with a $1/E$ epithermal spectrum by means of detectors which do not follow the $1/v$ — law in this energy range, a correction to eq. II.2 is required. Westcott et al. (1958) introduced an effective cross section, $\hat{\sigma}$, to express this connection.

$$\hat{\sigma} = \sigma_0[\mathrm{g}(T) + \mathrm{rs}(T)] \quad \text{(II.3)}$$

where T is the neutron temperature (see Section II.2). The index, r, indicates the relative contribution of the epi-Maxwellian component, this being a property of the particular moderating assembly. The index, r, can be found by determining the cadmium ratio, see ICRU (1964). The g and s are a measure of the deviation of the cross section from the $1/v$ — law in the Maxwellian (g) and epi-Maxwellian (s) region and are both functions of neutron temperature. They are tabulated by Westcott (1960); for a $1/v$ detector g = 1, s = 0. The reaction rate per atom becomes

$$R = \hat{\sigma} n v_0 \quad \text{(II.2a)}$$

The flux density for a moderated neutron spectrum may be expressed in various ways and for all results it should be clearly stated what is being measured:

1) "Conventional" flux density

$$nv_0 = \frac{R}{\hat{\sigma}} = \frac{R}{\sigma_0(\mathrm{g} + \mathrm{rs})} \quad \text{(II.2b)}$$

2) Thermal flux density below cadmium cut-off (the cadmium thickness should be stated):

$$n_{\mathrm{th}} v_0 = \frac{R - R_{\mathrm{Cd}}}{\sigma_0 \mathrm{g}} \quad \text{(II.4)}$$

This notation is used when it is difficult to establish the index, r, or if the epi-Maxwellian component does not follow the $1/E$ law. R_{Cd} is the reaction rate of the probe irradiated when covered by cadmium. A small correction may arise when for the epi-Maxwellian component between 0.1 eV and the cadmium cut-off at 0.5 eV the capture cross section is not proportional to $\mathrm{g}\sigma_0/v$ (Axton, 1963)

3) "Conventional" Maxwellian flux density:

$$n_{\mathrm{M}} v_0 = n v_0 \left(1 - \frac{4\mathrm{r}}{\sqrt{\pi\mu}}\right) \quad \text{(II.5)}$$

This notation contains the Maxwellian spectrum only and is obtained by subtracting the epi-Maxwellian component from the total flux density. The cut-off energy for the epi-Maxwellian component is given by $E = \mu\kappa T$, where $\mu \simeq 4$. For a well moderated system ($\mathrm{r} < 0.05$), $n_{\mathrm{M}} v_0$ will be only a few percent lower than $n_{\mathrm{th}} v_0$

4) "True" Maxwellian flux density

$$n_{\mathrm{M}} \bar{v} = n_{\mathrm{m}} v_0 \frac{2}{\sqrt{\pi}} \sqrt{\frac{T}{T_0}} \quad \text{(II.6)}$$

The average flux density is a function of neutron

temperature where again $T_0 = 293.6°K$ corresponds to $v_0 = 2200\ ms^{-1}$.

2. Neutron Temperature

For detectors whose cross sections closely obey the $1/v$ law the temperature of the Maxwellian component is of little significance. However, for detectors which show resonances, particularly when close to the thermal region, the Westcott g-value becomes a strong function of temperature.

Only in very favorable cases, such as in a D_2O bath at the end of a thermal column of a reactor (e.g., Bigham, 1959) is the neutron temperature identical to the moderator temperature. Normally, incomplete thermal equilibrium, combined with neutron capture in the system and neutron escape through the boundaries, causes an increase of the neutron temperature, T_n, above the moderator temperature, T_m. This increase is approximately proportional to the ratio of the epithermal to the Maxwellian flux density. A relation involving the index r (eq. II.3)

$$\frac{T_n - T_m}{T_m} = Cr \qquad (II.7)$$

is expected where T is measured in °K and C is a constant which for a gas moderator, with an atomic weight $\gg 1$, as model is calculated to be C = 1.65 (Beckurts and Wirtz, 1964). For solids, particularly graphite moderators, crystal binding effects are thought to produce higher temperature differences. This is supported by the few measurements available for graphite and a value of C, about twice the expected value is found (Küchle, 1957; Coates, 1961; Geiger and Van der Zwan, 1966).

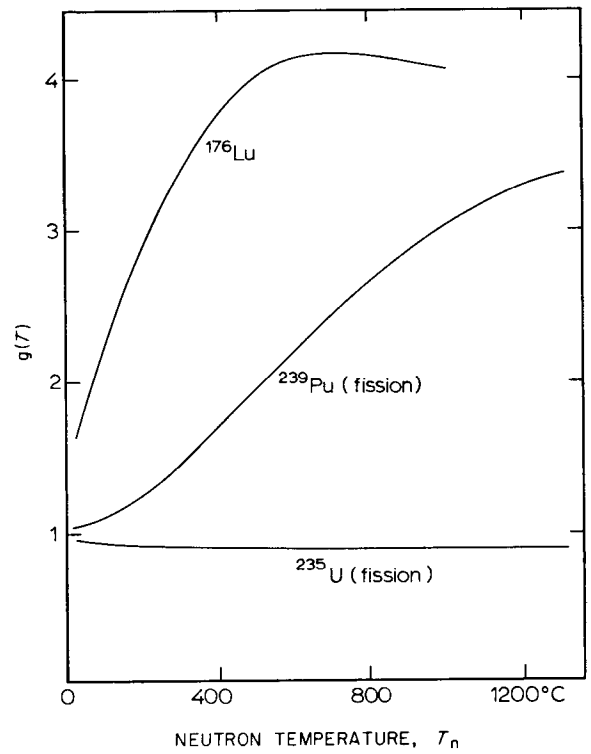

Fig. II.1. The g(T) values for ^{176}Lu, ^{239}Pu and ^{235}U (Westcott, 1960).

The neutron temperature in a moderator assembly is measured most conveniently by activating a detector, the cross section of which shows a strong deviation from the $1/v$ law near the thermal region. For temperatures up to 400°C ^{176}Lu, which has a resonance at 0.142 eV, is particularly suitable (Schmid and Stinson, 1960). The activity of the product nucleus ^{177}Lu, which decays with a half-life of 6.74 days can easily be measured. The g(T) values have been calculated by Westcott (1960) and are shown in Fig. II.1. The measuring technique consists of comparing the reaction rate in a pure Maxwellian flux of known temperature (e.g., the D_2O bath mentioned above) with the reaction rate in the assembly of interest. After subtracting the generally present epi-Maxwellian component (compare Section II.3) the quantity g(T) in the assembly can thus be obtained. The flux density ratio in the two assemblies has to be determined by simultaneous activation of a temperature independent $1/v$ detector. It is possible to use the same lutetium foil for this purpose and count the $^{176}Lu^m$ isotope ($T_{1/2} = 3.69h$) resulting from activation of ^{175}Lu which, below the Cd cut-off shows a $1/v$ cross section (Schmid and Stinson, 1960; Baston et al., 1960).

For temperatures above 400°C where g(T) for ^{176}Lu is no longer a strong function of temperature, ^{239}Pu is a more suitable detector, see Fig. II.1. The temperature is determined by comparing a fission rate measurement on ^{239}Pu with that of ^{235}U. For ^{235}U, g(T) varies little with temperature (Fig. II.1). These measurements can be made with high temperature fission chambers (Campbell et al., 1958) or by counting fission product activities in irradiated foils (Stinson et al., 1960).

3. The Epi-Maxwellian Component

Since the epi-Maxwellian component contributes to the reaction rate in a moderator assembly, a more detailed discussion is appropriate. As seen in section (II.1), no knowledge of the spectrum is needed when determining reaction rates for true $1/v$ detectors and when the flux density is expressed as "conventional flux density" (eq. II.2b) since, in this case, g = 1, s = 0.

When reaction cross sections do not follow the $1/v$ law, the g and s values must be determined from the

activation cross section resonance parameters (Westcott, 1960).

For the measurement of neutron fluences and flux densities in the slowing-down region a fairly large number of methods are now available. The simplest one to use is based on the assumption mentioned above, that the flux density per energy interval is proportional to $1/E$, applicable only to an "ideal" moderating system (Amaldi, 1959), as can be realized in a well-moderated reactor. In this case only the index, r, has to be measured. Such a determination can usually be obtained by means of a single resonance detector. Normally a detector for this purpose responds mainly to the lower energy component.

In a number of systems, e.g., a thermal neutron flux standard containing point sources distributed in a moderator (compare Section III.2b), the epi-Maxwellian component may deviate from the $1/E$ law. Consequently, different indices, r, would be found by the cadmium ratio method when using detectors with different resonance energies. For accurate flux density determinations a more detailed knowledge of the epi-Maxwellian spectrum is needed (e.g., the value of the exponent, β, in eq. III.4). This can be obtained by a multigroup calculation or by measurement.

If it is desired to measure the energy spectrum of the slowing down neutrons, observations with two or more different detector systems are required. It is, of course, advantageous if the sensitivity regions of such detector systems are as far apart as possible (compare Section III.2b).

4. Techniques for Thermal Flux Density Determinations

The measurement of a thermal neutron flux density is based on the absolute determination either of a reaction rate directly, e.g., the $^{10}B(n,\alpha)$ reaction (De Juren and Rosenwasser, 1954), or of the disintegration rate of a radioactive nuclide. Absolute counting of gold detector foils is preferred since Au is an almost perfect $1/v$ detector in the thermal region and has a high neutron capture cross section; gold also is a monoisotopic element and can be obtained at high purity. Absolute disintegration rates of radioactive foils are often measured by the β-γ coincidence method (e.g., Campion, 1959; Axton, 1963). It is often convenient to calibrate a standard flux assembly by absolute counting. A list of institutes having such assemblies is given in ICRU (1964) Table II B 1. Some others, which make use of some fixed arrangement of neutron sources in a moderator, may now be added: Michikawa et al. (1961), Matescu and Nohorniak (1962), Hargrove and Geiger (1964), Cochran et al. (1964), Peetermans (1965). The neutrons, however, may also be produced by using, for example, a well moderated reactor running at low power (Axton, 1963) or accelerator produced deuterons on beryllium (Axton et al., 1967). The highest accuracies which have been achieved in thermal neutron flux density determinations are between ± 1 and $\pm 2\%$. In all these cases absolute counting of gold foil was carried out.

Comparison with the unknown flux density can be made using a large variety of detectors by relative counting which is easily done to high precision. For low flux densities, counting of β-particles in a 4π gas counter is recommended; for higher densities, γ-counting is more suitable. Data on some important neutron detecting reactions are given in Table II.I. None of the activation detectors are perfect $1/v$ detectors, vanadium foil is probably one of the most suitable ones (Fehr and Bryce, 1965; Geiger and Van der Zwan, 1967). For relative measurements, as well as for absolute measurements of lesser accuracy, a wide range of elements for neutron detection may be incorporated into instruments for routine operation. The nuclear reaction caused by a neutron is then detected. The instruments may be proportional counters, ion chambers, semiconductor detectors, photographic emulsions, etc. The appropriate detecting nuclides are listed in Table II.I. Methods for measurement of high and low flux densities have been listed in ICRU (1964) and the present discussion will be limited to new developments.

With the wider availability of ^3He gas, proportional counters filled with ^3He instead of $^{10}BF_3$ have become important since higher pressures and therefore higher sensitivities can be achieved (Mills et al., 1962). Where there is a need for very small neutron probes, semiconductor particle detectors can be used in conjunction with converters from neutrons to charged particles; ^{10}B, ^6Li, ^{157}Gd or fission foils (Dearnaley et al., 1962; Sakai, 1962; Rauch et al., 1967). Spatial resolution similar to that of foil detectors is possible (Babcock et al., 1959). However, a limiting factor in high intensity radiation fields is radiation damage to the semiconductor detector. This has been overcome by using an evacuated tube of sufficient length between converter and detector (Ajdačić et al., 1963).

If an immediate indication of the neutron flux density is not required, recording of nuclear tracks from converter foils may be chosen. Most insulators are capable of recording tracks of heavily ionizing particles which can, after etching, be made visible in a microscope (Fleischer et al., 1965b). Materials such as cellulose nitrate record tracks of particles as light as deuterons; it is possible therefore to use the $^{10}B(n,\alpha)$ reaction with this method. Most widely used, however, are ^{235}U fission foils together with mica, glass or mylar films. An extremely large range of flux densities can

TABLE II.I—*Useful thermal neutron detectors*

Data from BNL 325 and Westcott (1960) except where noted otherwise.

Nuclide or Element	Reaction	Half-Life of Nuclide Produced	$\sigma_0 \times 10^{24}$ for Production cm^2	Westcott		Application
				g(20°C)	s(20°C)	
^{3}He	$(n,p)^{3}H$	12.3 y	5327	1.0		PC, SD
^{6}Li	$(n,t)^{4}He$	stable	945	1.0		SC, SD
^{10}B	$(n,\alpha)^{7}Li$	stable	3837	1.0	negligible	PC, PT, SC, IC, SD
^{23}Na(IS)	$(n,\gamma)^{24}Na$	15.0 h	0.534	1.0	0.15[a]	BC
^{45}Sc(IS)	$(n,\gamma)^{46}Sc$	85d	22.3			FC
^{51}V	$(n,\gamma)^{52}V$	3.8 min	4.9	1.0	0.083[b]	FC
^{55}Mn(IS)	$(n,\gamma)^{56}Mn$	2.58h	13.3	1.0	0.666	BC, FC
^{59}Co(IS)	$(n,\gamma)^{60}Co$	5.24y	36.6	1.0	1.736	FC
^{63}Cu	$(n,\gamma)^{64}Cu$	12.8h	4.5	1.0	0.77[a]	FC
^{115}In	$(n,\gamma)^{116}In^{m}$	54 min	157	1.019	19.8	FC
^{157}Gd	$(n,\gamma)^{158}Gd$	stable	242000	0.854	−0.85	SD
^{197}Au(IS)	$(n,\gamma)^{198}Au$	2.70d	98.8	1.005	17.3	FC
^{235}U	fission	many	577	0.976	−0.04	PC, IC, PT, SD

Notes: (IS) denotes natural monoisotopic element.

Main applications: BC: bath counting, incorporated into salt solution
FC: foil counting
PT: particle track production in insulating materials
IC: use in ionization chamber
PC: proportional counter
SC: incorporated into scintillator
SD: converter for semiconductor detector

[a] Calculated from excess resonance integrals of Dahlberg et al. (1961).

[b] Geiger and Van der Zwan (1967).

be measured by varying both the time and the amount of fissionable material.

At the very highest flux densities the impurities already present in the recording material are used. Track recording in most materials is nearly 100% efficient. With fission foils that are thick compared to the fission fragment range, absolute measurements to ±5% are possible; the sensitivity of the system, i.e. the number of fission fragment tracks divided by the product of the number of incident neutrons and the cross section of the fissionable material, is 1.2×10^{-5} $barn^{-1}$, in good agreement with theoretical calculations (Prêtre et al., 1966).

Efforts have been made to overcome the inconvenience of track counting by microscope. A read-out based on the average luminous flux passing through opaque plastic film has been proposed (Prêtre et al., 1966). This film has to be thinner than the minimum fission fragment range and should be appropriately etched to produce clean holes. Another method is to aluminize one side of a thin film (Unruh et al., 1967). Etching takes place from the other side; the etchant penetrates through the hole and dissolves the aluminized area around the hole. When the film is held over a bright light, the holes become clearly visible.

5. Corrections in Thermal Flux Density Determinations

It should be remembered that a variety of corrections enter the relation between flux density and reaction rate. These have been discussed in detail in ICRU (1964) and include: flux depression, self shielding in detector, effect of cadmium absorbers on neutron spectrum and others.

The correction for flux depression can be avoided by using very thin foils or by using liquid foils with a moderating ratio equal to that of the medium in which they are to be used (Walker et al., 1965). The use of liquids may also facilitate absolute counting.

III. Specification of Fast and Intermediate Neutron Radiation Fields

This chapter will be concerned with techniques for measuring fluence, flux density and energy spectrum for intermediate and fast neutrons, i.e. from 0.5 eV to 20 MeV.

In the following, both such measuring techniques as are independent of neutron energy, and thus give a measure of the total neutron flux density or fluence, and such techniques as are suitable for measurements of neutron spectra will be discussed.

1. Measuring Techniques

A summary of the different methods discussed in this section is given in Table III.I.

a. Detectors Based on Proton Recoil Measurements

There are many types of detectors based on proton recoil measurements, either as simple detectors or as spectrometers. One of the main advantages of these detectors is that they are based on a cross section which has been accurately determined over a wide range of energies. Proton recoil detectors are capable of covering the energy range from about 1 keV to the highest energies of interest in the present survey (20 MeV). The energy of the recoiling proton is measured by its ionization or by its track length in a solid or liquid or gas. Detailed reviews of various recoil detection methods are given by several authors (Marion and Fowler, 1960). In most applications the hydrogen which provides the scattered protons forms part of the medium in which the energies of the scattered protons are measured. The most widely used counters of this type are the proportional gas counter, the solid or liquid scintillation counter and the photographic emulsion. Some work has also been done with hydrogen-filled cloud chambers. A different principle is used in the radiator type of counter in which a solid hydrogenous material acts as the radiator and the scattered protons are measured in an adjoining counter.

For all proton recoil detectors other than the telescope types the derivation of neutron spectra requires the unfolding of the proton recoil energy distribution.

The Proportional Counter. The proportional counter has been used to determine the spectrum over the energy range from 1 keV to 1 MeV. See Fig. III.1 (Weale et al., 1966). To cover this range it is necessary to use gas filling pressures from about half an atmosphere to about 10 atmospheres, depending on the counter design. Wall effects become important at sufficiently high energies at any operating pressure. It has been shown, however, that corrections for these effects can be made (Parker et al., 1963). This type of counter can be used in a monodirectional neutron beam or in a reactor environment. For the reactor environment a spherical counter of the type described by Benjamin et al. (1964) affords an easier subject for wall effect corrections. The lower energy limit for a given operating pressure is usually determined by background counts from γ-rays associated with the neutron field. It has been shown that a discrimination against γ-rays can be obtained by the use of pulse-shape discrimination (Bennet, 1962); see also Fig. III.1. It should be noted that computer calculations are required for the accurate analysis of proportional counter results. (See, for example, Benjamin and Kemshall, 1967). The use of a proportional counter as a neutron spectrometer from 1 keV to 1 MeV is still in an early stage of its development. Highly developed techniques are required both in the production and the use of the spectrometer.

Solid or Liquid Scintillators. An organic scintillator provides a very efficient fast neutron detector. For example a 0.25 cm^3 crystal of stilbene gives a count rate of 0.11 counts per second for a flux density of one neutron per square centimeter and second for 1 MeV neutrons. Several plastic scintillating materials and organic liquids with comparable efficiencies are also available. Apart from its high efficiency the solid or liquid scintillator has the advantages of fast response and of compactness. However, there are a number of disadvantages. First, the relation between proton energy and light pulse is non-linear. This necessitates calibration and a rather laborious analysis of results if the scintillator is to be used for spectrometry (Burrus, 1965; Verbinski et al., 1967). Second, the material also acts as an efficient γ-ray detector. Correction has to be made for γ-ray effects or the γ-ray counts must be eliminated by pulse-shape discrimination. This disadvantage has effectively fixed a lower neutron energy limit of about 1 MeV for reliable spectrometry work. Third, double scattering tends to increase the proportion of proton recoils at the higher energies. This effect can be reduced by reducing the scintillator size, and it can be determined accurately by calibration. Fourth, part of the energy of some of the recoil protons is lost because they escape from the surface of the scintillator. This effect can be calculated and it can be reduced by increasing the scintillator size. Furthermore, in stilbene the light output varies appreciably according to the angle between the proton recoil direction and the crystal axis. This effect can be avoided by using a scintillating plastic or liquid.

Despite the various corrections required for precise work, the solid or liquid scintillator recoil detector is an important instrument for neutron detection over the energy range 0.5 MeV to 20 MeV.

The Radiator Counter. A proton recoil radiator counter consists usually of a solid hydrogenous radiator placed so that some of the scattered protons enter a gas-filled or semi-conductor counter in which their energies are measured. If the angle of acceptance of recoils is con-

TABLE III.I—*Summary of methods for the measurement of fast and intermediate neutron fields*

Measurement	Methods and Section	Energy Range	Sensitivity[a]	Accuracy[a]	Comments
Flux Density	Proton-recoil proportional counter. III.1.a	1 keV–3 MeV	Low to Medium	High	Complicated to use for lower energies, frequently requires computer, sensitive to γ-rays at low neutron energies
Spectrum		1 keV–3 MeV	Low to Medium	Medium	
Flux Density	Proton-recoil solid and liquid scintillators. III.1.a	>0.5 MeV	High	Low	Simple but non linear, very sensitive to γ-rays, complications due to reactions with carbon above 15 MeV
Spectrum		>0.5 MeV	High	Low	
Flux Density	Radiator Counter. III.1.a	>0.1 MeV	Low	High	Collimated beam or point neutron source, sensitive to γ-rays
Spectrum		>0.1 MeV	Low	High	
Fluence	Proton recoils in emulsions. III.1.a	>0.5 MeV	High	Low	Cheap and easy, but time consuming. Suitable for small spaces, insensitive to γ-rays
Spectrum		>0.5 MeV	High	Low	
Fluence	Threshold detector. III.1.b	>0.6 MeV	Medium	High	Cheap, suitable for small spaces, insensitive to γ-rays.
Spectrum		>0.6 MeV	Low	Low	
Fluence or Flux Density	Cd and B shielded detectors. III.1.b	<0.5 MeV	High	High	Cheap, in part suitable for small spaces, sensitivity to γ-rays depends on type
Spectrum		<25 keV	Medium	Low	
Fluence	Resonance detectors. III.1.c	<1 keV	Low to medium	Low to medium	Small size, insensitive to γ-rays
Spectrum		1 eV–2.85 keV	Low to medium	Low to medium	
Flux Density	$^{3}He(n,p)T$ proportional or surface barrier counter. III.1.d	<5 MeV	Medium	Medium	Sensitive to thermal neutrons and to γ-rays, complications due to recoils
Spectrum		0.2–5 MeV	Medium	Medium	
Flux Density	$^{6}Li(n,\alpha)T$ scintillator, surface barrier counter. III.1.d	<5 MeV	Medium	Medium	Sensitive to thermal neutrons and to γ-rays
Spectrum		0.2–5 MeV	Medium	Medium	
Flux Density	$^{10}B(n,\alpha)^{7}Li$ proportional counter, ion chamber, scintillation counter. III.1.d	low energy	Medium	Medium to high	Sensitive to thermal neutrons, fairly insensitive to γ-rays, easy to use
Fluence or Flux Density	$^{235}U(n,f)$ or $^{239}Pu(n,f)$ fission chamber, activation foil or track registration. III.1.d	mainly low energies	Medium to high	High	Sensitive to thermal neutrons, insensitive to γ-rays
Fluence or Flux Density	Manganese sulphate bath, other bath methods, graphite assemblies. III.1.e	all energies	High	High	Very large size, energy independent response, insensitive to γ-rays, but sensitive to thermal neutrons. Essentially suitable for neutron emission rates for small sources, can be used for collimated beams
Flux Density	Large liquid scintillators. III.1.e	all energies	High	High	As preceding method, but sensitive to γ-rays
Flux Density	Long counter. III.1.e	all energies	Medium to high	High	Fairly large size, energy independent response, not sensitive to γ-rays, restricted to point sources and unidirectional fluxes. Simple and reliable
Fluence or Flux Density	Moderating spheres. III.1.e	all energies	Medium	Medium	Fairly large size, can be made to have low sensitivity to γ-rays, insensitive to thermal neutrons, isotropic sensitivity. Simple to use, but spectrum determinations require complicated calculations
Spectrum		all energies	Medium	Low	
Spectrum	Time of flight. III.1.f	all energies	Low	High	Requires pulsed source and long flight path

[a] The terms "high," "medium," and "low" are used for sensitivity and accuracy because the absolute values depend strongly on the neutron energy and on the construction of the instrument. In general, detectors listed as having "low" sensitivity are of little interest for protection purposes.

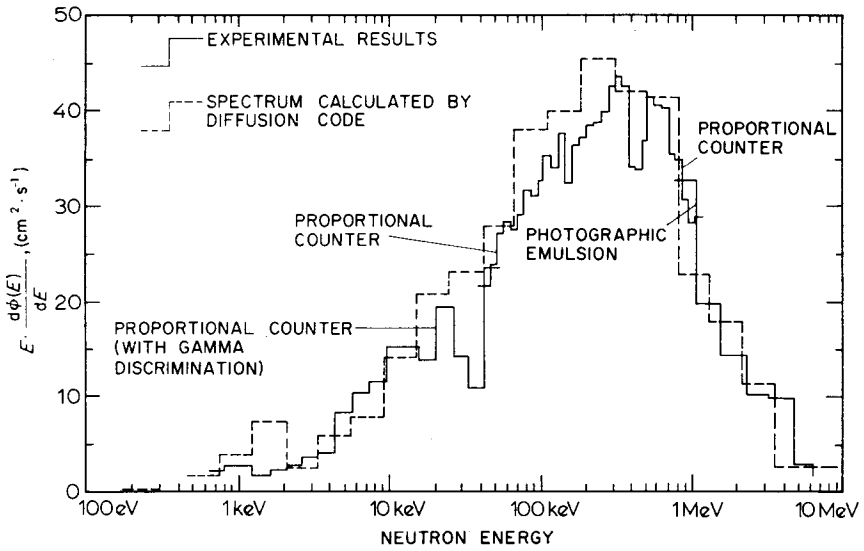

Fig. III.1. Neutron spectrum at the centre of a fast critical assembly measured by proton recoil techniques (Weale et al., 1966).

fined to protons recoiling with nearly the full neutron energy, the system is a useful spectrometer for beam work and it is known as a neutron telescope. In this form it has been used chiefly in nuclear physics experiments to measure the energy spectrum of neutrons produced in nuclear reactions. In another form the radiator is the hydrogenous filling of one proportional counter and the scattered protons are detected partly in that counter and partly in an adjoining counter (Perlow, 1956). In this form it has been used for reactor beam spectrometry (Redman and Roberts, 1958). A little-used but elegant version of this principle uses the "shaped-foil" radiator so that only one n-p scattering angle is involved, with a point neutron source or collimated beam (Salgir and Walker, 1967). Because of the loss of proton energy in the radiator, the solid-radiator telescope has a lower energy limit of about 1 MeV for neutron energy determinations. The gas-radiator telescope has been used to count neutrons at a minimum energy of 100 keV but there are serious difficulties in establishing a reliable calibration. A very compact form of the solid-radiator counter uses a surface barrier counter to measure the energies of protons scattered from a thin polythene radiator (Dearnaley and Ferguson, 1962).

The Photographic Emulsion Method. The photographic emulsion method can be used in mono-directional beams or in multi-directional neutron fields, either as a detector for neutron fluence or as a spectrometer. A typical emulsion plate consists of a 1 cm² piece of emulsion, 600 microns thick, mounted on a glass plate. There are well-established methods of developing the track of a recoil proton in such an emulsion and the emulsion provides an effective scatterer with a hydrogen atom content of about 3×10^{22} cm^{-3} (Marion and Fowler, 1960). Track lengths can be measured with a microscope of magnification about 400. A 5 MeV proton has a track length of about 180 microns and this can be measured with high accuracy; but accuracy deteriorates considerably at proton energies below 500 keV, and 100 keV is about the limit of measurement. A 100 keV proton has a range of about 1 micron and the track in normal emulsion contains only a few grains. In routine scanning work it is difficult to select such tracks reliably and consistently from the background grains due to γ-radiation. This is not simply a question of grain size, but is determined also by the density of the background, i.e. by the sensitivity of the emulsion. Finer grain emulsions with grain diameters of 0.07 μm have been reported (Campan et al., 1963). The emulsion sensitivity has to be as low as possible so that electron tracks are discriminated against, but must be high enough to record the passage of the fastest recoil protons of interest. An emulsion which is just sensitive enough to record the tracks of 5 MeV protons can be scanned satisfactorily for proton tracks against a background γ-ray exposure of 20 roentgen. A semiautomatic method has been described for recording proton track lengths (Benjamin and Nichols, 1963). With this technique it is possible for one person to maintain a steady output of 100 tracks per hour. Since about 5000 tracks are required to determine a 10 energy-group multi-directional neutron spectrum to an accuracy of 10% or better, each such spectrum takes 50 operator hours to produce. Proton tracks are measured at energies down to 200 keV and the derived neutron spectrum appears to be meaningful for energies above 500 keV (Weale et al., 1966; Bradna, 1967).

TABLE III.II—*Threshold detectors*

	E_{min} MeV	$E_{0\,pr}$ MeV	$E_{0\,eff}$ MeV	σ_{eff} mb	$\bar{\sigma}_{fission}$ eq. III.2 mb	$\bar{\sigma}_{fission}$ for most recent fission spectrum mb
$^{103}Rh(n,n')^{103}Rh^m$(a)	0.04					
$^{115}In(n,n')^{115}In^m$(b)	0.34	0.6	1.12	282	178	212
$^{31}P(n,p)^{31}Si$	0.7	2.0	2.70	126	31.1	40.7
$^{32}S(n,p)^{32}P$	1.0	2.2	2.70	234	57.3	76.1
$^{58}Ni(n,p)^{58}Co + {}^{58}Co^m$	0.0	1.9	2.79	478	97 (R)	120(F)
$^{54}Fe(n,p)^{54}Mn$	0.0	2.4			66 (R)	89(F)
$^{27}Al(n,p)^{27}Mg$	1.9	3.8			3.66	4.67
$^{56}Fe(n,p)^{56}Mn$	3.0	5.7	6.34	54	1.08	1.26
$^{46}Ti(n,p)^{46}Sc$	1.6				11 (R)	13.0(F)
$^{24}Mg(n,p)^{24}Na$	4.9	6.4	7.0	120	1.45 (R)	1.62(F)
$^{27}Al(n,\alpha)^{24}Na$	3.2	6.6	7.49 7.42	83(B) 62(Z)	0.683	0.765
$^{63}Cu(n,2n)^{62}Cu$	11.0	12.0	12.7 12.84	830(B) 650(Z)	0.116	0.125
$^{237}Np(n,f)$	0.0	0.6	0.63	1640	1269	1368
$^{238}U(n,f)$	0.0	1.3	1.62	582	277	335

Notes: E_{min} indicates the minimum neutron energy required for the reaction calculated on the basis of nuclear masses.

$E_{0\,pr}$ indicates the "practical" threshold energy. As such, the energy at which the cross section has one tenth of the value it has at the beginning of the plateau, has been chosen. As the latter point is not clearly defined, the value of $E_{0\,pr}$ is somewhat vague. It does, however, give an idea of the energy range covered by the detector reaction.

The two columns headed $\bar{\sigma}_{fission}$ indicate the average cross-section for neutrons produced in the fission of ^{235}U by thermal neutrons. The best values for these cross-sections have changed appreciably in recent years, since it has become known that the energy spectrum of fission neutrons does not agree well with eq. III.2 (Grench and Menlove, 1968; McElroy, 1969).

Most of the values listed have been obtained by McElroy, mainly on the basis of measurements of Grundl (1968) and of Fabry (1967). Column 6 contains $\bar{\sigma}$-values for a spectrum corresponding to eq. III.2 with $E_n = 1.29$ MeV, column 7 $\bar{\sigma}$-values which would fit the most accurate spectrum available for neutrons from the thermal fission of ^{235}U. For this spectrum some data have been included taken from Fabry (1968) for detectors not included in McElroy's list. These data are marked (F). Corresponding figures for a spectrum of eq. III.2, marked (R), have been recalculated on the basis of relative values from earlier work (Senaux, 1965; Zijp, 1965a).

The quantities σ_{eff} and $E_{0\,eff}$ are defined by the relation:

$$\bar{\sigma}_{fission} \int_0^\infty \frac{d\phi(E)}{dE}\, dE = \sigma_{eff} \int_{E_{0\,eff}}^\infty \frac{d\phi(E)}{dE}\, dE$$

where $d\phi(E)/dE$ represents the energy spectrum of fission neutrons, produced by the action of thermal neutrons on ^{235}U. In principle an infinite number of combinations of σ_{eff} and $E_{0\,eff}$ can be made to fulfill this condition, but the combination selected is the one which does not cause a change in the calculated activity if a slight variation occurs in the exponential term in eq. (III.2), i.e., in E_n (Grundl and Usner, 1960). The values used for $d\phi(E)/dE$ are once more equal to those given by eq. III.2 for $dN(E)/dE$. However, the present values of σ_{eff} have been recalculated from the earlier determinations (Beaugé, 1963; Zijp, 1965a) to correspond to the $\bar{\sigma}$-values in column 6 of Table III.II. In cases where the two earlier values lead to very different values of σ_{eff}, both have been listed and identified by (B) and (Z).

(a) $^{103}Rh^m$ is difficult to measure absolutely. A method for this purpose has been described recently (Nagel and Aten, 1966).

(b) $^{115}In^m$ is measured by means of the 0.335 MeV γ-ray line, which occurs in 50 ± 2 percent of the transitions (Heertje et al., 1964; Grench and Menlove, 1968).

The photographic emulsion has the advantages that it is a very compact detector with excellent energy resolution and it provides a permanent record of the results. It requires only a small neutron fluence. Typically, a neutron fluence of 10^8 cm^{-2} at energies greater than 500 keV gives a convenient track density. The disadvantage of the method is the laborious nature of the scanning process.

b. Threshold Detectors, Including Cd and B Shielded Detectors

Threshold detectors are a convenient means of measuring the neutron fluence or flux density over a single broad range of the fast neutron spectrum (Table III.II). By using a set of threshold detectors together it is also possible to attain an approximate measure of the fast neutron spectrum. Most of the available detectors are based on activation but those using fission reactions can also be made in the form of fission chambers or in an arrangement containing fission foils combined with mica, glass or plastic sheets for fission track registration (cf. Section II.4). The activation sample has the advantages of small bulk and considerable robustness but the method is relatively insensitive and does not give a direct reading. In many cases the sensitivity can be increased—if required—by a chemical separation of the active product.