

**NUCLEAR DATA AND MEASUREMENTS SERIES**

**ANL/NDM-23**

**Remarks Concerning the Accurate Measurement  
of Differential Cross Sections for Threshold Reactions Used in  
Fast-Neutron Dosimetry for Fission Reactors**

by

Donald L. Smith

December 1976

**ARGONNE NATIONAL LABORATORY,  
ARGONNE, ILLINOIS 60439, U.S.A.**

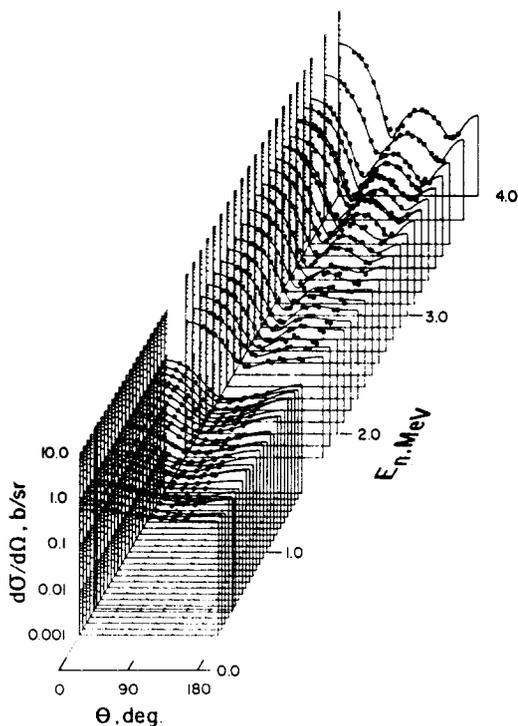
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In January 1975, the research and development functions of the former U.S. Atomic Energy Commission were incorporated into those of the U.S. Energy Research and Development Administration.

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## NUCLEAR DATA AND MEASUREMENTS SERIES

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## PREFACE

This report contains the text of a paper presented at the International Atomic Energy Agency (IAEA) Consultants' Meeting on Integral Cross-Section Measurements in Standard Neutron Fields held in Vienna, Austria, November 15-19, 1976.

Donald L. Smith  
Argonne National Laboratory  
December 1976

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ABSTRACT

Some remarks are submitted concerning the measurement of differential cross sections for threshold reactions which are used in fast-neutron dosimetry for fission reactors. The objective is to familiarize the reader with some of the problems associated with these measurements and, in the process, to explain why the existence of large discrepancies in the data sets for many of these reactions is not surprising. Limits to the accuracy which can be expected for these cross sections in the near future--using current technology and available resources--are examined in a general way and recommendations for improving the accuracy of the differential data base for dosimetry reactions are presented.

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## I. INTRODUCTION

Apparently the only reliable way to quantitatively predict macroscopic phenomena (such as radiation damage) which will be observed for commercial power reactors--on the basis of test results from low-power critical facilities or from high-fluence materials test reactors--is to possess a knowledge of the neutron spectra and relative power or fluence levels of these facilities as well as an understanding of the neutron-energy dependence of the phenomena of interest. Neutron dosimetry is the term applied to that technology which is used for the purpose of defining the intensities and spectral properties of neutron fields in nuclear reactors. Detailed quantitative knowledge about a variety of selected nuclear reactions is basic to the application of current neutron dosimetry techniques.

The development of consistent sets of cross section data for a selected group of favorable dosimetry reactions has not turned out to be an easy task. The current accuracy goal of better than 5% ( $1\sigma$ ) has not been achieved with the possible exception of a few so-called "Category I" reactions. At an early stage, it became clear that an international effort would be desirable in order to achieve the stated goal and that it would have to involve investigation of decay schemes and a variety of "benchmark" integral measurements as well as differential (monoenergetic) measurements.

Considerable progress has been made in recent years towards improving the knowledge of decay data for the radioactive products

from fast-neutron reactions for dosimetry monitors. The development of the Ge(Li) detector for gamma-ray spectrometry is largely responsible for this progress since most dosimetry reactions produce gamma-ray active daughters which can be investigated using this method. However, the importance of some careful evaluations [e.g. Ref. 1] should not be depreciated.

Similarly, a great deal of effort has been expended in the development of benchmark and standard neutron fields for integral testing of differential cross sections for dosimetry reactions [e.g. Ref. 2]. There has been a generally well-considered effort to measure the integral cross sections for selected dosimetry reactions using the available facilities. While the agreement of various integral data is generally not within the stated accuracy goal, the situation is steadily improving [e.g. Ref. 2].

Sophisticated numerical procedures for unfolding reactor-neutron spectra from the results of integral measurements have evolved in the last few years [e.g. Ref. 3]. However, these procedures do not yield unique results. Therefore, the need for accurate, direct measurement of differential cross sections remains undiminished [e.g. Ref. 4].

The accuracies to which differential cross section data for most dosimetry reactions are known are far short of the requirements. In fact, this area is currently a weak aspect of neutron dosimetry. There are both historical and technical reasons why this condition exists.

Differential cross section data is obtained from measurements performed at "monoenergetic" accelerator facilities. Although there are notable exceptions, the main emphasis at the majority of these laboratories has been on basic-nuclear-structure research rather than on development of a data base for nuclear applications. The accumulation of detailed, accurate data for quantitative applications requires redundancy of measurements and an attention to numerous details which are unnecessary for the realization of many basic research objectives. Traditionally, this sort of quantitative activity has been generally considered unappealing by members of the basic-research community. Finally, most basic nuclear research has centered on charged-particle reactions rather than neutron reactions.

The neutron energy region from 8-14 MeV has traditionally been a difficult one for monoenergetic measurements, and will probably continue to be a source of problems, because of the lack of a good monoenergetic neutron-source reaction to cover it. The scarcity and uncertainty of data in this region reflects this problem [5]. It will be seen in Section IV.F that this handicap will most probably be overcome by brute force. That is, that the characteristics of secondary reactions associated with proton bombardment of triton and deuteron bombardment of deuterium will have to be studied in sufficient detail to permit accurate corrections to the data to be determined for measurements made in this region. Progress in this energy region will be strongly dependent upon the availability of large accelerators with the requisite

beam energy and intensity capabilities.

Most monoenergetic measurements are ratio measurements, and there has been little consistency in the selection of standards. Many of the available data sets are reported only as cross sections with no published ratios. Thus, without knowledge of the exact values used for the standard cross sections, it is difficult to re-normalize old results to account for revisions of the standard cross sections. The alternative is probably to reject them.

With the exception of certain fission cross sections and some other standards such as the  ${}^6\text{Li}(n,\alpha){}^3\text{H}$ ,  ${}^3\text{He}(n,p){}^3\text{H}$  and  ${}^{10}\text{B}(n,\alpha){}^7\text{Li}$  reactions, there has been little coordinated effort to improve the accuracy of differential cross sections for reactions relevant to dosimetry applications. Until just recently, the major contribution to improving the knowledge of these cross sections has come from compilation and evaluation efforts [e.g. Refs. 4-6]. Careful evaluation of the existing data base [6] has led to the rejection of some data sets which are greatly discrepant with respect to other corresponding sets. However, the uncertainties in many of the evaluated cross sections are still large-especially in energy intervals where the available data are sparse [4]. Evaluation cannot generate a reliable set of cross sections from a poor data base.

The remainder of this paper is concerned with a description of some details associated with differential cross section measurements. The objective is to indicate several potential sources of random and systematic error associated with these measurements. Estimates of reasonable accuracy goals are made on the basis of current

technology and available resources for the measurements. Although the treatment is general, details from specific reactions are presented as examples. The experimental techniques used in differential measurements are diverse. No attempt is made to treat all of these thoroughly. As the title of the paper suggests, only threshold reactions of interest for fast-neutron dosimetry are considered. Measurement of reactions with no clearly defined thresholds--such as the  $(n,\gamma)$  reaction--involves different techniques than those to be discussed here. Furthermore, emphasis is placed on measurement of cross sections for reactions producing gamma-ray active daughters and for which detection of the gamma radiation forms a part of the measurement. In recent years, beta-particle detection techniques have declined in importance for dosimetry applications. This subject is not treated in this paper. Measurement of fission cross sections by detection of the fragments with an ion chamber or track recorder is described elsewhere [e.g. Refs. 7 and 8]. Detection of fission fragments is an attractive method since the fragment energies are large and their range in matter is short. The quantity of sample material used is generally about two-orders-of-magnitude smaller than is the case for gamma-ray measurements; however, the detection efficiency is commensurately larger so the sensitivities of differential measurements are similar for both techniques. In principle,  $(n,p)$  and  $(n,\alpha)$  cross sections could be measured by charged-particle detection. Presently, these cross sections can be measured with greater accuracy by the activation method whenever that is applicable. It must be kept in mind that

comments made concerning various experimental techniques relate only to their application in differential dosimetry measurements. The capabilities and limitations of these techniques may very well be quite different in other contexts.

## II. CONCEPT OF DIFFERENTIAL CROSS SECTION MEASUREMENTS

In a broad sense, there are similarities between integral and differential cross section measurements for dosimetry reactions. Both involve determination of the neutron dose received over a specified time interval, characterization of the neutron spectrum, measurement of the induced activity in the sample, conversion of the observed activity to a reaction rate based on a knowledge of the decay properties of the daughter, and finally, computation of a cross section after the application of relevant corrections. The major differences are:

- i) Geometry: In differential measurements, most of the neutrons incident upon the sample originate from a small region of space (the target). This is also usually the case in  $^{252}\text{Cf}$  irradiations. However in reactors, the neutrons impinge on the sample from all directions in space with less spatial bias.
- ii) Neutron spectrum: In differential measurements one strives to obtain a neutron source which is nearly monoenergetic. In integral measurements, the neutrons are continuously distributed over wide energy ranges.
- iii) Neutron intensity: The intensity of accelerator neutron sources is generally much lower than for integral facilities

(often by orders of magnitude). This leads to poorer statistics in the measurement of small cross sections. The necessity for larger samples leads to additional uncertainties caused by absorption and scattering of radiation as well as geometrical effects.

iv) Fluence stability: Standard and reference neutron fields utilized for integral measurements are generally quite stable and reproducible (usually to within  $\sim 1\%$ ), whereas accelerator sources can be quite unstable and are not easy to reproduce with high accuracy.

To a large extent, these differences are responsible for the generally poorer accuracy of differential data in comparison with integral data. However, there are procedures for dealing with these problems, and it appears possible to obtain differential data which are nearly as accurate as integral data except at energies for which the cross sections are very small (microbarns) or for reaction products with long half lives (years).

The general relationship between the differential cross section, the reaction rate and the observed detector counts for a simple irradiation with a steady source of monoenergetic neutrons is given by the formulas

$$R = F N G_n \eta_n \sigma \quad (1)$$

$$C = b \epsilon G_r \eta_r R (1 - e^{-\lambda t_E}) \cdot e^{-\lambda(t_W + t)} \quad (2)$$

where

- R = reaction rate,
- F = neutron fluence,
- N = total number of sample atoms,
- $G_n$  = geometric factor for neutron irradiation,
- $\eta_n$  = neutron absorption and scattering factor,
- $\sigma$  = differential cross section,
- C = detector count rate (detection of daughter gamma-ray activity) at time t after the start of a count,
- b = factor to account for mass yields (of fission products), decay branching, isotopic abundance, etc.
- $\epsilon$  = gamma-ray detection efficiency,
- $G_r$  = geometric factor for measurement of decay radiation,
- $\eta_r$  = decay radiation absorption and scattering factor,
- $\lambda$  = decay time constant,
- $t_E$  = irradiation time (assumed constant fluence level),
- $t_W$  = wait time between end of irradiation and start of daughter activity measurement.

### III. SENSITIVITY OF DIFFERENTIAL CROSS SECTION MEASUREMENTS

There are limits to the sensitivity of differential cross section measurements which are imposed by various experimental considerations. These limits are important because they define minimum values for measurable cross sections. Measurements attempted below these limits are not likely to yield very accurate results. Experimental factors to consider in estimating the sensitivity of conventional differential measurements are:

- i) Neutron intensity: Neutron generators which are designed to yield 14-MeV neutrons are generally capable of producing as many as two-orders-of-magnitude higher intensity neutron fluences than are accelerators used to produce neutrons over a wider range of lower energies. The present discussion excludes neutron generators since neutrons with energies  $< 14$  MeV are the main concern for fission reactors. Considering such factors as target stability, neutron yield from conventional source reactions, resolution and geometry, the neutron fluence available for differential-cross-section measurements cannot be expected to exceed  $F \sim 10^9$  neutrons /sec/sr on a sample using presently available accelerators.
- ii) Sample size: The size of samples to be used for differential measurements is limited by consideration of geometry and the absorption and scattering of radiation. In the present discussion it is assumed that gamma-ray activity is measured. A practical limit for the size of a sample (all atoms) is  $N < 2 \times 10^{23}$  atoms. This limit must be lowered if the gamma-ray energy is relatively low or if the atomic number is large so that gamma-ray absorption effects are significant.
- iii) Detector count rate: If the full-energy-peak yield of a particular gamma ray emitted from an irradiated sample is less than  $C \sim 1$  count/sec, experience has shown that the effects of background are troublesome and it is difficult to

make yield measurements with the accuracy desired.

- iv) Detector efficiency: Unless the decay gamma-ray spectrum is quite simple, accurate measurement of decay yields has to be made using a Ge(Li) detector. The absolute gamma-ray detection efficiency  $\epsilon \sim 0.05$  is a reasonable upper limit for gamma rays with  $E_\gamma \lesssim 1$  MeV detected by a Ge(Li) detector.
- v) Irradiation time: Because of the inherent tendency of accelerator beam intensities to drift with time, neutron fluence is rarely constant. It is feasible to make a correction for this effect by recording the neutron fluence level as a function of time. When possible, it is desirable to limit irradiation times so that  $\lambda t_E < 0.1$ . Under these conditions, the measurement of reaction rates will not be severely influenced by the stability of the neutron fluence. Experience with accelerator operation has shown that  $t_E \sim 10^5$  sec is a practical upper limit for the irradiation time.
- vi) Wait time: It is possible to adhere to the requirement  $\lambda t_W \ll 1$  for all useful dosimetry reactions. This avoids waste of activity acquired from the irradiation.
- vii) Count time: There is little to be gained from counting samples for much longer than the half life of the daughter activity ( $t_C \lesssim t_{1/2}$ ). However achievement of the desired accuracies for measured differential cross sections requires that the statistical uncertainties in the detector counts be

kept as small as possible. The uncertainty due to statistics should certainly not exceed  $\sim 3\%$  which requires that  $C t_C > 10^3$ .

These considerations provide constraints on Eqs. (1) and (2). It is possible to derive an order-of-magnitude formula based on Eqs. (1) and (2), as well as other considerations mentioned above, which is useful in estimating the smallest cross section values which can be measured with reasonable accuracy using available facilities and experimental techniques which are discussed in later sections of this paper.

The formula is

$$\sigma_{\min} = 10^{-7} b f_C (t_{1/2}/t_E) \quad (3)$$

where  $b$  and  $t_E$  are defined in Section II and

$t_{1/2}$  = half life for daughter activity,

$$f_C = \begin{cases} 1 & \text{if } C t_{1/2} \geq 10^3 \text{ sec} \\ (10^3/t_{1/2}) & \text{otherwise.} \end{cases}$$

The factor  $f_C$  is included to permit consideration of counting limitations expressed in Item (vii) above ( $f_C > 1$  implies that counting limitations reduce the sensitivity).

Table I provides examples of the application of this formula over a wide range of half lives. It is interesting to compare the estimated limits with the current minimum cross section values reported in the literature. In general, it is the region from threshold up to  $\sim 6-8$  MeV excitation (the approximate nucleon binding energy) which is of greatest importance for dosimetry monitor reactions.

It would require a separate study to ascertain whether these limitations to differential measurements will present any significant problems with regard to the establishment of a data base for dosimetry reactions which is adequate to meet the needs for reactor applications. Clearly, such a study should be made so that necessary and sufficient accuracy levels can be established for specific reactions. Otherwise, the risk exists that important measurements will not be undertaken, or that resources will be wasted on achieving high accuracy where it is not required.

#### IV. CHARACTERISTICS OF MONOENERGETIC ACCELERATOR NEUTRON SOURCE

Neutrons are produced at "monoenergetic" accelerator facilities by means of nuclear reactions induced by the bombardment of targets with charged-particle beams. If the neutrons produced were truly monoenergetic, with precisely known energies, and emanated isotropically from an idealized point source, then the only tasks facing the experimenter would be to measure the fluence and correct for minor geometric effects. However, since the neutron sources are not ideal in this regard, it is necessary for the experimenter to correct cross section data for the effects of the departure from ideality. Factors which have to be considered are:

- i) Neutron energy resolution: Even if the neutrons are produced entirely by a single reaction process with a well-defined Q value, the energies of neutrons incident on the sample will be spread somewhat owing to finite target thickness, kinematics and finite charged-particle-beam-energy resolution. The consequences are important.

- ii) Neutron-energy definition: Methods utilized in definition of neutron energy are subject to uncertainty.
- iii) Angular-distribution effects: Neutron emission is not isotropic. This can lead to uncertainties in determination of the neutron fluence at the sample.
- iv) Secondary-neutron reactions in the target: There are often several neutron-producing reaction channels open for a given incident charged-particle energy.
- v) Background from target assemblies: At higher energies, all target assemblies produce background neutrons. The background can be variable if it is due to build up of contaminants in the vacuum system or to variations in the beam optics during an irradiation.

These particular factors will be examined to determine how they can influence the measurement of accurate differential cross sections and the generation of recommended cross section values by means of evaluation of the existing data base. Following this, the characteristics of the most commonly used "monoenergetic" neutron-producing reactions will be reviewed briefly.

#### A. Neutron-Energy Resolution

At low energies, neutron-energy resolution is dominated by charged-particle energy loss in the targets. Here yield considerations establish a lower practical limit for target thickness. However, at higher energies, particularly for neutron-producing reactions involving deuterium and tritium targets, the resolution is dominated by kinematic effects. At all energies, the energy

resolution of the charged particle beam--normally a few kilovolts-- contributes to the neutron-energy resolution. All factors considered, the practical limit of neutron-energy resolution is  $\sim 20 - 100$  keV for  $E_n < 5$  MeV,  $\sim 100 - 200$  keV for  $E_n < 10$  MeV and  $> 200$  keV for  $E_n > 10$  MeV in the measurement of differential cross sections.

There are two important consequences of finite resolution in differential cross section measurements. One is that significant uncertainties can exist in determination of the effective reaction thresholds and of the magnitudes of cross section values for energies just above these thresholds. Evaluators are familiar with situations where cross section data sets for some reactions are in disagreement by factors of two or more near threshold. Faced with data bases such as these, it is unrealistic to expect evaluators to produce sets of numbers which are reliable. Threshold uncertainties can be very significant if the threshold falls in a region of a reactor spectrum where the neutron intensity changes rapidly with energy. A second consequence of finite resolution is experienced in reactions where the cross section fluctuates considerably with energy. Examples are the  $^{27}\text{Al}(n,p)^{27}\text{Mg}$  and  $^{32}\text{S}(n,p)^{32}\text{P}$  reactions. Data sets measured with different resolutions are practically incomparable for such reactions unless the energy scales and resolutions are precisely determined. Historically, this requirement has rarely been met and evaluators have had to deal with conditions which discouraged the production of reliable evaluated numbers.

With present technology, it is possible to determine the energy resolution of specific measurements to within a few kilovolts for targets less than 100-keV thick. By utilizing available range-energy data for analyzing the passage of heavy charged particles in matter, and by including detailed analysis of the effects of kinematics in computation of cross sections from raw data, measurers of differential data could progress a long way towards elimination of the deleterious effects of resolution uncertainty in their experimental data. It is impossible to select a universal number which will represent a lower limit to the uncertainty which is likely to persist in differential data due to resolution uncertainties. The effects depend too critically on reaction details. However, a comparison has been made of the results of two recent measurements of the neutron inelastic-scattering cross section for iron--a reaction which is notoriously susceptible to resolution uncertainties. The agreement of the coarse ( $\sim 50$  keV) resolution data with the fine ( $\sim 1-2$  keV) resolution data is excellent (within a few percent over a 1-MeV neutron-energy range) when the latter data set is energy-averaged to  $\sim 50$  keV resolution [10].

Both measurers and evaluators must share in the responsibility of minimizing resolution uncertainties so as to improve the differential data base for dosimetry reactions. Measurers need to accurately determine the resolution of their measurements and report and properly document their values. Evaluators need to energy-

average all data for individual reactions to equivalent resolution, and reject data sets which do not report experimental resolutions or which are poorly documented, prior to performing their evaluations. While such a procedure may seem harsh and undemocratic, it appears to be the only practical way to establish the standards of quality in the data base for dosimetry reactions which will be required to meet the stated objectives.

#### B. Neutron-Energy Definition

Much of what was said in the preceding section about the effects of resolution uncertainty applies to energy definition. By utilizing recent Q-value and nuclear mass data, carefully calibrated electromagnetic beam analysis and, in certain instances, time-of-flight techniques, it now appears possible to determine fast-neutron energies to better than  $\sim 5$  keV as long as the resolution is  $< 100$  keV. This accuracy should be sufficient for dosimetry applications.

However, complications arise when several samples are irradiated simultaneously by placing them at several angles relative to the incident charged-particle beam. Accurate definition of the neutron energy then involves careful determination of the geometry as well as the calibration procedures mentioned previously. Although this procedure has been utilized by competent researchers who very carefully attended to the requisite details, it is apparent that the method has a large potential for systematic error and should be avoided when possible. Unfortunately, this is about the only way researchers at small accelerator facilities can enlarge their range of accessible

neutron energies. The problem is one of economics as well as technology. However, achievement of the goal of a better-than-5-keV accuracy for routine neutron energy definition is predicated on the assumption that irradiations will be performed with neutrons emitted near zero degrees.

### C. Angular Distribution Effects

Neutron emission from charged-particle-induced reactions is not isotropic. Neutron yield at back angles can be an order-of-magnitude smaller than at zero degrees for some of these reactions. While some very careful experiments which took this effect into consideration have been performed, it is evident from some other data sets in compilations of differential dosimetry data that large systematic errors exist because samples were exposed at angles other than near zero degrees, while the neutron fluence was measured in the forward direction, and incorrect angular-distribution data for the neutron source reactions were used in analysis of the experimental results.

Even if care is taken to correct the activation data from multiple-sample exposures for the source-reaction angular distribution, it is likely that the uncertainty in the final results will be at least  $\sim 5\%$  due to this effect alone since the angular distributions are usually not known any more accurately than this. Therefore, it is recommended that multiple-angle measurements be avoided if the uncertainties in the differential data base are to be reduced below the present level. Measurements of neutron fluence and

sample irradiations should both be performed in the vicinity of zero degrees. Then, small corrections for neutron-emission anisotropy can be made with considerable reliability.

#### D. Secondary-Neutron Reactions

The principal neutron-source reactions used for monoenergetic measurements on reactions of interest for fast-reactor dosimetry are the  ${}^7\text{Li}(p,n){}^7\text{Be}$ ,  $\text{T}(p,n){}^3\text{He}$ ,  $\text{D}(d,n){}^3\text{He}$  and  $\text{T}(d,n){}^4\text{He}$  reactions. The latter reaction is primarily a source of neutrons with energies above 14 MeV, however it has been used for measurements at energies as low as  $\sim 12$  MeV by means of back-angle irradiations. Each of these reactions yields a single neutron-energy group over a limited range of proton or deuteron energy. However, these reactions are commonly used at higher energies where other neutron-producing-reaction channels are open. Under these conditions, the neutron sources can no longer be considered monoenergetic. These secondary reactions will be discussed in Section IV.F. In practice, for lithium, deuterium and tritium targets, the secondary reactions yield lower-intensity and lower-energy neutrons by comparison with the primary reactions. For some reactions which have been used in activation measurements, the yield of lower-energy neutrons exceeds that of the highest-energy neutrons [e.g. Ref. 11].

The effect of secondary neutrons on measurement of differential cross sections depends upon two major factors:

- 1) The intensity of these neutrons relative to the primary neutrons and the magnitude of the energy gap between

mary and secondary neutrons.

- ii) The relative shapes of the excitation functions for the standard and "unknown" reactions in a particular measurement.

If the secondary neutrons form a significant fraction (say > 30%) of the total fluence, if the standard or "unknown" cross section is changing rapidly in energy, or if the thresholds for the standard and "unknown" reaction are quite different, then the presence of secondary neutrons can produce a significant error in the measured "unknown" cross section unless accurate corrections are made. For example, in a measurement of the differential cross section for the  $^{27}\text{Al}(n,p)^{27}\text{Mg}$  reaction at  $\sim 10$  MeV, relative to  $^{238}\text{U}$  fission, failure to correct for neutrons from the  $\text{D}(d;n,p)\text{D}$  reaction would lead to an error of  $\sim 13\%$  [12]. There is need for improved knowledge of the neutron spectra from secondary reactions. There should be no fundamental problems encountered in doing this, and the contribution to the overall uncertainty in measurements of dosimetry cross sections resulting from secondary neutrons can probably be limited to < 1% for primary neutron energies below  $\sim 10$  MeV. The relative yield of secondary neutrons at higher energies is presently not well-enough known to permit any quantitative predictions to be made; however, the situation is certain to be less favorable than at lower energies.

#### E. Background Neutrons

Background neutrons are produced when charged particle beams

collide with components of the flight tube and target assemblies. Background usually varies with time and can be a significant problem at higher bombarding energies. Use of high -Z materials in fabrication of the flight tube and target assemblies is essential, especially for slits, collimators and target beam stops. The use of clean components and well-trapped high vacuum systems is desirable. In any event, the effects of background neutrons can be measured directly with good accuracy. If this is done carefully, the uncertainty in the measured cross sections due to this effect will generally be negligible. An example of the relative importance of background is seen from the results of a measurement of the cross section for  $^{58}\text{Ni}(n,p)^{58}\text{Co}$  relative to  $^{238}\text{U}$  fission using the  $^7\text{Li}(n,p)^7\text{Be}$  reaction as a neutron source. For 7.5-MeV protons and a 50-keV lithium target,  $\sim 7\%$  of all fissions and 4% of all sample activations were produced by background neutrons--predominantly from (p,n) reactions with the tantalum beam stop. If no correction for this background were made, the measured ratio would be in error  $\sim 3\%$  [13].

#### F. Details of Specific Neutron Source Reactions

It is unfortunate that there are very few satisfactory neutron-source reactions which can be used for routine monoenergetic measurements at neutron energies up to  $\sim 20$  MeV. The requirements for a useful source reaction are [11]:

- i) Relatively-intense neutron yield with a dominant fraction--preferably all of the neutrons belonging to the discrete

group with the highest energy.

iii) The targets should be relatively convenient to use and not excessively expensive or hazardous to laboratory personnel.

iv) The source reaction should be useful over a significant range of neutron energies.

The  ${}^7\text{Li}(p,n){}^7\text{Be}$ ,  $\text{D}(d,n){}^3\text{He}$ ,  $\text{T}(p,n){}^3\text{He}$  and  $\text{T}(d,n){}^4\text{He}$  reactions generally satisfy these requirements to varying degrees and they are the most widely used source reactions. Other less common reactions have been used from time-to-time for special applications.

#### F.1 The ${}^7\text{Li}(p,n){}^7\text{Be}$ Reaction

This reaction has been a popular neutron source at accelerator laboratories for many years. Target preparation is simple, inexpensive and not hazardous. The targets are generally fabricated by vacuum evaporation of lithium metal on high-Z backings. With proper cooling, these targets can dissipate several hundred watts/cm<sup>2</sup> of beam power and provide relatively stable yield for more than 100 hours continuous operation. The neutron reactions produced by proton bombardment of natural lithium are listed in Table II. This neutron source is monoenergetic over a relatively narrow energy range (Fig. 1). However, the  ${}^7\text{Li}(p,n){}^7\text{Be}$  reaction remains dominant over a larger energy range and this source has been used for precision cross-section measurements at neutron energies up to  $\sim 6$  MeV at the Argonne National Laboratory Fast-Neutron Generator [13,14]. The characteristics of the 2nd neutron group are well known in this energy range [14-16]. There is some information available on the continuous group of neutrons from the  ${}^7\text{Li}(p,n){}^3\text{He}{}^4\text{He}$  breakup

reaction which yields significant numbers of neutrons for  $E_p > 6$  MeV [14]. As a result of this breakup reaction, the useful range of the lithium neutron source is  $E_n < 7$  MeV.

## F.2 The T(p,n)<sup>3</sup>He Reaction

This reaction has been adopted at many laboratories as a substitute for the  ${}^7\text{Li}(p,n){}^7\text{Be}$  reaction. It has been evaluated for proton energies below  $\sim 10$  MeV [17]. It is a monoenergetic source over a much wider energy range (Table II and Fig. 1) and, if a gas target is used, it is a more intense source of neutrons above  $\sim 1$  MeV than the  ${}^7\text{Li}(p,n){}^7\text{Be}$  reaction (Table III and Fig. 2). The main drawback to this reaction is that it is inherently hazardous to laboratory personnel. Extensive precautions are required to insure that tritium is not released in the laboratory environment. Metal tritium (predominantly Ti-T) targets are safer to use, but they offer lower yield and are subject to background problems at higher energies (Table III).

Breakup neutrons are produced at proton energies above  $\sim 8.5$  MeV from the T(p,np)D reaction (Table II). There is little quantitative information on this reaction. The T(p,d)D reaction competes strongly with the T(p,n)<sup>3</sup>He reaction at proton energies above  $\sim 10$  MeV [18] and this is reflected in the steady decrease with energy of the T(p,n)<sup>3</sup>He reaction at higher energies (Fig. 2). More measurements are needed to determine the detailed characteristics of secondary neutron production from protons or tritium. However, because of the anticipated increase of this secondary production

with energy and the apparent decline of the  $T(p,n)^3\text{He}$  reaction cross section, this neutron source appears to be relatively unfavorable for the production of high-energy neutrons. A practical limit of  $\sim 10$ - $11$  MeV is estimated, though this is speculation based on limited information (Fig. 1).

### F.3 The $D(d,n)^3\text{He}$ Reaction

This reaction appears to satisfy most requirements for a desirable source of neutrons with energies above  $\sim 4$  MeV (Table II and Figs. 1 and 2). Metal deuterium targets are impractical--except for low-energy measurements--because of  $(d,n)$ -reaction background. Gas targets can be fabricated readily and are safe (though sometimes troublesome) to use [12]. The  $D(d,n)^3\text{He}$  reaction has been evaluated for deuteron energies below  $\sim 10$  MeV [17]. Breakup neutrons from the  $D(d,np)D$  reaction are produced for deuteron energies above  $\sim 5$  MeV (Table II). There is a fair amount of quantitative information available about the breakup neutrons [12,19]. While the breakup cross section increases rapidly with energy above threshold, the  $D(d,n)^3\text{He}$  reaction also increases with energy so that the relative yields change much less rapidly. For deuteron energies of  $\sim 7$  MeV, the breakup neutrons constitute  $\sim 25\%$  of all zero-degree neutrons [12]. At  $\sim 10$ -MeV deuteron energy, the corresponding figure is  $\sim 35$ - $40\%$ . There is a large energy gap between the  $D(d,n)^3\text{He}$  and  $D(d,np)D$  neutrons.

It was previously mentioned that the neutron-energy range 8-14 MeV is a difficult region for measurements because of the lack of a

suitable monoenergetic neutron source reaction (Fig. 1). A case can be made for using the  $D(d,n)^3\text{He}$  reaction to cover this entire energy region, and this has been done by Santry and Butler using the Chalk River Tandem Van de Graaff accelerator [e.g. Ref. 20]. Unfortunately, 10-11 MeV deuteron beams are not available at most of the laboratories currently involved in activation cross section measurements. Also, if this reaction is to be used for accurate cross section measurements up to  $\sim 14$  MeV, it appears necessary to obtain more accurate and detailed information on the  $D(d,np)D$  reaction than is currently available.

#### F.4 The $T(d,n)^4\text{He}$ Reaction

This reaction is predominantly a source of high-energy neutrons ( $> 14$  MeV) and is of less importance for fission-reactor applications (Table II and Figs. 1 and 2). It will not be treated in this paper.

#### F.5 Other Reactions

There are a number of neutron-source reactions, other than those mentioned above, which are used from time-to-time in fast-neutron differential measurements. Invariably, these suffer from one or more defects such as low relative yield of the high-energy group, overall low-intensity neutron production, target fabrication difficulties, limited useful range, high cost, etc. For example, the  $^{14}\text{C}(d,n)^{15}\text{N}$ ,  $^{15}\text{N}(d,n)^{16}\text{O}$  and the  $^9\text{Be}(d,n)^{12}\text{C}$  reactions have been used with considerable success by researchers at Geel in some selected activation cross section measurements over the neutron-

energy range 6-12 MeV using a small Van de Graaff accelerator [11, 21]. The common feature of these measurements was that they were made for reactions with high thresholds so that only the highest-energy neutron group produced the observed reactions.

#### V. TREATMENT OF THE EFFECTS OF NON-IDEAL NEUTRON SOURCES IN DETERMINATION OF DIFFERENTIAL CROSS SECTIONS

Eqs. (1) and (2) assume that an ideal monoenergetic neutron source is utilized for a cross section measurement. The properties of realistic sources were reviewed in Section IV. Detailed knowledge of the characteristic of the source is of little value unless it is integrated into the data processing procedures.

Symbolically, this is accomplished by substituting the equation

$$R = F_1 N G_{n1} \eta_{n1} \sigma_1 \left( 1 + \sum_{i=2}^m \frac{F_i G_{ni} \eta_{ni} \sigma_i}{F_1 G_{n1} \eta_{n1} \sigma_1} \right) \quad (4)$$

for the reaction rate  $R$  in the place of Eq. (1). The sum is over the groups of secondary neutrons from the target. In fact, individual groups corresponding to a particular reaction can be further subdivided if they are broad in energy owing to resolution or breakup effects.

The fluence ratios  $(F_i/F_1)$  can be computed if the detailed properties of the source reaction are known. Computation of the factors  $G_n$  and  $\eta_n$  is often tedious but usually straightforward. If the factors  $(F_i/F_1)$  are not too large, then the determination of the ratios  $(\sigma_1/\sigma_1)$  is not critical. The ratios  $(\sigma_i/\sigma_1)$  can

generally be estimated with sufficient accuracy from systematics or previously available data. If need be, an iterative process can be used to improve the accuracy. This method is exhibited in detail in available reports [12,13].

Practical integration of these detailed corrections into routine data processing requires utilization of digital computers. Fortunately, adequate computational capability exists at most of the laboratories where such measurements are being made.

## VI. DETERMINATION OF NEUTRON FLUENCE

Determination of neutron fluence  $F$  and the gamma-ray detection efficiency  $\epsilon$  are the dominant elements of a cross section measurement. The former is usually the most difficult task and is one of the most controversial areas of fast-neutron physics.

There are two general methods for deducing neutron fluence. One involves a direct count of the incident neutrons passing through the sample. The second involves measurement of the ratio of the unknown cross section to some standard cross section. In the latter case, the absolute fluence need not be known, but the accuracy to which the unknown cross section can be determined is limited by the accuracy of the standard cross sections as well as by the precision of the ratio measurement. Two rather precise ratio measurements can lead to discrepant cross sections if the standard cross sections are inconsistent. This is an important point which should always be taken into consideration by evaluators.

## A. Measurements Involving Absolute Fluence Determination

Some neutron cross section measurements at neutron energies up to a few MeV have been made using "black" neutron detectors with very high efficiencies (nearly 100%) for detection of incident neutrons (e.g. Ref. 22). This method requires extensive shielding in order to form well-collimated neutron beams and pulsed-beam time-of-flight techniques to reduce background. Furthermore, there is a limitation on fluence levels which can be measured due to electronic considerations. Generally, these conditions can be met in practice. Although this method has been utilized extensively for measurement of fast-neutron capture cross sections (e.g. Ref.23), it has not been used often in threshold-dosimetry-reaction measurements. Although limited in energy range, this method appears capable of yielding good accuracy (< 3%) in the measurement of neutron fluence up to a few MeV.

Calibrated flat-response detectors--predominantly long counters--have been used extensively in activation measurements for neutron energies up to several MeV (e.g. Ref.20). The efficiency of these detectors is flat to within  $\sim 10\%$  from  $\sim 0.1$ -5 MeV [24]. They are generally calibrated using a standardized radioactive neutron source. It is essential that care be taken to determine the response of a long counter to scattered neutrons for each new experimental configuration. While these detectors possess the advantage of high stability of response when laboratory conditions are constant, the possibilities for systematic error are many owing to the sensitivity of the response to changes in these laboratory conditions. Here it is to be understood that laboratory conditions comprise not only the physical layout of the target area but also the characteristics of the

neutron-source reaction used for the measurements. If great care is taken in measurements using a long counter, it appears that the neutron fluence can be determined to within 3-5% [25].

Associated-particle detection is a method of absolute neutron fluence determination which has been used in fast-neutron studies [e.g. Refs. 26 and 27]. In this method, recoil  $^4\text{He}$  particles from the  $\text{T}(d,n)^4\text{He}$  reaction or  $^3\text{He}$  particles from the  $\text{T}(p,n)^3\text{He}$  or  $\text{D}(d,n)^3\text{He}$  reaction are detected. There is a one-to-one relationship between the charged-particle emission and neutron emission. This method suffers from two serious defects. First, due to kinematic considerations there are limited ranges of neutron energies for which the method can be applied. Secondly, there are numerous technical problems--associated mainly with electronic and geometric considerations--which can lead to serious errors in fluence determination. While the method can probably be used to measure neutron fluence to  $\sim 3\%$  accuracy [26], great care has to be taken to avoid systematic errors.

A fundamental problem associated with measurement of absolute neutron fluence by all the methods described above is that generally what is being determined is the absolute fluence within a particular solid angle which is strongly correlated with the monitor detector, but not necessarily with the sample being irradiated. In short, there is room for additional systematic error in determination of the true fluence at the sample. Furthermore, absolute fluence measurements have to be made each time a cross section is

measured. This requires extensive duplication of effort which can be partially avoided if the ratio method is used.

## B. Ratio Measurements

In ratio measurements the absolute neutron fluence is not measured each time which results in an economy of effort. However, one has to either make an absolute fluence determination at some point in the program, or transfer the responsibility to others by relying solely on the use of "standards". The ratio method is the most widely used approach in fast-neutron measurements and should be fully discussed. All of the problems related to this method can be grouped into two categories which are described by the following questions:

- i) "How accurately are the "standard" cross sections known?"
- ii) "What are the experimental limitations which govern the extent to which accuracy in the knowledge of a standard cross section can be fully utilized in measurements of the cross section ratio for an unknown relative to that standard?"

It is apparent that a crucial issue is the selection of a practical standard for use in differential dosimetry reaction measurements by the ratio method. It is suggested that the choice of this standard is not a foregone conclusion at this time, but one which should be carefully examined.

### B.1 Use of Several Dosimetry Threshold Reactions as Standards

Unfortunately, there has been rather indiscriminate use of

several threshold reactions--the more common of which are the  $^{27}\text{Al}(n,p)^{27}\text{Mg}$ ,  $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ ,  $^{32}\text{S}(n,p)^{32}\text{P}$ ,  $^{56}\text{Fe}(n,p)^{56}\text{Mn}$  and  $^{58}\text{Ni}(n,p)^{58}\text{Co}$  reactions--as "standards" for measurement of less-well-known cross sections. This is unacceptable practice in development of a consistent data base for dosimetry. Not one of these reactions is well enough known at present to be used as a standard in a program with a goal of 5% or better over all accuracy. Furthermore, the excitation functions for some of these reactions exhibit fine structure which guarantees that they will remain unsuitable for use as standards.

Much of the data resulting from measurements of this sort should be rejected in critical evaluations of dosimetry reactions. Otherwise, there will remain many hidden, convoluted normalization factors in the evaluated cross sections and progress toward development of a consistent set of dosimetry reaction cross section values will be severely hampered.

## B.2 Use of The Hydrogen Standard

It has generally been contended with axiomatic certainty that the  $\text{H}(n,n)p$  reaction is the ultimate standard for differential measurements on dosimetry reactions. There is a good basis in fact for this contention. This reaction possesses almost all the desired features for a standard. The cross section is large and varies smoothly with energy. Cross-section values can actually be computed theoretically with an accuracy which may exceed experimental capabilities. The reaction can be used over a wide range of neutron energies. Furthermore, the cross section appears to be very well

known ( $\sim 1-2\%$ ) for neutron energies of interest for fast-neutron dosimetry. For a number of years, there were no changes in the accepted cross section values for this reaction. A few years ago, there were some minor revisions which altered the accepted values by  $\sim 1-2\%$  at certain energies [28,29].

However, the problem with the hydrogen standard is not that the cross section itself is uncertain, but rather that it is very difficult experimentally to achieve accuracies approaching the cross section accuracy in routine measurements which in one way or another are based on this standard. There is considerable potential for systematic error in measurements with hydrogenous proportional counters, hydrogenous scintillators or proton recoil counters. Errors can result from uncertainties in determining the hydrogen content of detectors and stabilizing this content. Spectra from hydrogenous detectors are often difficult to analyze and there is always unwanted background which must be subtracted. Because of the large energy loss generally experienced by neutrons which scatter from hydrogen, recoil proton spectra are broad and overlap the background spectra. Proportional counters and scintillators are also gamma-ray sensitive which increases the background. Each type of hydrogenous detector has its own characteristic features. These are worthwhile reviewing briefly.

Gas proportional counters have low efficiency, require extensive geometric corrections and suffer from wall effects at higher energies [30]. The proton recoil spectra extend to zero pulse height and suffer from noise and background interference. Efficiency

calibration is critically dependent upon gas pressure. Gamma-ray interference is a problem. If care is taken, fluence can be measured with an accuracy of  $\sim 3\%$  at energies up to  $\sim 1$  MeV [26,30].

Organic scintillators offer the advantages of well-defined geometry and hydrogen content, large efficiency and a wider energy range for applications when compared with a gas proportional counter. However, the proton-recoil pulse-height spectra also extend to zero, there is a strong gamma-ray sensitivity, and the pulse-height response is nonlinear [31,32]. The accuracies claimed for the calibration of these detectors are rarely better than  $\sim 5\%$ .

Proton recoil detectors, which detect protons emitted in the forward direction by means of surface barrier detectors, have low efficiency, but offer the desirable features of gamma-ray insensitivity and improved capability for discrimination of proton events from the background [33]. Accurate calibration of these detectors entails careful determinations of the geometry and of hydrogen content with time. Furthermore, it is very difficult to monitor these radiators for such effects. However, these detectors can probably be routinely calibrated to an accuracy of  $\sim 5\%$  or better [33].

So, the attractiveness of the hydrogen standard is somewhat diminished by the difficulties involved in trying to use it in cross section measurements for dosimetry reactions. This does not imply that it should be abandoned. On the contrary, the technology for utilizing the hydrogen standard will probably develop in due time to the point where most of the current problems associated

with its use are satisfactorily overcome.

### B.3 Use of Fission Standards

There has been extensive experimental and evaluative effort devoted to improving the accuracy of fast-neutron fission cross sections which are considered important for fission-reactor applications [34]. As a result of this effort, the cross sections for some of these reactions, especially  $^{235}\text{U}(n,f)$ , are known with sufficient accuracy to merit consideration as standards. Recent estimates of the uncertainty in  $^{235}\text{U}$ ,  $^{238}\text{U}$  and  $^{237}\text{Np}$  fast-neutron fission are reproduced in Table IV. Most of the data on  $^{238}\text{U}$  and  $^{237}\text{Np}$  has been obtained from ratio measurements relative to  $^{235}\text{U}$  fission. Therefore the accuracy of these cross sections is tied to  $^{235}\text{U}$ . The uncertainties in the  $^{238}\text{U}$  and  $^{237}\text{Np}$  fission cross sections near threshold appear to be largely due to discrepancies in energy scales for various measurements. When these are resolved, the accuracy will undoubtedly improve. Progress in improving the accuracy of the  $^{237}\text{Np}$  fission cross section has been limited since this reaction does not play a major role in current fission reactor technology. However, this reaction has merits for use as a standard and effort should be made to improve the accuracy of its cross section.

$^{235}\text{U}$  fission is sensitive to low-energy neutrons. At higher bombarding energies, where lower-energy neutron groups and background neutrons are present, threshold reactions such as  $^{238}\text{U}$  and  $^{237}\text{Np}$  fission are more practical standards than  $^{235}\text{U}$ . As indicated in Table IV,  $^{235}\text{U}$  and  $^{238}\text{U}$  fission can be used as standards

in differential dosimetry measurements below 15 MeV with the expectation that the uncertainty in the results attributable to the standard cross section will not exceed 5-6%. Furthermore, the accuracy of the  $^{235}\text{U}$  and  $^{238}\text{U}$  cross section is likely to continue to improve as the result of sustained research effort in support of fission-reactor technology.

Although the cross sections for standard fission reactions are presently not known as accurately as the hydrogen standard, it is much easier to perform accurate measurements relative to the fission standards than it is to use the hydrogen standard. At present, it seems that measurements relative to the fission standards are likely to produce differential cross section values which are as accurate as the corresponding measurements relative to the hydrogen standard. Improvement of the accuracy of measurements relative to the hydrogen standard will require perfection of measurement techniques, whereas in the case of the fission standards it is an improvement in cross section accuracies which is required. At present, it does not seem likely that accuracies of much better than  $\sim 5\%$  in fluence determination can be achieved routinely regardless of which method is selected.

It is worthwhile to review briefly some of the desirable features of measurements relative to fission standards. There are two ways in which fission can be utilized as a standard. One involves measurement of induced gamma-ray activity from irradiated samples of fissionable material. This is the less desirable

method since the uncertainties in the mass ratios and decay characteristics must be added to the cross section uncertainty. Direct detection of fission-fragments with an ionization chamber is preferable. Low-mass chambers (Fig. 3) can be constructed to minimize scattering [e.g. Ref. 13]. The unknown sample and fissionable deposit can be placed back-to-back to minimize systematic errors in fluence determination. The background pulses from noise and alpha activity are easily distinguished from the larger fission fragment pulses. The calibration of fission detectors is almost entirely governed by the mass and isotopic abundances of the fissionable material used. A well-developed technology exists for determining these parameters for fissionable deposits to  $\sim 1\%$  accuracy [8.35]. Furthermore, the calibrated fissionable deposits can be accurately and conveniently monitored thereafter using alpha counting techniques. In principle, it would be possible to calibrate a set of standard fissionable deposits for use in some integral measurements [7] as well as differential measurements. This procedure would be very desirable in development of an accurate cross-section data base for dosimetry applications.

#### B.4 Other Standards

Other reactions which are widely used as standards in other types of fast-neutron measurements are  $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ ,  $^{16}\text{Li}(n,\alpha)^3\text{H}$ ,  $^3\text{He}(n,p)^3\text{H}$  and  $^{10}\text{B}(n,\alpha)^7\text{Li}$ . These standards are not particularly useful for differential dosimetry reaction measurements,

so they will not be treated in this paper.

## VII. CALIBRATION OF GAMMA-RAY DETECTORS

This paper is concerned with the measurement of differential cross sections for threshold dosimetry reactions which produce gamma-ray active daughters. Careful calibration of the gamma-ray detectors used in these measurements is essential if high accuracy is to be achieved. The techniques used in calibration of gamma-ray detectors are widely known and present no special problems. They will be mentioned briefly in this paper.

There are few differences, in this regard, between integral and differential measurements. Activity measurements in differential studies are likely to be less accurate than their integral counter parts (excluding  $^{252}\text{Cf}$  measurements). The reasons are:

- i) The activity in samples irradiated at accelerator facilities is distributed less uniformly than for reactor irradiations because of the differences in the neutron environments.
- ii) The activity in samples irradiated at accelerator facilities is often considerably lower than is obtained in integral measurements. Statistics can limit accuracy.
- iii) Generally, larger samples are used in differential measurements than in integral measurements, so absorption corrections are larger.
- iv) The lower activities of samples irradiated at accelerator facilities, by comparison with integral measurements,

necessitates the placement of samples close to the detector and introduces uncertainties due to geometric and coincidence summing effects.

It seems likely that Ge(Li) detectors will eventually replace all other types of detectors for gamma-counting applications. Furthermore, mixed calibration standards are now available from centers such as the National Bureau of Standards (U.S.A.), the IAEA (Vienna, Austria) and the National Physical Laboratory (U.K.) which facilitate the task of calibrating these detectors. Assuming an accuracy of  $\sim 1-2\%$  in the standard reference materials used for calibration of gamma-ray detectors, it appears that a practical limit of attainable accuracy in calibration is  $\sim 2-3\%$  for differential measurements.

#### VIII. THE ROLE OF RADIOACTIVITY DATA IN DIFFERENTIAL CROSS SECTION MEASUREMENTS

Here, there are no differences between the requirements for integral and differential measurements. The status of the decay data for several radioactive species has been evaluated by Helmer and Greenwood [1]. Similar compilation and evaluation effort is also in progress elsewhere, e.g. the Nuclear Data Group at Oak Ridge National Laboratory and the Table of the Isotopes Project at Lawrence Radiation Laboratory (Berkeley) in the U.S.A.

The effects of uncertainty in radioactive-decay data vary from one reaction to another and must be analyzed in each separate case. This is beyond the scope of the present paper. However, it does not appear that this aspect will be a limiting

factor in development of a dosimetry cross section data set which is accurate to  $\sim 5\%$ .

#### IX. CORRECTIONS FOR GEOMETRIC EFFECTS AND THE ABSORPTION AND SCATTERING OF RADIATION

It is difficult to make any quantitative statements concerning the uncertainties in differential data which result from the application of corrections for geometric effects and the absorption and scattering of radiation. So much depends on the particular experimental details. What is clear is that these corrections need to be made carefully in order to avoid significant systematic errors which are very difficult to trace. This involves the application of adequate techniques (e.g. Monte Carlo analysis) and utilization of consistent supplementary nuclear data (e.g. neutron total and scattering cross sections). There is a great need for improved standards in reporting of quantitative data so that evaluators can honestly compare various data sets during the process of selecting the best possible values for reaction cross sections. There should be available a detailed description of the data analysis procedures and documentation of the origins of all supplementary nuclear data used in the analysis. Magnitudes of various estimated sources of error in the measurements and the method used to calculate the overall error should be reported. It would be useful if researchers could agree upon sets of supplementary nuclear data, which are readily available from data centers or widely circulated compilations, to use in analysis of their results.

If the corrections are properly treated, the uncertainties due to geometrical considerations can be negligible. It is possible to limit neutron absorption and scattering effects to  $\sim 10\%$ . These can be calculated with an accuracy of  $\sim 10\%$  which leads to an uncertainty of  $\sim 1\%$  in the cross section. Similarly, uncertainties in gamma-ray absorption can be minimized so that the net uncertainty in the cross section from this effect can be limited to  $< 1\%$ .

## X. CONCLUSIONS

At this point it is possible to draw some conclusions concerning the best accuracy which can be expected in the measurement of differential cross sections for threshold dosimetry reactions within the framework of present technology. Optimistically, it appears that the uncertainty cannot be expected to fall much below the 4-7% range (Table V) for measurements within the limits which can be computed using Eq. (3). The major source of uncertainty comes from neutron fluence determination. At present, certain standard fission cross sections appear to be more convenient to use in practice than the hydrogen standard for routine measurements. The cross sections obtained from measurements using fission standards appear to be as reliable as those measured using the hydrogen standard.

The prognosis is that some improvements in experimental techniques and in the knowledge of standard cross sections will be required to meet the 5% accuracy goal; however, the requirements are not excessively stringent measured against present capabilities

and the 5% accuracy goal is not unrealistic in this context.

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Table I

Approximate Sensitivity Limits in the  
Measurement of Differential Cross Sections  
for Several Dosimetry Reactions

Reaction	$t_{1/2}$	$t_E$	$b$	$f_C$	$\sigma_{\min}^a$	Current Status
$^{27}\text{Al}(n,p)^{27}\text{Mg}$	9.5 m	1-2 m	1	$\sim 2$	$\sim 2\mu\text{b}$	$\sim 0.1\text{ mb}^b$
$^{56}\text{Fe}(n,p)^{56}\text{Mn}$	2.58 h	$\sim 20$ m	1	1	$\sim 1\mu\text{b}$	$\sim 5\mu\text{b}^c$
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	15 h	$\sim 2$ h	1	1	$\sim 1\mu\text{b}$	$\sim 0.1\text{ mb}^b$
$^{48}\text{Ti}(n,p)^{48}\text{Sc}$	44 h	$\sim 7$ h	$\sim 1.5$	1	$\sim 1\mu\text{b}$	$\sim 30\mu\text{b}^c$
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	71 d	$\sim 1$ d	$\sim 1.5$	1	$\sim 10\mu\text{b}$	$\sim 3\mu\text{b}^c$
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	312 d	$\sim 1$ d	$\sim 20$	1	$\sim 0.5\text{ mb}$	$\sim 10\text{ mb}^b$
$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$	5.24 y	$\sim 1$ d	$\sim 1.5$	1	$\sim 0.3\text{ mb}$	$\sim 1\text{ mb}^b$

<sup>a</sup> Computed using Eq.(3) from the text.

<sup>b</sup> Ref. 5.

<sup>c</sup> Ref. 9 (special techniques were used to measure the lowest-energy points for the  $^{58}\text{Ni}$  reaction).

Table II

## Neutron Sources : Energetics

1. Li + p

Reaction	Q-Value (MeV)	Threshold (MeV)
${}^7\text{Li} + \text{p} \rightarrow \text{n} + {}^7\text{Be}$ (gnd. state)	-1.644	1.881
${}^7\text{Li} + \text{p} \rightarrow \text{n} + {}^7\text{Be}^*$ (1st state)	-2.079	2.380
${}^7\text{Li} + \text{p} \rightarrow \text{n} + {}^7\text{Be}^{**}$	-6.18	7.06
${}^7\text{Li} + \text{p} \rightarrow \text{n} + {}^3\text{He} + {}^4\text{He}$	-3.23	3.68
${}^6\text{Li} + \text{p} \rightarrow \text{n} + {}^6\text{Be}$	-5.07	5.92
${}^6\text{Li} + \text{p} \rightarrow \text{n} + \text{p} + {}^5\text{Li}$	-5.67	6.62

2. D + d

Reaction	Q-Value (MeV)	Threshold (MeV)
$\text{D} + \text{d} \rightarrow \text{n} + {}^3\text{He}$	+3.268	0
$\text{D} + \text{d} \rightarrow \text{n} + \text{p} + \text{D}$	-2.225	4.45
$\text{d} + \text{d} \rightarrow 2\text{n} + 2\text{p}$	-4.45	8.90

3. T + p

Reaction	Q-Value (MeV)	Threshold (MeV)
$\text{T} + \text{p} \rightarrow \text{n} + {}^3\text{He}$	-0.765	1.020
$\text{T} + \text{p} \rightarrow \text{n} + \text{p} + \text{D}$	-6.258	8.342
$\text{T} + \text{p} \rightarrow 2\text{n} + 2\text{p}$	-8.483	11.31

4. T + d

Reaction	Q-Value (MeV)	Threshold (MeV)
$\text{T} + \text{d} \rightarrow \text{n} + {}^4\text{He}$	+17.639	0
$\text{T} + \text{d} \rightarrow 2\text{n} + {}^3\text{He}$	-2.990	4.98
$\text{T} + \text{d} \rightarrow 2\text{n} + \text{p} + \text{D}$	-4.653	7.75
$\text{T} + \text{d} \rightarrow 3\text{n} + 2\text{p}$	-10.708	17.84

Table III

Comparison of Several 100-keV  
Thick Neutron-Source Targets

<u>Reaction</u>	<u>Charged- Particle Energy (MeV)</u>	<u>Maximum Neutron Energy (MeV)</u>	<u>Target</u>	<u>Zero-Degree<sup>a</sup> Reaction Cross Section (mb/sr)</u>	<u>Target Atoms per cm<sup>2</sup></u>	<u>Relative<sup>b</sup> Figure of Merit (no dim)</u>
${}^7\text{Li}(p,n){}^7\text{Be}$	2.3	0.6	Li-metal	150	$6.7 \times 10^{19}$	1
$\text{T}(p,n){}^3\text{He}$	3	2.3	$\text{T}_2$ Gas	120	$2.2 \times 10^{20}$	2.7
$\text{T}(p,n){}^3\text{He}$	3	2.3	Ti-T <sup>c</sup>	120	$2.2 \times 10^{19}$	0.3
$\text{D}(d,n){}^3\text{He}$	10	~13	$\text{D}_2$ Gas <sup>d</sup>	100	$3.1 \times 10^{20}$	3.1

<sup>a</sup> Cross section at energy of maximum yield derived from Fig. 2.

<sup>b</sup> Computed by multiplying the target atoms per cm<sup>2</sup> times the maximum zero-degree differential cross section.

<sup>c</sup> Tritiated-titanium metal.

<sup>d</sup> Deuterated-metal targets are only useful for relatively low-energy deuterons because of competition from (d,n) reactions in metal targets.

Table IV

Estimated Uncertainties in Current  
Knowledge of the Fast-Fission Cross  
Sections for  $^{235}\text{U}$ ,  $^{238}\text{U}$  and  $^{237}\text{Np}$

<u>Reaction</u>	<u>Energy Range</u>	<u>Percent Uncertainty</u>
$^{235}\text{U}(n,f)^a$	200-400 keV	5
	All other energies in range 25 keV- 20 MeV	3
$^{238}\text{U}(n,f)^b$	< 2 MeV	10
	2 - 6 MeV	4
	6 - 15 MeV	6
	> 15 MeV	10
$^{237}\text{Np}(n,f)^c$	0.51 - 1 MeV	10
	1 - 5.4 MeV	3
	5.4 - 14.1 MeV	10
	14.1	4
	> 14.1	10

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<sup>a</sup> Values derived from the report of a working group on absolute fission cross sections (Ref. 34). These uncertainties are considerably smaller than the assigned uncertainties in the ENDF/B-IV Dosimetry File (Ref.6).

<sup>b</sup> Values derived from the report of a working group on fission cross section ratios (Ref.34).

<sup>c</sup> Ref. 6.

Table V

Estimates for Major Sources of  
Error in Measurements  
of Differential Dosimetry Cross Sections

1. Neutron fluence determination	3 - 5%
2. Gamma-ray detector calibration	2 - 3%
3. Statistics	1 - 3%
4. Gamma-ray absorption	< 1%
5. Neutron absorption and multiple scattering	1%
6. Secondary neutrons	< 1%
7. Isotopic properties and decay data	Variable
Total error <sup>a</sup>	<hr/> >4 - 7%

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<sup>a</sup> Partial errors combined in quadrature to yield the total error.

FIGURE CAPTIONS

Fig. 1. Probable useful ranges of neutron-source reactions commonly used for monoenergetic cross section measurements in the range  $E_n = 0 - 20$  MeV.

(ANL Neg. No. 116-76-244)

Fig. 2. Zero-degree laboratory differential cross sections for neutron source reactions used for monoenergetic cross section measurements in the range  $E_n = 0 - 20$  MeV.

(ANL Neg. No. 116-76-245)

Fig. 3. Apparatus used for ratio measurements of activation cross sections relative to uranium fission (Ref. 13).

(ANL Neg. No. 116-1181)

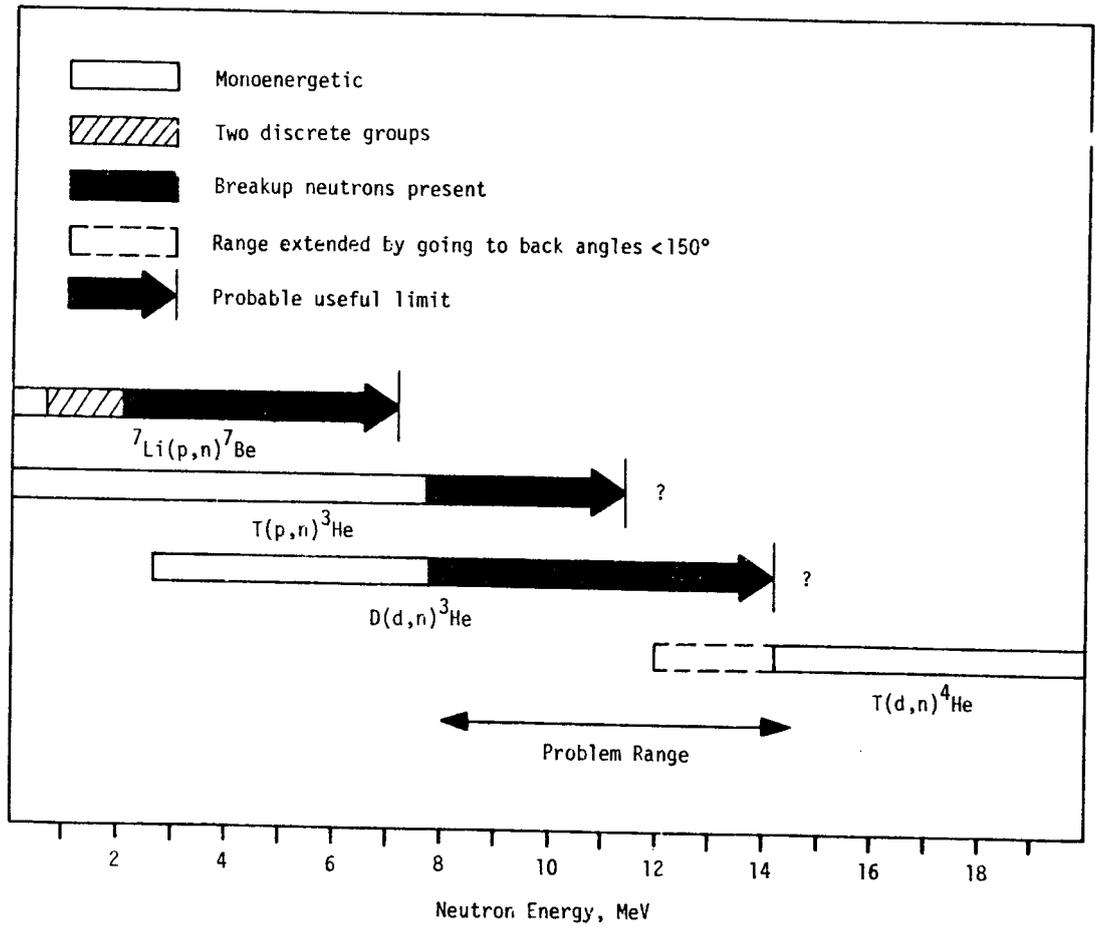


Figure 1

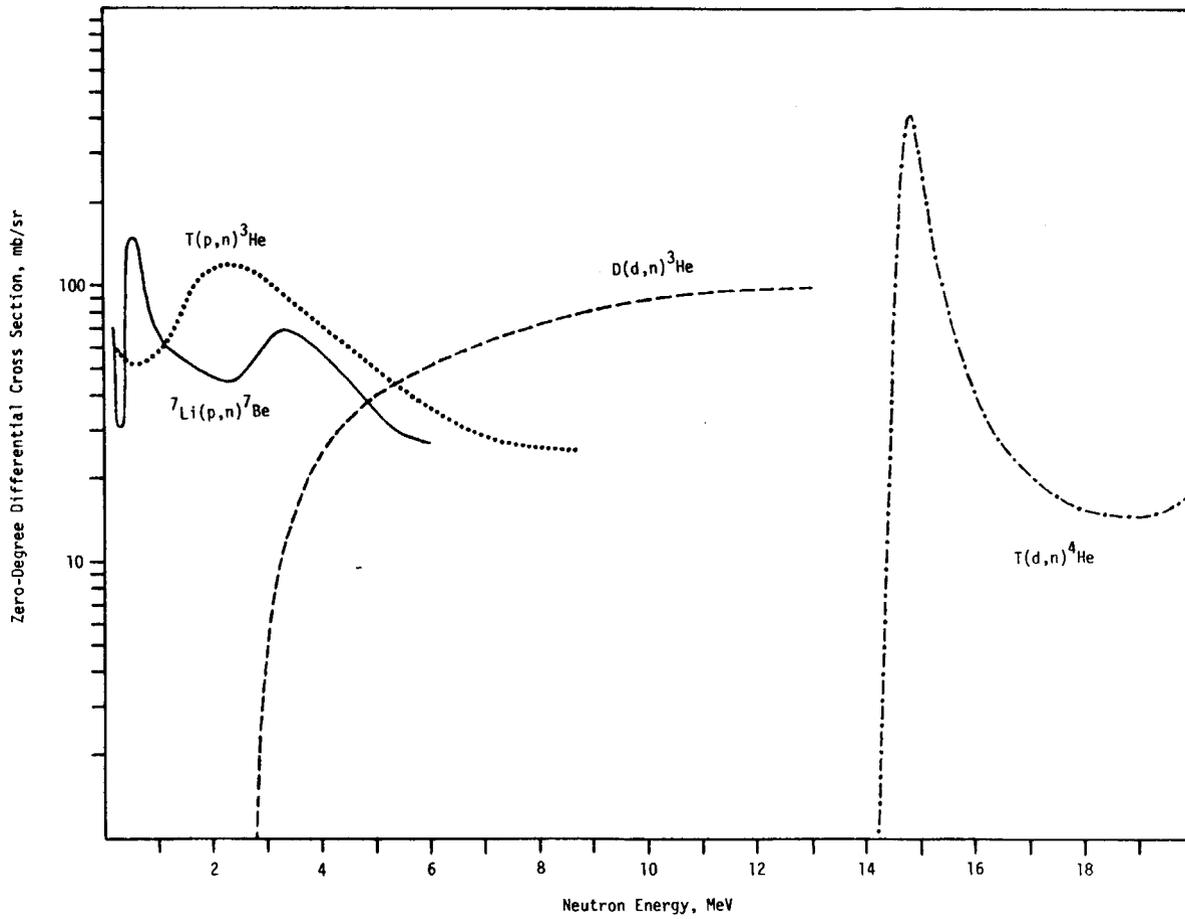


Figure 2

Figure 3

