Determination of the Neutron Energy Distribution
From 8 to 20 MeV in a Fast Flux Facility by
Using (n, 2n) Reactions as Threshold Monitors

Ronald Lee Gill

A. M. S. Thesis Submitted to Iowa State University, May 1974

Ames Laboratory, USAEC
Iowa State University
Ames, Iowa 50010

Date Transmitted: July 1974

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

MASTER

PREPARED FOR THE U. S. ATOMIC ENERGY COMMISSION
DIVISION OF RESEARCH UNDER CONTRACT NO. W-7405-eng-82

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Available from: National Technical Information Service
Department A
Springfield, VA 22151

Price: Microfiche **\$1.45**
Printed Copy: \$4.00

Determination of the neutron energy distribution
from 8 to 20 MeV in a fast flux facility by
using (n,2n) reactions as threshold monitors

by

Ronald Lee Gill

A Thesis Submitted to the
Graduate Faculty in Partial Fulfillment of
The Requirements for the Degree of
MASTER OF SCIENCE

Department: Chemistry
Major: Physical Chemistry

Approved:

A. F. Voigt
In Charge of Major Work

Robert E. McCaskey
For the Major Department

Martin J. Ulmer
For the Graduate College

Iowa State University
Ames, Iowa

1974

TABLE OF CONTENTS

	Page
INTRODUCTION	1
Statement of Problem	1
Literature Review	1
Threshold monitors	1
Unfolding techniques	3
Description of Methods	6
Threshold monitor reaction	6
Cross sections	7
Unfolding techniques used	8
EXPERIMENTAL PROCEDURES	11
Choice of Detector	11
Calibration of Detector	11
Threshold Monitors to be Used	13
Irradiation Conditions	16
Description of V3 facility	16
Short irradiations	16
Long irradiations	17
Counting Conditions	17
RESULTS	20
Experimental	20
Preliminary irradiations	20
Final irradiations	23
Calculated Results	23

LIST OF FIGURES

	Page
Figure 1. Final result from the unfolding procedure.	28

Cross section fits	23
Unfolding results	24
Comments	29
LITERATURE CITED	31
BIBLIOGRAPHY	32
APPENDIX: COMPUTER PROGRAM FOR UNFOLDING THE DATA	34

LIST OF TABLES

	Page
Table I. $(n,2n)$ threshold monitors initially chosen.	15
Table II. Results of the irradiations.	23
Table III. Coefficients of the polynomial equations used to calculate the $(n,2n)$ cross sections, in millibarns, of the thresh- old monitors.	25

INTRODUCTION

Statement of Problem

The knowledge of the neutron flux of a reactor is desirable in nearly all experiments involving irradiation of some material. In most systems, however, a priori knowledge of the flux is not, in practice, available and the experimenter must design his experiment in a manner that does not require knowledge of the flux. Nevertheless, it is useful and often necessary to have some knowledge of the energy distribution of the neutrons in the reactor spectrum, particularly at high energies where the flux can no longer be adequately described by a Maxwellian distribution.

The purpose, then, of this investigation is to develop a system of threshold monitors that will enable an experimenter to determine the energy distribution of a neutron flux at energies greater than 8 MeV, and to use this system to characterize the flux in a fast neutron facility.

Literature Review

Threshold monitors

In order for a nuclide to serve as a threshold monitor it must undergo a particular nuclear reaction above some minimum, or threshold energy. Below the threshold energy, the reaction should not occur. Ideally, the threshold should be well defined, the cross section for the nuclear reaction

should be free of resonances, and the cross section should have a well defined maximum energy, above which the reaction should not occur. Irradiation of such a nuclide would give a good measure of the integrated flux between the threshold and maximum energy.

In practice, however, the above criteria need not be rigorously met, and generally serve as guide lines for selecting the most suitable monitor. Provided that the cross section as a function of energy has been accurately determined, any nuclear reaction can serve as a monitor reaction.

The cross sections are typically expressed in one or more of four general methods.

1. Some experimenters use an average cross section determined by measuring the total activity induced by bombardment in a known flux [1,2]. The flux in turn, is calculated from the activity induced in a nuclide of "known" average cross section. In other words the average cross section is really a relative cross section, whose accuracy depends on how accurately the cross section of the standard is known. Using an average cross section has the advantages that the cross section is easily determined and the calculation of the flux becomes trivial. However, the results obtained with this method are less reliable than those calculated by other means.

2. Theoretical calculations of cross sections, particularly statistical model calculations [3], have been useful in pro-

viding information where experimental data are lacking.

3. Combining experimental and theoretical data to give semi-empirical cross sections has been especially useful where total cross sections are used as the monitoring reaction.

4. Where there are sufficient experimental data to accurately define the excitation function, an experimenter will generally use the empirical cross section versus energy curves as these usually will give the best evaluation of the experimentally determined flux.

Unfolding techniques

Methods for calculating the flux from experimental data vary widely according to the experimental procedures and the desired description of the flux. If an average flux is desired, the unfolding consists merely of noting that the measured activity is proportional to the product of the average flux and the average cross section, and determining the proportionality constant. In such cases, the average flux is quoted at the effective threshold for the reaction. The effective threshold is the energy at which it appears that all bombarding particles below this energy produce no reaction and all above this energy are equally effective in producing the reaction. Obviously, this energy must be greater than the real threshold, and it can be determined by the graphical method described by Hughes [4].

A mathematically simple, but experimentally cumbersome, procedure for determining the neutron energy distribution of the flux, is to use a monoenergetic neutron beam and determine the flux at a large number of different energies. Here again, the activity produced is proportional to the product of the cross section and the flux.

Unfolding the data is not usually mathematically trivial. In many cases exact solutions are not only difficult to obtain, but may be undesirable as well, since the data will have some non-negligible error associated with it. An acceptable solution, then, must not necessarily be unique, but must yield, upon proper mathematical treatment, the experimentally determined information. Careful consideration of the previous statement leads to another condition on the solution, namely, that the solution must satisfy all the relevant physical requirements of the system. Gold [5] has called such a solution an "appropriate" solution. For a neutron flux spectrum the most important physical requirements are that the flux be positive or zero at all energies and that it is not discontinuous.

Several methods of unfolding the data are available. If the flux may be adequately expressed by some function with undetermined parameters, a pattern search or a rotating coordinate search, as described by Wilde [6], may provide a powerful method for fitting the data. In these methods, a logi-

cal and systematic procedure for guessing the best parameters is defined. While these methods are powerful for unfolding data about a function of known form, they are rather inefficient for a point by point type of unfolding.

In point by point unfolding, the best value for the desired parameter is determined at each of many desired intervals, again subject to the constraint that the solution meets the relevant physical requirements. Using a large number of points will generally give a better result than would be obtained by using fewer points of evaluation. It is for this reason that the pattern and rotating coordinate searches are inefficient, as these methods would treat each of the points of evaluation as a parameter to a system of linear equations.

Methods suitable for unfolding the type of data obtained in neutron flux spectra measurements are described by Draper [7] and evaluated by Dierckx et al.[8]. Iterative techniques seem to be the most versatile and efficient for determining neutron spectra. The methods begin at an initially guessed flux and proceed to a final flux which hopefully will represent an appropriate solution. A wide variety of weighting methods and correction coefficients are used by different experimenters, with some success, and the methods actually used are predominantly a matter of personal preference. One of these iterative methods will be discussed later, in some detail.

Description of Methods

Threshold monitor reaction

Recently, Gove and Wapstra [9] compiled an extensive table of Q-values for various nuclear bombardment reactions. Careful study of this compilation showed a large number of potentially useful threshold monitors with Q-values between 7 MeV and 20 MeV, representing several different reactions.

Since the cross section versus energy data, or excitation functions, would need to be fit to some function, it was decided to use only one type of reaction, thus assuring that a consistent fitting method could be used throughout. If, for example, sharp resonances were not present in one type of reaction but were in another, different fitting techniques might be necessary in the region of the resonance peak, thereby complicating the excitation function fits.

In order to further insure the simple fitting of the excitation functions, only (n,2n) reactions were used as threshold monitoring reactions. The excitation functions for (n,2n) reactions are well behaved in the region of 8 MeV to 20 MeV. No sharp resonances for any nuclide were found in this range of neutron energies. Furthermore, the excitation functions for (n,2n) reactions rise quickly from the threshold. Other reactions rise much more slowly. Thus the (n,2n) reactions are the best approximation to a square pulse type function available, and certainly meet all the requirements

stated earlier for a good threshold monitor reaction.

Cross sections

The arbitrary upper energy limit of the flux was chosen as 20 MeV primarily on the basis of the available experimental excitation function data. Since the neutron energy distribution of the flux is the desired result, the energy dependence of the cross sections must be known experimentally. The best data, then, should be that obtained from bombardment by monoenergetic neutron beams. Literature references listed in CINDA 72 [10] indicate that such data is available to 20 MeV. In any cases where it is not available, an extrapolation to 20 MeV should be reliable due to the smooth behavior of $(n,2n)$ excitation functions directly below this energy.

It is possible to fit the excitation function data to theoretical equations. However, determining the best parameters for the equations often involves some rather complex fitting methods. One of the simplest fitting techniques, a least squares fit to a polynomial, may be used here. Naturally the entire excitation function cannot be fit to a single polynomial, but small segments can be easily fit to a polynomial. An excitation function, then, can be fit to a series of piecewise polynomials, provided some constraints are met.

The first, and most obvious constraint is that the calculated data agree with the experimental data, preferably to within a single standard deviation. Another, perhaps somewhat arbitrary constraint is that no segment should be fit to a polynomial of fourth degree or higher. Polynomials greater than third degree generally introduce oscillations that are not characteristic of the physical behavior of the systems being considered here. Lastly, the transition from one polynomial segment to the next should be smooth. More explicitly, the cross section calculated at the upper energy limit for one segment should equal the cross section calculated at the lower energy limit for the next segment, and the change of the slope, in making the transition, should not be abrupt. This requires that the series of polynomial segments be single valued and continuous, within experimental limits. This latter constraint is not so much a rigorous constraint as it is a subjective judgment, on the part of the experimenter, as to what is "single valued" and "continuous" enough.

Unfolding techniques used

The iterative unfolding technique employed was essentially the method described by Draper [11]. Suppose a system of threshold monitors, with a total of I monitors, is irradiated. The activity of the i 'th monitor is then given by

$$A_i = \int \sigma(E) \Phi(E) dE. \quad (1)$$

This may be approximated by the sum

$$A_i = \sum A_{ij} = \sum \sigma_{ij} \Phi_j \quad (2)$$

where the subscript j represents the energy intervals and A_{ij} is the partial activity of the i 'th monitor, produced by the j 'th energy group. The initial guess for the flux, the Φ_j 's, is either read in or calculated for each energy group. Next a weight function, w_{ij} , is defined as

$$w_{ij} = 0.5 (A_{ij} + A_{ij-1}) / A_i \quad (3)$$

and

$$w_{i1} = A_{i1} / A_i \quad (4)$$

for the i 'th monitor and the j 'th energy group. Now the ratio of the experimentally measured activity to the calculated activity for the i 'th monitor is

$$R_i = M_i / A_i. \quad (5)$$

Continuing, the procedure defines the logarithm of the geometric mean of ratios R_i with weights w_{ij} as the correction factor to the flux for the next iteration.

$$C_j = \frac{\sum_{i=1}^I w_{ij} \ln(R_i)}{\sum_{i=1}^I w_{ij}} \quad (6)$$

The next flux is computed according to

$$\Phi_j' = c_j \Phi_j. \quad (7)$$

The calculated values of the new flux are then substituted into Eq. (2) and the procedure is repeated until the value of

$$Q' = \frac{1}{2} \sum [(M_i - A_i) / M_i]^2 \quad (8)$$

has reached a prefixed limit or until a specified number of iterations has been performed.

Throughout this procedure the activity may be expressed in any manner desired. However, to avoid inconsistencies, the activity was expressed as the saturation activity in disintegrations per second per nucleus. If N atoms of a nuclide with a decay constant λ are irradiated for a length of time t , the saturation activity in disintegrations per second per nucleus is given by

$$A_S = A_m / [N(1 - e^{-\lambda t})] \quad (9)$$

where A_m is the disintegrations per second as measured at the end of the irradiation. This corrects for varying sample size and growth factors.

EXPERIMENTAL PROCEDURES

Choice of Detector

After deciding which reaction to use as the monitoring reaction, it was necessary to choose an appropriate detector. Since the product of an $(n,2n)$ reaction has the same atomic number as the target, no chemical separations would be possible. Therefore, the sample would be counted without separation of other reaction products, indicating the need for an energy sensitive detector. This leaves essentially two possible detectors: either NaI(Tl) scintillation detectors or Ge(Li) semiconductor detectors. The former has the advantage of higher efficiency than the latter; however, the resolution of NaI(Tl) detectors is inferior to that available with Ge(Li) detectors. Even though the production rate of the $(n,2n)$ product may be an order of magnitude or so less than other products, suggesting the desirability of the highest efficiency possible, the necessity of counting a particular gamma in the presence of many others, requires the use of Ge(Li) semiconductor detectors.

Calibration of Detector

In order to do absolute counting, the efficiency of the detector with respect to the gamma energy, in a suitable counting geometry, needed to be determined. Three IAEA calibrated gamma sources (^{57}Co , ^{22}Na and ^{137}Cs) were used to cal-

ibrate a ^{110}mAg and a ^{152}Eu source. These latter two sources could then be used to obtain easily a large number of calibration points in a desired geometry.

An assembly of ten shelves, each 2.0 cm apart, with the first shelf 2.0 cm from the face of the detector, was available for use in fixing the position of the sample relative to the detector. Since the standards were in the form of disks, it was decided to use this type of geometry for counting the samples.

The activity of the ^{110}mAg source was calculated on the basis of the 657.6 keV and 677.3 keV gammas, assuming that the efficiency for their detection was equal to that measured for the 661.6 keV gamma of the IAEA ^{137}Cs source. The activity was taken as the average of the activity obtained for each of the gammas. It was discovered that the ^{152}Eu source was contaminated with ^{154}Eu . This presented no problem except in those cases where both nuclides happen to have gammas of the same energy. In these instances, an adjusted gamma yield was calculated and the gamma, with the adjusted yield, was assumed to come from ^{152}Eu . The ^{152}Eu source was calibrated on the basis of the 656.5 keV gamma, relative to the ^{137}Cs efficiency. The ^{154}Eu source was calibrated on the basis of the 1274.34 keV gamma, relative to the ^{22}Na 1274.53 keV gamma efficiency. Using these two sources, the efficiency as a function of energy was determined for each shelf.

The ^{110m}Ag source was used on the shelves closer to the detector, since it was a weaker source than the ^{152}Eu and ^{154}Eu source.

Plotting the natural logarithm of the efficiency versus the natural logarithm of the energy, for each shelf, yielded a straight line, with equal slope for all ten shelves, from 120 keV to 1500 keV. The detector calibration, thus, was considered valid in this energy range only.

Threshold Monitors to be Used

In selecting the specific nuclides to serve as threshold monitors many conditions must be met. Since $(n,2n)$ reactions are being used, probably the first condition is that for a given atomic number, only the stable isotope with the lowest mass number can be used. If higher mass numbers are used, where they exist, then it will be impossible to distinguish between the $(n,2n)$ product of the higher mass number and the (n,γ) product of the lower mass number. It is equally important that the cross section and decay scheme data are available in the literature since the unfolding depends on knowing the cross section and the induced activity.

Other conditions, with an experimental origin, must also be considered. The abundance of the gamma to be counted must be large enough to provide good counting statistics. Many of the $(n,2n)$ products are positron emitters with only weak gam-

ma transitions, so although a generous amount of the product may be produced during an irradiation, the gamma radiation may not be observable. The annihilation radiation may be very intense, but since it is not unique there are only a few cases where it may be used for determining the activity of a particular nuclide.

As mentioned before, the activity of the $(n,2n)$ products may be an order of magnitude or so less than that for products of other reactions. For this reason isotopes with gammas of similar energies should be avoided. When there are such interferences present, the $(n,2n)$ activities will generally lose out to the more intense, undesired activities.

Consideration should also be given to the half lives of the nuclides produced, their isotopic abundances and the chemical forms in which the target nuclei are available. Some of the previous difficulties, particularly with respect to interfering activities, may be eliminated if the half life of the $(n,2n)$ product is long compared to that of the interferences. In such cases, the interferences can be allowed to decay away. The monitor should be available in chemical forms whose assays are known or easily determined. The chemical should not degrade significantly during irradiation and should be suitable for proper packaging for irradiation.

Discussing specific tolerances on the conditions mentioned above would serve no purpose since each experimental

arrangement will be different and have its own characteristic limitations. Thus, these conditions are to be adjusted, by trial and error, to fit the requirements and purposes of a specific project. Taking all the above conditions into consideration, eleven nuclides were tentatively chosen as threshold monitors, subject to their suitability to the experimental conditions. These monitors are listed in Table I.

Table I. (n,2n) threshold monitors initially chosen.

Monitor	-Q (MeV)	Half life of (n,2n) product	Gamma to be counted (KeV)	Gamma yield per 100 decays	Chemical form ¹
¹⁷⁵ Lu	7.659	156. day	994.0	43.0	Lu ₂ O ₃
¹⁶⁹ Tm	8.060	93.1 day	447.7	27.0	Tm ₂ O ₃
⁹³ Nb	8.826	10.1 day	934.9	99.0	Nb metal
¹³³ Cs	8.979	6.6 day	667.8	97.7	CsI ²
¹²⁷ I	9.139	12.8 day	386.3	34.0	HIO ₃
¹⁴¹ Pr	9.397	3.4 min	511.0	100.0	Pr ₆ O ₁₁
⁷⁵ As	10.243	18. day	596.0	58.0	As ₂ O ₃
¹⁹ F	10.431	110. min	511.0	194.0	teflon ³
⁸⁹ Y	11.468	107. day	898.2	91.8	Y ₂ O ₃
⁶⁴ Zn	11.856	38. min	670.1	11.2	Zn metal
⁵⁸ Ni	12.203	36. hrs	1377.4	79.2	Ni metal

¹The monitors were irradiated as powders except where indicated otherwise.

²CsI was not available for irradiation.

³Teflon was irradiated in the form of a thin sheet.

Irradiation Conditions

Description of V3 facility

All irradiations were performed in V3, a vertically oriented converter assembly, accessible from the top of the Ames Lab Research Reactor. There was no pneumatic transfer system directly to the experimental area, so the samples had to be transported manually. The elapsed time from end of irradiation to the beginning of data collection was from two to three minutes for samples that were to be counted as soon as possible. This precluded the use of any short-lived activities. The V3 facility has no circulating gas to remove heat from the sample. Because of this, sample containers irradiated for only a few minutes began to show observable thermal degradation.

Short irradiations

Because of the heating effects mentioned above, polyethylene rabbits could not be used for long irradiations. The rabbits themselves did not show heat damage unless the irradiation time was longer than ten minutes. However, the soft polyethylene vials containing the sample itself began to melt for irradiations longer than five minutes. Enclosing the sample in aluminum foil caused the styrofoam packing material to melt in the vicinity of the sample, presumably due to heat generated by reactions of the sample and of the foil. The decision reached was that for irradiations of duration less

than or equal to five minutes, the sample would be sealed in a soft polyethylene vial which was in turn placed in a polyethylene rabbit and packed with styrofoam. The use of polyethylene was desirable because it is relatively inexpensive and there is less activity induced in polyethylene than other commonly used irradiation containers, such as aluminum.

Long irradiations

Since some of the monitors initially chosen have half lives of many days, some long irradiations were necessary. These irradiations, ranging from half an hour to greater than one hour, were carried out using aluminum rabbits. The samples were wrapped in high purity aluminum foil. After irradiations of an hour, the activity induced in the aluminum rabbit was naturally very intense. Generally the activity induced in the container had decayed sufficiently after 24 hours to permit handling.

Counting Conditions

After a sample was irradiated it had to be prepared for counting. The sample had to be mounted in a manner such that the counting geometry was as close as possible to the detector calibration geometry, such that the sample could be easily weighed and such that there was no risk of sample loss and resulting contamination of equipment. Since the calibration standards were disks that were placed in aluminum plan-

chets for counting, the samples were placed in identical planchets that had been filled with paraffin. Prior to the irradiation, the paraffin had been melted and cooled. The planchet was then weighed. After the sample was irradiated, it was spread as uniformly as possible on the solidified paraffin. The planchet was heated and, as the paraffin remelted, the sample would sink, becoming covered with paraffin. When the paraffin cooled, the sample was completely sealed. Reweighing the planchet assembly gives the weight of the sample by difference. Preliminary tests on a group of blanks showed that there was no change of weight after repeated melting and cooling of the paraffin, provided that the temperature of the hot plate was not too high.

After mounting a sample, it was counted on a 20 cc Ge(Li) detector in conjunction with a Nuclear Data 50/50 system. Most of the monitors with long half lives were counted for an hour. The length of decay before counting varied from a few minutes to several weeks, depending on the half lives and magnitudes of the interferences present. Initially some attempts were made to calculate the optimum decay time and counting time for a particular monitor. However, the inability to predict accurately the amounts of each activity produced during a given irradiation made such calculations ineffective. Consequently, each monitor was counted for what seemed to be a reasonable length of time and that

time was adjusted, by trial and error, to improve the results if necessary. The best decay time was selected by repeatedly counting each monitor, over several half lives, until the standard deviation of the activity reached a minimum.

After collecting the data on the multichannel analyzer, the gamma spectra were analyzed by a computer program, ICPXGT, written in FORTRAN for an IBM 360/65 computer. The program analyzes the smoothed second derivative of the spectra to find Gaussian peaks. The position, height, width, area and energy of each peak is then determined. The program also has the capability of listing and plotting a spectrum.

RESULTS

Experimental

Preliminary irradiations

Before acquiring any final data, some preliminary irradiations were performed to determine experimentally the usefulness of the proposed monitors. The results of these preliminary irradiations are discussed briefly below.

1. Lutetium was irradiated as lutetium oxide. Although the isotopic abundance of ^{176}Lu is only 2.59%, the (n, γ) cross sections are large enough that a considerable amount of 161 day $^{177\text{m}}\text{Lu}$ was produced. The total gamma activity from $^{177\text{m}}\text{Lu}$ was so intense that none of the gammas from ^{174}Lu were observed. Because of this, lutetium had to be rejected as a threshold monitor. This made the minimum neutron energy to be investigated 8.060 MeV.
2. Thulium was irradiated as thulium oxide. The (n,2n) product, ^{168}Tm was easily observed in the presence of the other radionuclides. Thulium, therefore, was an acceptable monitor.
3. Niobium was irradiated as the metal. The activities of the products of most competing reactions were small due to their long half lives. ^{92}Nb was easily observed and hence niobium was an acceptable monitor.
4. Cesium was not irradiated because no supply of the desired compound was on hand. Because of the expense of cesium

compounds, cesium was rejected as a threshold monitor.

Rejecting cesium introduced no problems since the next nearest monitors had Q values within 0.1 MeV of that for cesium.

5. Iodine was irradiated as iodic acid. Because of the short half lives of interfering radionuclides, they could be allowed to decay away. Iodine was an acceptable monitor.

6. Praseodymium was irradiated as praseodymium oxide. Multichannel scaling experiments on the annihilation radiation showed that it consisted of many components, far more than could be estimated by present decay curve resolution techniques. As such, praseodymium was rejected as a monitor. While this left a gap of more than an MeV between Q values of successive monitors, the excitation functions overlapped sufficiently to avoid the appearance of a gap in the total cross section versus energy distribution.

7. Arsenic was irradiated as arsenic trioxide. A very large amount of ^{76}As was produced. The gamma intensity was so large that initially none of the $(n,2n)$ product could be observed. After the sample was allowed to cool for several weeks, the ^{74}As could easily be counted. Thus arsenic was an acceptable monitor.

8. Fluorine was irradiated as teflon. Within ten minutes after an irradiation, all of the components, except the desired ^{18}F had completely decayed away. Multichannel scaling

experiments on the annihilation radiation showed that it was a single component gamma with the proper 110 minute half life. The gamma spectra showed that there were some other minor components, probably due to impurities in the teflon, present in the sample. However, these minor components presented no difficulties. Therefore, fluorine was an acceptable threshold monitor.

9. Yttrium was irradiated as yttrium oxide. Because of the large amount of ^{90}Y produced it was necessary to cool the sample for several weeks before counting. After proper cooling, the ^{88}Y activity could easily be observed and determined. Yttrium, thus, was an acceptable monitor.

10. Zinc was irradiated as the metal. There was no difficulty in counting the ^{63}Zn gammas, so zinc was an acceptable monitor.

11. Nickel was also irradiated as the metal. It was necessary to cool the sample for several days before the ^{57}Ni could be accurately counted. Nickel was an acceptable monitor.

In summary, then, it was necessary to reject three of the eleven originally proposed monitors: ^{175}Lu , ^{133}Cs and ^{141}Pr . This left eight good monitors, making the system sensitive to neutrons above 8.060 MeV.

Final irradiations

The results of the preliminary irradiations made the determination of the activity of each threshold monitor a routine task. After suitable irradiation, decay and counting times, the saturation activity was calculated according to Eq. (9). The results of these irradiations are given in Table II.

Table II. Results of the irradiations.

Monitor	Irradiation time (min)	Decay time (min)	Counting time (min)	Saturation activity (dps per nucleus)
^{169}Tm	65.75	1936.0	60.00	$6.1805 \times 10^{-15} \pm 3.35\%$
^{93}Nb	65.75	1375.0	60.00	$8.8254 \times 10^{-16} \pm 2.87\%$
^{127}I	60.00	3017.0	60.00	$2.8852 \times 10^{-15} \pm 2.68\%$
^{75}As	65.75	20600.0	60.00	$4.1866 \times 10^{-16} \pm 2.83\%$
^{19}F	5.00	10.0	60.00	$1.8997 \times 10^{-17} \pm 1.08\%$
^{89}Y	65.75	11726.0	60.00	$3.8774 \times 10^{-16} \pm 3.73\%$
^{64}Zn	5.00	10.0	63.40	$1.5918 \times 10^{-17} \pm 6.80\%$
^{58}Ni	60.00	1287.0	60.00	$7.0432 \times 10^{-18} \pm 5.52\%$

Calculated Results

Cross section fits

Fitting the experimental excitation functions to piecewise polynomials was somewhat easier than expected as no more than six segments were needed to fit any one of the monitors

and agreements of better than one percent between the observed and calculated cross sections were easily obtained. Considering the errors involved in measuring cross sections, the agreements obtained are certainly much less than the experimental errors, even under the best of circumstances.

Table III lists all of the coefficients for each polynomial segment and the energy range over which the equation is valid, for each monitor. As an illustration of the use of Table III consider writing down the three equations necessary to express the cross section, in millibarns, for the reaction $^{169}\text{Tm}(n,2n)^{168}\text{Tm}$. The equations are:

$$\begin{aligned}\sigma(E) &= 3042.61 - 906.108xE + 65.5846xE^2 + 0xE^3 & 8.06 \leq E < 10.00 \text{ MeV} \\ \sigma(E) &= -9287.40 + 1641.86xE - 65.9090xE^2 + 0xE^3 & 10.00 \leq E < 12.20 \text{ MeV} \\ \sigma(E) &= -1181.07 + 291.178xE - 9.64286xE^2 + 0xE^3 & 12.20 \leq E \leq 20.00 \text{ MeV}.\end{aligned}$$

Unfolding results

The unfolding procedure described in Eqs. (1) to (8) was applied to the experimental data to obtain an appropriate energy distribution for the neutron flux. Two different fluxes, a $1/E$ distribution and a Watt spectrum, were tested for their usefulness as an initial guess for the neutron energy distribution.

Table III. Coefficients of the polynomial equations used to calculate the (n,2n) cross sections, in millibarns, of the threshold monitors.

	E ⁰ (MeV)	E ¹ (MeV)	E ² (MeV)	E ³ (MeV)	E min (MeV)	E max (MeV)
¹⁶⁹ Tm	3042.61	-906.108	65.5846	0.0	8.060	10.000
	-9287.40	1641.86	-65.9090	0.0	10.000	12.200
	-1181.07	291.178	-9.64286	0.0	12.200	20.000
⁹³ Nb	-153.763	17.4216	0.0	0.0	8.826	9.400
	-601.000	65.0001	0.0	0.0	9.400	9.600
	10330.9	-2205.15	117.857	0.0	9.600	10.400
	-5843.92	993.749	-40.1785	0.0	10.400	11.600
	-3613.02	736.458	-46.5930	1.03709	11.600	14.800
	-119.929	38.0357	0.0	0.0	14.800	20.000
¹²⁷ I	-396.486	43.3840	0.0	0.0	9.139	9.600
	-2140.26	225.052	-0.00268	0.0	9.600	10.000
	-517.687	59.9148	-17.2323	1.74620	10.000	12.850
	-130.611	1965.76	-67.1620	0.0	12.850	15.000
	-6397.06	1030.33	-34.4071	0.0	15.000	20.000
¹⁹ F	689.619	-178.200	14.4185	-0.35315	10.431	17.700
	71.9815	3.06436	-0.10077	0.0	17.700	20.000
⁷⁵ As	107902.	-27781.6	2353.32	-65.3520	10.600	13.250
	-13063.8	1881.59	-62.3106	0.0	13.250	14.900
	346.128	87.4873	-2.29296	0.0	14.900	20.000
⁸⁹ Y	-861.666	75.1366	0.0	0.0	11.468	12.200
	107637.	-24962.4	1903.53	-47.5584	12.200	13.750
	-19086.3	2477.66	-75.846	0.0	13.750	20.000
⁶⁴ Zn	14230.1	-3131.82	224.807	-5.21997	11.856	15.200
	-3992.48	472.752	-12.8095	0.0	15.200	17.300
	-922.517	120.932	-2.73137	0.0	17.300	20.000
⁵⁸ Ni	399.089	-67.3062	2.83554	0.0	12.203	12.950
	15357.6	-3383.19	247.097	-5.97703	12.950	14.400
	-837.189	105.675	-3.15530	0.0	14.400	15.700
	-118.321	14.6432	-0.27219	0.0	15.700	20.000

It was discovered that for a large number of iterations the results were essentially independent of the initial guess. However, for a small number of iterations the Watt spectrum gave results superior to that of the $1/E$ distribution, with respect to fitting the experimental data and producing a smooth flux. The Watt spectrum used as the initial guess was described by

$$\Phi(E) = 4.75 \times 10^6 \sinh(2E)^{1/2} e^{-E} \quad (10)$$

evaluated at each of the energy intervals.

The value of Q' , Eq. (8), reached a minimum of 0.38570 after nine iterations, with an average difference between the measured and calculated activities of the monitors of 59.747%. However, the flux obtained resembled the initial guess too closely. After 100 iterations, Q' increased to 0.391417, but the flux didn't resemble the initial guess. Although the flux was smooth in the region above 14 MeV, below this energy it showed some rather unrealistic oscillations. A smooth line through the oscillatory region was calculated and the resulting flux used as an initial guess for another unfolding trial. Using this initial guess, Q' reached a minimum of 0.385167 after six iterations, with an average difference of 59.684% between the measured and calculated activities of the monitors. Since the calculated activity was less than the measured activity, for each monitor, the flux was increased by 59.684% at each point of evaluation

and the resulting flux was used as the initial guess for still another unfolding trial. For this trial, Q' reached a minimum of 0.182553 after one iteration, with an average difference of 32.739% between the measured and calculated activities of the monitors. All other attempts to improve these results failed. Therefore, this flux, illustrated in Figure 1, was taken as the best result possible for the cross section data available and the system of monitors used.

The uncertainty in the flux is somewhat difficult to estimate. At any given energy, there may be as few as one or as many as eight monitors contributing to the total activity. The cross sections probably have uncertainties of from five to twenty percent for each contributing monitor. The uncertainty contributed from the total activity of the monitors ranges from a few percent to less than ten percent, but the contribution from just one particular neutron energy is not clear. If it is presumed that this latter contribution is negligible, a somewhat drastic assumption, it appears that the uncertainty contributed solely from the cross sections should be on the order of several tens of percent. So the 33% difference between the measured and calculated activities is perhaps not so bad and the precision of the resultant flux appears to be limited by the uncertainty of the cross sections.

Figure 1. Final result from the unfolding procedure.

Comments

The poor agreement between the measured and calculated activities raises some questions with respect to the validity of the results and methods. In the process of evaluating the data, it was noticed that altering the shape of the over-all cross section versus energy curve could drastically change the shape of the flux unfolded from the data. Alteration of other parameters, such as the initial guess for the flux or the activities, had relatively minor effects on the resultant flux. This is not so surprising since the energy distribution of the activities is not known and the flux is allowed to vary. The cross section versus energy distribution, then, is the only energy-dependent distribution which must both be known and not allowed to vary during the unfolding procedure. If this were also allowed to vary, the results would have no significance whatever.

The hyperbolic sine function, or Watt spectrum, used as an initial guess for the flux is probably not a realistic representation for the neutron energy distribution present in the fast flux facility used in this study. Watt's experiments were designed to eliminate neutron energy losses due to scattering so that a true spectrum of fission neutrons could be observed [12]. No such precautions were taken in this experiment. Qualitatively, the effect of this would be to increase the relative number of neutrons at lower energies at

the expense of higher energy neutrons. The gross structure of Figure 1 seems to indicate this sort of phenomena taking place.

In conclusion, then, the unfolding technique used appears to become less useful as the uncertainties in the cross sections increase. Although the uncertainties of $(n,2n)$ cross sections seem to be generally larger than for other reactions, the shape of the flux obtained is perhaps not totally unrealistic. As stated before, the solution obtained is certainly not a unique one. As such, the solution should not be taken as an absolute measurement of the fast neutron flux but rather it should be considered as a representation of the neutron energy distribution, consistent within the system of $(n,2n)$ monitors used.

LITERATURE CITED

1. J. P. Francois, R. Gijbels and J. Hoste, J. Inorg. Nucl. Chem. 35, 381 (1973)
2. J. P. Francois, R. Gijbels and J. Hoste, J. Inorg. Nucl. Chem. 35, 389 (1973)
3. S. D. Bloom, J. M. Green, H. W. Hubbard and S. W. Moszkowski, Nucl. Sci. Eng. 46, 255 (1971)
4. D. J. Hughes, Pile Neutron Research (Addison-Wesley Publishing Co., Inc., Cambridge, Mass., 1953) Chap. 4, p. 96
5. R. Gold, ANL-6984, (1964)
6. D. J. Wilde, Optimum Seeking Methods (Prentice-Hall, Inc., Englewood Cliffs, N.J., 1964) Chap. 5, pp. 123-193
7. E. L. Draper, Jr., Nucl. Sci. Eng. 46, 22 (1971)
8. R. Dierckx, M. L. Nimis, V. Sanguist and M. Terrani, Nucl. Instr. Methods 105, 1 (1972)
9. N. B. Gove and A. H. Wapstra, Nuclear Data Tables 11, 127 (1972)
10. CINDA 72, An Index to the Literature on Microscopic Neutron Data (The International Atomic Energy Agency, Vienna, 1972) Vols. 1 and 2
11. E. L. Draper, Jr., Nucl. Sci. Eng. 46, 22 (1971)
12. B. E. Watt, Phys. Rev. 87, 1032 (1952)

BIBLIOGRAPHY

- W. E. Alley and R. M. Lesser, UCRL-50484 Rev 1., (1972).
- S. D. Bloom, J. M. Green, H. W. Hubbard and S. W. Moszkowski, Nucl. Sci. Eng. 46, 255 (1971).
- BNL 325, 2nd Ed., Suppl. no. 2 (1964), Vols. 1, 2A, 2B, 2C.
- T. W. Bonner, R. A. Ferrell and M. C. Rinehart, Phys. Rev. 87, 1032 (1952).
- M. Borman, H. H. Bissem, E. Magiera and R. Warnemunde, Nucl. Phys. A157, 481 (1970).
- E. T. Bramlitt and R. W. Fink, Phys. Rev. 131, 2649 (1963).
- CINDA 72, An Index to the Literature on Microscopic Neutron Data IAEA, Vienna (1972).
- B. L. Cohen, Phys. Rev. 81, 184 (1951).
- L. F. Curtiss, Introduction to Neutron Physics (D. Van Nostrand, Princeton, New Jersey, 1959).
- R. Dierckx, M. L. Nimis, V. Sanguist and M. Terrani, Nucl. Instr. Methods 105, 1 (1972).
- E. L. Draper, Jr., Nucl. Sci. Eng. 46, 22 (1971).
- E. L. Draper, Jr., Nucl. Sci. Eng. 46, 31 (1971).
- P. A. Egelstaff and M. J. Poole, Experimental Neutron Thermalization (Pergamon Press, New York, 1969).
- A. Foderaro, The Elements of Neutron Interaction Theory (MIT Press, Cambridge, Mass., 1971).
- J. P. Francois, R. Gijbels and J. Hoste, J. Inorg. Nucl. Chem. 35, 381 (1973).
- J. P. Francois, R. Gijbels and J. Hoste, J. Inorg. Nucl. Chem. 35, 389 (1973).
- R. Gold, ANL-6984 (1964).
- N. B. Gove and A. W. Wapstra, Nucl. Data Tables 11, 127 (1972), Part V.

- A. K. Hankla, R. W. Fink and J. H. Hamilton, Nucl. Phys. A180, 157 (1972).
- D. L. Hill, Phys. Rev. 87, 1034 (1952).
- D. J. Hughes, Pile Neutron Research (Addison Wesley Publishing Co., Cambridge, Mass., 1953).
- C. M. Lederer, J. M. Hollander and I. Perlman, Table of the Isotopes (John Wiley and Sons, New York, 1967) 6th Ed.
- B. E. Watt, Phys. Rev. 87, 1032 (1952).
- D. J. Wilde, Optimum Seeking Methods (Prentice Hall Englewood Cliffs, New Jersey, 1964).

APPENDIX: COMPUTER PROGRAM FOR UNFOLDING THE DATA

A G level FORTRAN computer program, named RINKLE, was written to unfold the monitor activation data. The program is listed below with spaces between each subroutine for ease of inspection.

```

      IMPLICIT REAL*8 (A-H,O-Z)
      REAL*4 WHEN(6),G0(3),G1(3)
      DIMENSION NAME(3,8),XEQN(4,6,8),XLO(6,8),XHI(6,8),NSEG(8)
      COMMON/C1/NREAD,NPRINT,NPUNCH
      COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
      COMMON/C3/AI(8),AIJ(1000,8),WIJ(1000,8),RI(8),CJ(1000)
      DATA G0/'CALC','ULAT','ED '/,G1/'READ',' IN ',' '/',
1XEQN/160*0./
      CALL DATE(WHEN)
      READ(NREAD,1) MON,NPTS,NIT,IGUESS,ELO,EHI,QMIN
      1 FORMAT(4I5,3D10.3)

C
C VARIABLE      COLS      EXPLANATION
C MON           1 - 5      # OF FLUX MONITORS   MAX=8
C NPTS          6 - 10     # OF FLUX POINTS    MAX=1000
C NIT           11 - 15    # OF ITERATIONS
C IGUESS        16 - 20    .EQ. 0 => FIRST GUESS OF FLUX CALCULATED
C                .NE. 0 => NPTS READ IN FOR FIRST FLUX
C ELO           21 - 30    LOWER ENERGY LIMIT OF FLUX (MEV)
C EHI           31 - 40    HIGHER ENERGY LIMIT OF FLUX (MEV)
C QMIN          41 - 50    CONVERGENCE CRITERION
C
      DO 9 I=1,MON
      READ(NREAD,2) (SDBPAC(I),(NAME(J,I),J=1,3),NSEG(I))
      2 FORMAT(D20,13,3A4,3X,I5)

C
C SDBPAC(I) 1 - 20 SATDPS - BARN PER ATOM CM **2
C NAME      21 - 32 NAME OF MONITOR
C NSEG      36 - 40 # OF POLYNOMIAL SEGMENTS
C
      ISEG=NSEG(I)
      READ(NREAD,3) ((XEQN(J,K,I),J=1,4),XLO(K,I),XHI(K,I),
1K=1,ISEG)
      3 FORMAT(4D15.8,2D10.3)
      9 CONTINUE

C
C XEQN(1,K,I) 1 - 15 ENERGY ** 0

```

```

C XEQN(2,K,I) 16 - 30 ENERGY ** 1
C XEQN(3,K,I) 31 - 45 ENERGY ** 2
C XEQN(4,K,I) 46 - 60 ENERGY ** 3
C XLO(K,I) 61 - 70 LOWER ENERGY LIMIT FOR CROSS SECTION
C XHI(K,I) 71 - 80 HIGHER ENERGY LIMIT FOR CROSS SECTION

```

```

C
C
C
C

```

PRINT OUT THE HEADING AND THE DATA READ IN

```

WRITE(NPRINT,4) WHEN
4 FORMAT('1',58('*'),5X,'RINKLE',5X,58('*')//10('*'),
11X,'EXECUTION OF THE CROSS SECTION * FLUX UNFOLDING PROG'
2,'RAM (WRITTEN BY RONGILL) BEGAN AT: ',6A4,1X,9('*')//)
IF (IGUESS) 5,6,5
5 WRITE(NPRINT,7) MON,NIT,NPTS,G1,ELO,EHI,QMIN
GO TO 8
6 WRITE(NPRINT,7) MON,NIT,NPTS,G0,ELO,EHI,QMIN
7 FORMAT('0',T46,'NUMBER OF MONITORS',T77,'=',I6/T46,
1'MAXIMUM NUMBER OF ITERATIONS',T77,'=',I6/T46,
2'NUMBER OF FLUX POINTS',T77,'=',I6/T46,
3'INITIAL FLUX POINTS',T77,'=',3A4/T46,
3'LOWER ENERGY LIMIT OF FLUX',T77,'=',F7.3,' MEV'/T46,
4'HIGHER ENERGY LIMIT OF FLUX',T77,'=',F7.3,' MEV'/T46,
5'MINIMUM CONVERGENCE CRITERION',T77,'=',1PD11.4)

```

```

C
C
C

```

CHECK FOR ERRORS

```

8 NERROR=0
IF (NPTS .GT. 1000 .OR. NPTS .LE. 0) CALL LARGE(NERROR)
IF (ELO .GE. EHI) CALL LIMITS(NERROR)
DO 10 I=1,MON
ISEG=NSEG(I)
DO 10 J=1,ISEG
10 IF (XLO(J,I) .GE. XHI(J,I)) CALL LIMITS(NERROR)
IF (NERROR .GT. 0) STOP
DEL=(EHI-ELO)/DFLOAT(NPTS)
WRITE(NPRINT,11) DEL
11 FORMAT(' ',T46,'ENERGY INTERVAL (MEV)',T77,'=',1PD11.4//)

```

```

C
C
C
C

```

SET UP ENERGY INTERVALS, INITIAL FLUX GUESSES
AND CROSS SECTIONS

```

DO 12 I=1,NPTS
12 EN(I)=ELO+DEL*DFLOAT(I)
IF (IGUESS .NE. 0) CALL READIN(NPTS,&14)
DO 13 I=1,NPTS
13 FLUX(I)=4.75D 06*DSINH(DSQRT(2.*EN(I)))*DEXP(-EN(I))
14 DO 18 I=1,MON
QR=-XLO(1,I)
WRITE(NPRINT,15) I,(NAME(J,I),J=1,3),QR

```

```

15 FORMAT('0',T40,'FLUX MONITOR NUMBER',I2,' : ',3A4,
1' Q = ',F7.3,' MEV')
ISEG=NSEG(I)
WRITE(NPRINT,16)
16 FORMAT('0',T24,'EQN',5X,'EN ** 0',8X,'EN ** 1',8X,
1'EN ** 2',8X,'EN ** 3',6X,'LOW MEV',3X,'HIGH MEV')
DO 18 J=1,ISEG
WRITE(NPRINT,17) J, (XEQN(K,J,I),K=1,4),XLO(J,I),XHI(J,I)
17 FORMAT(' ',T24,I2,4(3X,1PD12.5),2(3X,0PF8.4))
18 CONTINUE
DO 20 I=1,MON
ISEG=NSEG(I)
KSEG=1
DO 19 J=1,NPTS
IF (XLO(1,I) .GT. EN(J)) GO TO 19
IF (XHI(ISEG,I) .LT. EN(J)) GO TO 20
IF (XHI(KSEG,I) .LT. EN(J)) KSEG=KSEG+1
CROSS(J,I)=XEQN(1,KSEG,I)+XEQN(2,KSEG,I)*EN(J)
1+XEQN(3,KSEG,I)*EN(J)**2+XEQN(4,KSEG,I)*EN(J)**3
IF (CROSS(J,I) .LT. 0.) CROSS(J,I)=0.
19 CONTINUE
20 CONTINUE

```

C
C
C

```

BEGIN THE UNFOLDING

DO 22 I=1,NIT
CALL RINKLE(NPTS,MON)
QP=0.
DO 21 J=1,MON
21 QP=QP+((SDBPAC(J)-AI(J))/SDBPAC(J))**2
QP=QP/DFLOAT(MON)
IF (QP .LE. QMIN) CALL SUCESS(I,NPTS,QP,MON,&26)
CALL CORECT(NPTS)
22 CONTINUE
WRITE(NPRINT,23) NIT,QP
23 FORMAT('0'////T16,'**** THE CONVERGENCE CRITERION WAS ',
1'NOT MET AFTER ',I5,' ITERATIONS. QPRIME =',1PD20.13,
2' ****')
CALL FLXLST(NPTS,MON)
CALL PUNPLT(NPTS)
26 STOP
END

```

BLOCK DATA

```

IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C1/NREAD,NPRINT,NPUNCH
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
DATA NREAD/5/,NPRINT/6/,NPUNCH/7/,CROSS/8000*0./
END

```

```

SUBROUTINE LARGE(NERROR)
COMMON/C1/NREAD,NPRINT,NPUNCH
NERROR=NERROR+1
WRITE(NPRINT,1)
1 FORMAT('0','***** ERROR ****',
1'SPECIFY 0<= NPTS <= 1000')
RETURN
END

```

```

SUBROUTINE LIMITS(NERROR)
COMMON/C1/NREAD,NPRINT,NPUNCH
NERROR=NERROR+1
WRITE(NPRINT,1)
1 FORMAT('0','***** ERROR ****',
1'SPECIFY LOWER ENERGY LIMIT < HIGHER ENERGY LIMIT')
1R ENERGY LIMIT')
RETURN
END

```

```

SUBROUTINE READIN(NPTS,*)
IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C1/NREAD,NPRINT,NPUNCH
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
READ(8,1) (FLUX(I),I=1,NPTS)
1 FORMAT(4D20.13)
RETURN 1
END

```

```

SUBROUTINE RINKLE(NPTS,MON)
IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
COMMON/C3/AI(8),AIJ(1000,8),WIJ(1000,8),RI(8),CJ(1000)
DO 2 I=1,MON
AI(I)=0.
DO 1 J=1,NPTS
AIJ(J,I)=CROSS(J,I)*FLUX(J)
1 AI(I)=AI(I)+AIJ(J,I)
2 RI(I)=SDBPAC(I)/AI(I)
DO 3 I=1,MON
WIJ(1,I)=AIJ(1,I)/AI(I)
DO 3 J=2,NPTS
3 WIJ(J,I)=0.5*(AIJ(J,I)+AIJ(J-1,I))/AI(I)
31 CONTINUE
DO 5 J=1,NPTS

```

```

T1=0.
T2=0.
DO 4 I=1,MON
T1=T1+WIJ(J,I)*DLOG(RI(I))
4 T2=T2+WIJ(J,I)
CJ(J)=T1/T2
IF (CJ(J) .LE. 0.) CJ(J)=1.
5 CONTINUE
RETURN
END

```

```

SUBROUTINE CORECT(NPTS)
IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
COMMON/C3/AI(8),AIJ(1000,8),WIJ(1000,8),RI(8),CJ(1000)
DO 1 I=1,NPTS
FLUX(I)=FLUX(I)*CJ(I)
IF (FLUX(I) .LT. 0.) FLUX(I)=0.
1 CONTINUE
RETURN
END

```

```

SUBROUTINE SUCESS(N,NPTS,QP,MON,*)
IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C1/NREAD,NPRINT,NPUNCH
WRITE(NPRINT,1) N,QP
1 FORMAT('0'////T21,'THE CONVERGENCE CRITERION HAS BEEN ',
1'MET AFTER ',I5,' ITERATIONS. QPRIME =',1PD20.13)
CALL FLXLST(NPTS,MON)
CALL PUNPLT(NPTS)
RETURN 1
END

```

```

SUBROUTINE PUNPLT(NPTS)
IMPLICIT REAL*8 (A-H,O-Z)
REAL*4 EX,FLUY
COMMON/C1/NREAD,NPRINT,NPUNCH
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
COMMON/C4/EX(1000),FLUY(1000)
WRITE(NPUNCH,1) (FLUX(I),I=1,NPTS)
1 FORMAT(1P4D20.13)
DO 2 I=1,NPTS
EX(I)=SNGL(EN(I))
FLUY(I)=SNGL(FLUX(I))
2

```

```

      IF (FLUY(I) .LT. 1.) FLUY(I)=1.
2  FLUY(I)=ALOG(FLUY(I))
      CALL GRAPH(NPTS,EX,FLUY,1,2,12.,-10.,0.,0.,-0.5,0.,
1 'NEUTRON ENERGY (MEV)', 'NEUTRON FLUX;',
2 'FAST NEUTRON FLUX;', 'N CM -2 SEC -1 ;')
      RETURN
      END

```

```

SUBROUTINE FLXLST(NPTS,MON)
IMPLICIT REAL*8 (A-H,O-Z)
COMMON/C1/NREAD,NPRINT,NPUNCH
COMMON/C2/SDBPAC(8),EN(1000),CROSS(1000,8),FLUX(1000)
COMMON/C3/AI(8),AIJ(1000,8),WIJ(1000,8),RI(8),CJ(1000)
WRITE(NPRINT,6)
6  FORMAT('0',T29,'MONITOR',T41,'MEASURED ACTIVITY',T63,
1 'CALCULATED ACTIVITY',T88,'PERCENT DIFFERENCE')
      DO 8 I=1,MON
      DIFF=(SDBPAC(I)-AI(I))/SDBPAC(I)
      DIFF=DIFF*100.
      WRITE(NPRINT,7) I,SDBPAC(I),AI(I),DIFF
7  FORMAT(' ',T31,0PI2,T42,1PD15.8,T65,D15.8,T88,D15.8)
8  CONTINUE
      DO 3 I=1,NPTS,118
      WRITE(NPRINT,1)
1  FORMAT('1',T8,2(10X,'INTERVAL ENERGY (MEV) FLUX N /',
1 ' CM**2 / SEC'))
      K=I+58
      IF (K .GT. NPTS) K=NPTS
      DO 3 J=I,K
      IND=J+59
      IF (IND .GT. NPTS) GO TO 4
      WRITE(NPRINT,2) J,EN(J),FLUX(J),IND,EN(IND),FLUX(IND)
2  FORMAT(' ',T20,0PI4,T30,F8.4,T44,1PD20.13,T76,0PI4,T86,
1 F8.4,T100,1PD20.13)
      GO TO 3
4  WRITE(NPRINT,5) J,EN(J),FLUX(J)
5  FORMAT(' ',T20,0PI4,T30,F8.4,T44,1PD20.13)
3  CONTINUE
      RETURN
      END

```