# A GRAY WEDGE EXPERIMENT

# FOR NEUTRONS

by

Gary R. Fleischmann

A Thesis Submitted to the Faculty of the

DEPARTMENT OF NUCLEAR ENGINEERING

In Partial Fulfillment of the Requirements
For the Degree of

MASTER OF SCIENCE

In the Graduate College

THE UNIVERSITY OF ARIZONA

1 9 7 0

## STATEMENT BY AUTHOR

This thesis has been submitted in partial fulfillment of requirements for an advanced degree at The University of Arizona and is deposited in the University Library to be made available to borrowers under rules of the Library.

Brief quotations from this essay are allowable without special permission, provided that accurate acknowledgment of source is made. Requests for permission for extended quotation from or reproduction of this manuscript in whole or in part may be granted by the head of the major department or the Dean of the Graduate College when in his judgment the proposed use of the material is in the interests of scholarship. In all other instances, including reproduction of the work of art, permission must be obtained from the author.

SIGNED: Gary R. Fleischman

## APPROVAL BY THESIS DIRECTOR

This thesis has been approved on the date shown below:

George W. Nelson — May 8, 1970

George W. Nelson
Professor of Nuclear Engineering

Date

ACKNOWLEDGMENTS

The author would like to express his gratitude to Dr. George Nelson for his help and advice throughout this work.

In addition the author is grateful to the Department of Nuclear Engineering for providing computer time and materials for this project.

The assistance offered by several fellow students as reactor operators and assistant reactor operators is also greatly appreciated.

I would like to express special thanks to my wife for her encouragement and patience during this work.

TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS--Continued

## ABSTRACT

A method is described for analyzing the energy spectrum of a beam of neutrons by passing them through a 1/v absorbing material whose thickness varies linearly with time. The attenuation data is shown to be related to the Laplace transform of the energy dependent flux. Taking the inverse Laplace transform of a curve fitted to the data gives an analytical expression for $\phi(E)$. The main problem found with the method is that the resultant spectrum is highly dependent upon the type of function used to fit the data and the more appropriate fitting functions do not lend themselves to curve fitting by a least squares computer program.

If this problem can be solved the method might be able to measure neutron spectra with fair resolution using compact, simple equipment. Isotope cross sections might also be measured over a wide range of energy.

# CHAPTER I

## INTRODUCTION

The topic of this thesis is an integral type neutron spectrometer, i.e., a spectrometer which measures an integral property of a neutron spectrum. This is generally accomplished by passing the neutrons through a material which absorbs nearly all neutrons below a characteristic energy (e.g., 0.4 ev for 0.020" thick cadmium foil). The output of a device of this type is a count rate (neutron speed dependent) equal to the integral (with respect to speed) of the counter efficiency and transmitted neutron flux, hence the name integral spectrometer.

The other major type of spectrometer operates on the principle of separating and measuring a group of neutrons in a narrow energy range from the neutron population, and is referred to as a differential type. The most widely used device of this type is the time-of-flight spectrometer which works on the principle that a burst of neutrons (of various energies) starting from the same point in space will reach a second location at a time which is characteristic of their speed. A common means of getting a pulse, or a series of pulses, of neutrons for this type of measurement is by the use of a "chopper" which may be a cylinder of a highly neutron absorbent material with a slit passing through it. When the cylinder is rotated in a neutron beam, very few neutrons pass through unless the slit is parallel to the beam direction. At this time a pulse of neutrons passes through. The neutrons passing through this chopper may either be counted as a function of time after the pulse, thus giving an indication of the energy distribution of the beam, or they may be passed through a second "chopper" at some distance along the beam. In the second case, the two choppers are phased such that only neutrons in a narrow energy band travel the distance in the allotted time. This type of system acts as a neutron "monochromator".

Another type of a differential spectrometer uses a crystal lattice as a diffraction grating. Very low energy neutrons will diffract from suitable crystals according to their relativistic wave-length in such a manner as to cause constructive interference under the right conditions. This phenomena is analogous to the scattering of electromagnetic radiation from crystals (x-ray diffraction), and is similarly governed by Bragg's law. Thus a given lattice spacing and angle of incidence will yield a particular wave length (energy) neutron beam which has undergone constructive interference. This method probably gives the best separation of neutrons by energy for very low energy neutrons. Higher energy neutrons are not defracted from the surface of the crystal but are transmitted through the surface thus limiting the energy range of this method.

The integral type spectrometer was the first of the two types to be utilized to measure neutron energy spectra. Attenuators had been used to study the energy of other particles and photons and it was quite natural to apply the same methods of analysis to neutrons when they were discovered. There are several naturally occurring isotopes which have a large cross section for low energy neutrons but which do not interact with faster neutrons to any appreciable extent. Some of these isotopes absorb neutrons in such a manner that the absorption cross section varies inversely with the neutron speed. This type of absorber is referred to as a 1/v attenuator or absorber.

Much of the early cross section work was done with 1/v type attenuators. The $^{10}B$ isotope in elemental boron was probably the most widely used in these attenuators which were fabricated as disks from a mixture of lead and boron. By varying their thickness one could discriminate or remove neutrons of energies less than a selected "cut-off" energy. Fermi, Anderson, and Marshall (1946) and Lichtenberger (1945) used disks of elemental boron and lead while Dancoff (1948) used enriched boron (92% $^{10}B$) in some of his work.

Dancoff's paper was of special interest in that he attempted to estimate a single resonance for several isotopes whose cross sections varied from a strictly 1/v behavior. His method involved the assumption that the spectrum of a beam taken from a heavy water reactor is given by:

$$\phi(E) \propto 1/E \qquad (1)$$

above some cut-off energy.

This assumption is plausable if the low energy portion of the spectrum (Maxwellian) is removed by filtering the beam through cadmium and if there is not a large component of unscattered fission neutrons. The latter condition is satisfied by not placing fuel in line with the beam, i.e., requiring the neutrons to scatter several times before they enter the beam. The gamma activity of a pure 1/v absorbing foil in the beam also containing boron was calculated as a function of boron thickness. Then, by measuring the induced activity of selected isotopes, as a function of boron thickness, and subtracting the amount expected if the "detector" foil were pure 1/v, the amount due to "monokinetic" effects was obtained. This difference followed approximately a logarithmic relationship. In order to simplify the analysis it was assumed that a semi-log plot would be fit by a straight line and from its slope the energy of the resonance was calculated.

The values arrived at by Dancoff are generally quite close to the values given in BLN 325 by Stehn, Goldberg, Magurno, and Wiener-Chasman (1964) for the lowest energy resonance of the isotopes. The scattering cross sections of both boron isotopes were not well known at that time, and there were serious questions about the enrichment of the $^{10}B$ used in these experiments.

Dancoff stressed that his results were not rigorous. A number of curves could have been plotted through the points corresponding to the difference between the foil activities and those for a pure 1/v absorbing foil. He also stated that details of the cross section versus energy curve could not be brought out with this type of experiment because monoenergetic neutrons were not used. The amount of detail measurable by this method remains to be determined.

As investigations in measurements of cross sections increased and more precise values were needed, the differential spectrometers were developed and have nearly replaced the earlier integral type. The integral type are still used in some dosimeters and in experiments where it is desirable to divide the neutron population into a small number of energy groups. One of the more recent applications of the integral type device found in the literature is a dosimeter made at Oak Ridge National Laboratory. The attenuators for this dosimeter consist of twenty-four disks made by pressing enriched boron powder (92% $^{10}B$) into thin walled aluminum containers. These disks ranged in thickness from 0.06 to 6.1 cm. and fit into a pear shaped counter which accepts neutrons traveling only in the axial direction and passing through the filters as shown by Blosser (1964). The density of the compacted boron powder was 1.4 g/cc and the compaction pressure was 30,000 psi.

The spectrometer developed here is an integral type in which the attenuator thickness varies in a more continuous manner than in disk type attenuation spectrometers. This is accomplished by uniformly and continuously varying the thickness of a 1/v absorber which attenuates a beam of neutrons. The neutrons which pass through the attenuator are detected and the counts stored in a multi-channel analyser such that they can be correlated with the thickness of the attenuator at the time they were being counted. The details of the experiment are discussed later (Chapter III). A basic question motivating this experiment is to determine whether or not additional information obtained would be useful in making more precise measurements of a neutron spectrum. This method of a continuously varying attenuator has been used to determine a fast reactor spectrum over its entire range as discussed by Kuzminov (1962). It is not known whether the details of a limited segment of an energy spectrum may be brought out by this method. If so, this type of device might also be used to make cross section measurements.

There is a need for a spectrometer capable of cross section measurements in the 10 kev to 1 Mev range. The mechanical chopper techniques generally lose their resolution in the range of 1 to 10 kev due to (1) physical limitations on the speed of rotation of the chopper drum, (2) length of the flight path, and (3) the fineness of time intervals in which neutrons are counted. The higher energy ranges (above 1 Mev) can be studied using an ion accelerator and suitable targets giving bursts of monoenergetic neutrons. Varying the target material, accelerator potential, and angle of measurement (with respect to the incident beam) gives a range of neutron energies.

## CHAPTER II

## THEORY

The mathematical description of this varying thickness, neutron spectrum analysis method is straightforward. The primary assumptions made are that the variation of count rate (counts per channel) with wedge thickness (channel numbers) can be described by a Laplace transform and that the attenuating material has a cross section which varies as the inverse of the neutron speed.

The starting point for the derivation is the equation for the count rate [C(x)] of a detector in a neutron beam that has passed through an attenuator of thickness <u>x</u>:

$$C(x) = \int_0^\infty A\ \varepsilon(v)\phi(v)e^{-\Sigma x}dv \qquad (2)$$

where:

$A$ = beam area

$v$ = neutron speed

$\varepsilon(v)$ = speed dependent counter efficiency

$\phi(v)$ = speed depencent neutron flux.

$\Sigma$ = interaction cross section for the attenuator material

In order to see that the above integral is a Laplace transform it is necessary to make a change of variables. Kuzminov suggested writing dv in terms of $\Sigma$. It is more convenient here to express $\Sigma$ in the equation in terms of 1/v since we have chosen an attenuating material with:

$$\Sigma = a/v + b \tag{3a}$$

where a and b are constants. We then let:

$$u = 1/v, \tag{3b}$$

$$\Sigma = au + b \tag{3c}$$

and

$$dv = -du/u^2. \tag{3d}$$

By making these substitutions the count rate equation becomes:

$$C(x) = \int_0^\infty A \quad \varepsilon(v)\phi(v)e^{-bx}\ e^{-axu}\ u^2 du \tag{4a}$$

letting ax = s:

$$C(x) = Ae^{-bx} \int_0^\infty \varepsilon(v)\phi(v)u^{-2}e^{-su}\ du \tag{4b}$$

where the above integral is recognized as the Laplace transform of $\varepsilon(v)\phi(v)u^{-2}$.

At this point it is necessary to introduce corrections for the experimental restrictions that: (1) the thickness of the attenuator varies across the width of the beam by ($\Delta x_1$) because the beam width is not negligible; (2) the thickness of the attenuator changes during the counting interval by an amount $\Delta x_2$. These corrections are made by integrating the instantaneous count rate over the range from x to x + $\Delta$x in both cases.

Thus taking C(x) from equation (2):

$$C(n) = \int_{x}^{x + \Delta x_2} \int_{x}^{x + \Delta x_1} \frac{C(x)\,dxdx}{\Delta x_1 \Delta x_2} \tag{5a}$$

where C(n) = counts/channel as a function of channel number.

Changing the order of integration gives:

$$C(n) = \int_{o}^{\infty} \int_{x}^{x + \Delta x_2} \int_{x}^{x + \Delta x_1} \frac{A \;\; \varepsilon(v)\phi(v)e^{-\Sigma x} \; dxdx \; dv}{\Delta x_1 \Delta x_2} \tag{5b}$$

$$= \int_{o}^{\infty} \int_{x}^{x + \Delta x_2} A \;\; \varepsilon(v)\phi(v)e^{-\Sigma x} \; \frac{(1-e^{-\Sigma \Delta x_1})}{\Delta x_1 \Delta x_2 \; \Sigma} \, dx \; dv \tag{5c}$$

$$= \int_{o}^{\infty} A \;\; \varepsilon(v)\phi(v)e^{-\Sigma\; x} \; \frac{(1-e^{-\Sigma \Delta x_1})(1-e^{-\Sigma \Delta x_2})}{\Delta x_1 \Delta x_2 \Sigma^2} \, dv \tag{5d}$$

Making the change of variables in equation (5d) as done to obtain equation (4b) gives:

$$C(n) = Ae^{-bx} \int_{o}^{\infty} G(u)e^{-su} \; du \tag{6a}$$

$$\text{where } G(u) = \varepsilon(u)\phi(u)u^{-2} \; \frac{(1-e^{-\Sigma \Delta x_1})(1-e^{-\Sigma \Delta x_2})}{\Delta x_1 \Delta x_2 \; \Sigma^2} \; . \tag{6b}$$

In the apparatus used for this experiment $\Delta x_1 >> \Delta x_2$ and the correction term was taken to be $\frac{1-e^{-\Sigma \Delta x_1}}{\Sigma \Delta x_1}$.

At this point we use the assumption that the inverse Laplace transform of the equation for the variation of C(n) with channel number (time) exists and take the inverse transform:

$$L^{-1} \frac{[C(n)e^{bx}]}{A} = L^{-1} \; L[G(u)] \tag{7a}$$

$$= G(u) = F(v) \tag{7b}$$

$$= v^2 \; \varepsilon(v)\phi(v) \; \frac{(1-e^{-\Sigma\Delta x_1})}{\Delta \, x_1 \; \Sigma} \tag{7c}$$

That is to say:

$$\phi(E) = \frac{\phi(v)}{v} = \frac{L^{-1} \, (C_n \; e^{bx})\Sigma\Delta x_1}{Av^3 \; \varepsilon(v)(1-e^{-\Sigma\Delta x_1})} \tag{8}$$

The solution of this equation is discussed later (Chapter IV).

An equation for the interaction cross section of a foil placed in the beam can be derived as follows:

$$\phi(v)_f = e^{-\Sigma_f d} \; \phi_o(v) \tag{9a}$$

or

$$\Sigma_f = (1/d)\ln \; \phi_o(v)/\phi_f(v) \tag{9b}$$

where $d$ = thickness of the foil

$\Sigma_f$ = elimination cross section of the foil

$\phi(v)_o$ = speed dependent flux without the foil in the beam

$\phi(v)_f$ = speed dependent flux with the foil in the beam

With proper modification of the experiment, the interaction cross section can be broken down into its components, i.e., absorption and large angle scattering.

Note that equations (8), and (9b) combine to give:

$$\Sigma_f = 1/d \; \ln \frac{L^{-1} \; [C(n) \; e^{bx}]_o}{L^{-1} \; [C(n) \; e^{bx}]_f} \tag{10}$$

## CHAPTER III

## EXPERIMENTAL

### Apparatus

The required experimental measurements involve passing a wedge of 1/v attenuator through a beam of neutrons and recording neutron transmission as of function of wedge thickness. The beam of neutrons was obtained from a General Atomic TRIGA MARK I reactor as shown in the instruction manual by Mikesell(1966) by means of a 33-foot long aluminum beam tube. The tube was originally constructed for experiments using a "slow chopper" as shown by Cooper (1963). In the early trials of the experiment the beam passed through the chopper box which rested above the reactor core. The chopper box was later eliminated by placing a short extension onto the main beam tube. This tube then extended down into the center of the core such that it was in the region of highest neutron flux. The neutron flux obtainable at the upper end of the beam tube was approximately $10^7$ neutrons/cm$^2$ - sec. at 100 KW. However, due to multichannel analyser dead time, a beam of less than $1.5 \times 10^4$ n/cm$^2$ sec. was used here. The tube was partially evacuated by a roughing pump to dimish neutron scattering by air. The pressure in the tube was reduced to approximately one-tenth of atmospheric, thereby reducing the scattering of neutrons out of the beam from about 38% to 5%.

A collimator was placed in the beam tube at the point where it widens to 3" diameter (Fig. 1), about three feet below the water level of the reactor. This consisted of a steel cylinder 1.6" I.D., 2.8" O.D., and 4" long. A similar cylinder of borated paraffin rested on the steel collimator.

The beam tube terminated in the center of a 50 gallon drum which was filled with a borax soap and water solution and modified to accommodate the tube, the gray wedge, and its driving mechanism, (Figs. 2 and 3).

The fabrication of a suitable wedge grew into a rather time-consuming part of the experiment. The original idea was to use a circular shaped wedge (helix). An aluminum shell was fabricated in the shape of a helix; however, several attempts to pack it with a uniformly dense filling of $B_4C$ powder failed. The second attempt was a straight wedge, also an aluminum shell with $B_4C$ packed into it. This was vibrated during packing and approximately 100 lbs. of force were applied to the powder through the small side of the wedge. The wedge produced in this manner gave the first analyzable data for the experiment; however, when an attempt was made to cycle the wedge more rapidly, the powder shifted in the aluminum shell, disturbing the uniform packing.

The final wedge was made by sintering a three-to-one mixture of powdered aluminum and $B_4C$ powder. The mixture was first compressed using about 18,000 lbs/sq. in. yielding a slab (1.7 cm x 4.6 cm x 7.6cm) which was then fired in a tube furnace at $570^{\circ}$ C for 90 minutes.

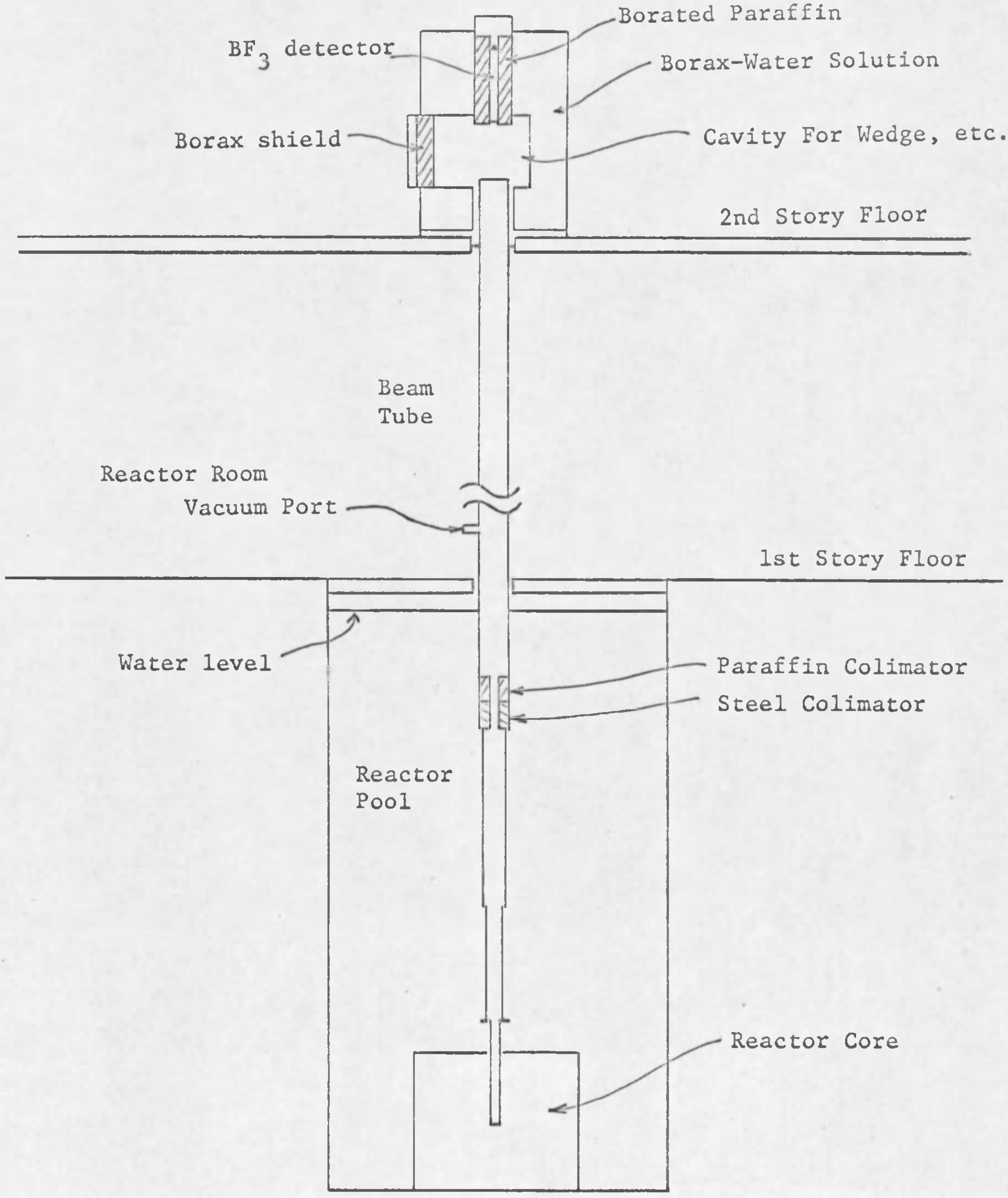


Figure 1. Beam tube in place (Not to scale)

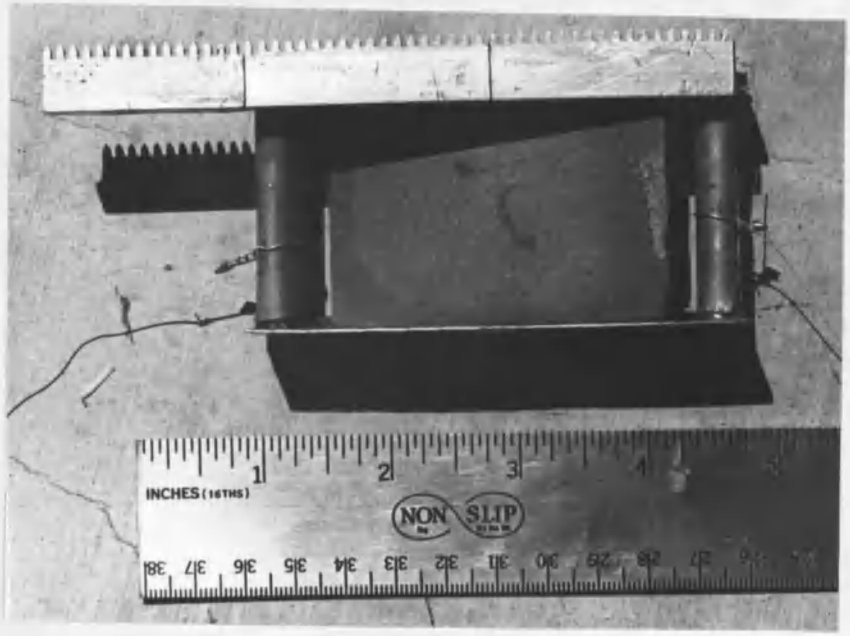


Figure 2. Wedge of sintered aluminum and boron carbide.

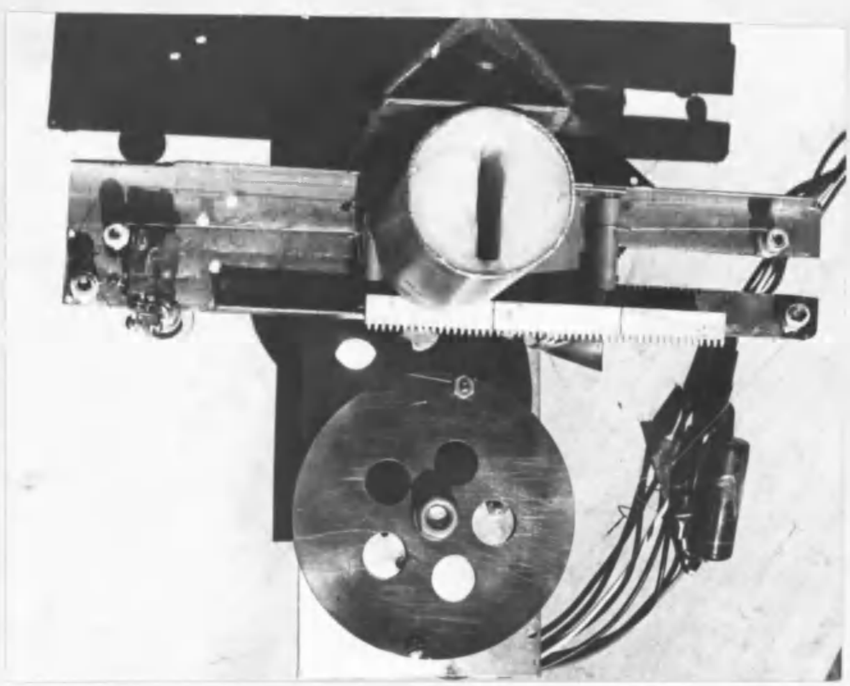

Figure 3. Wedge, associated drive mechanism, and colimator.

This wedge was not completely satisfactory in that the material density (1.3 g/cc) was approximately half of the theoretical value (~2.5 g/cc). Thus it was a weak material. Several different ratios of $B_4C$ to aluminum had been tried using a 1" diameter cylindrical die. A pressure of 10,000 psi yielded a very dense machinable disk with the three-to-one mixture; however, the product of the larger die was less satisfactory. The ram of the larger die did not fit precisely (~10 mil gap). This allowed the die to leak and bind, the $B_4C$ - aluminum being an abrasive, gummy mixture to press. The die was badly galled while the fourth slab was being pressed. Secondly, increasing the thickness of the sample allowed the pressure to distribute in a less uniform manner and thus contributed to a less dense slab. In addition there was a change in coloration from the center of the slab to the outer edges suggesting segregation during sintering or possibly reaction of the mixture with the nitrogen gas. It is felt that using hot press techniques in an evacuated atmosphere would give a much denser and perhaps more uniform slab, but lack of equipment prevented application of this technique here.

The drive for the wedge was a small 60 rpm synchronous motor which was geared down to 1/2 rpm. In the final experiment a 4" diameter wheel was used to wind a nylon cord attached to the wedge (Fig. 2).

The remainder of the equipment for this experiment includes a $BF_3$ neutron detector measuring 1" diameter x 10" long(40 cm pressure), and a Technical Measurement Corporation 1024 channel analyser (Model

CN-1024 Digital Computer Unit) with a multiscaler logic unit. Fig. 4 shows the counting equipment diagrammatically.

## Procedure

The apparatus was placed (Fig. 1) and the tube evacuated to .1 atmosphere. The reactor was checked out, brought to critical, and the power level increased until the desired count rate was obtained with the $BF_3$ counter with no wedge present. The neutron and gamma dose rates were checked, both in the reactor room and at the position of the experimenter near the end of the beam tube, at 10 watts and also at the desired operating level (1-3 kw).

After the reactor power level had stabilized, the drive mechanism was activated and the wedge passed through the beam. As the wedge started forward it tripped a switch starting the multiscaler logic unit, storing counts in successive channels of the 1024 channel analyser. The time per channel was .1 sec. for the later runs and the wedge speed was .27 cm/sec. The motor driving the wedge was stopped and reversed after 38 and 45 seconds respectively and counts were taken as the wedge returned to its original position. This cycle was repeated ten times for each spectrum. The foils of Cd, In, and $B_4C$-Al were placed on the front of the detector tube for their cross section measurements.

Preparation for the experiment included finding the detector plateau and setting the amplifier such that no gamma rays were counted. The amplifier gain and discriminator were set so that a gamma ray would have to lose 16 times as much energy in the detector as $Co^{60}$ radiation to be counted.

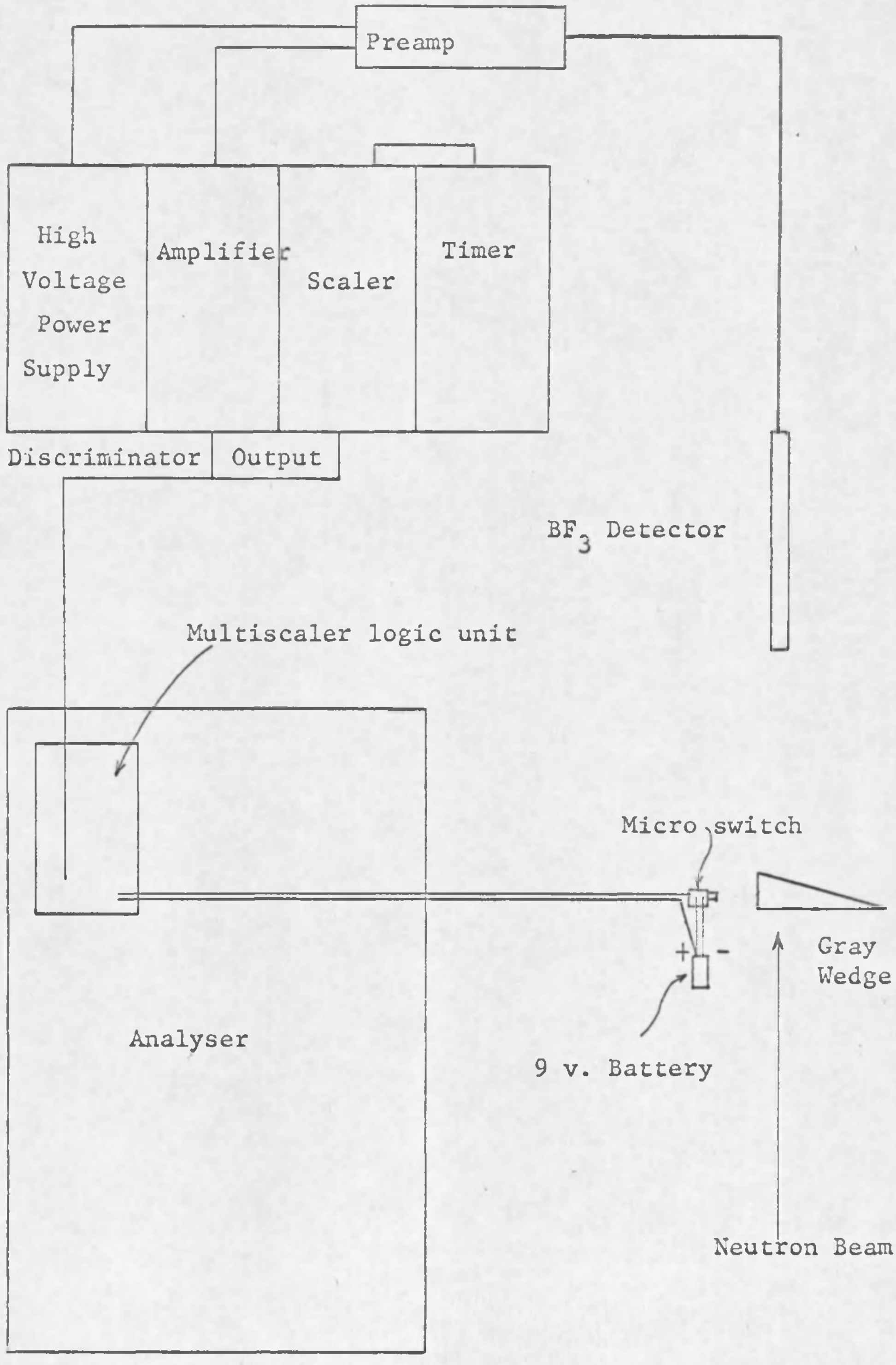


Figure 4. Box diagram of counting equipment

## CHAPTER IV

## DATA ANALYSIS

The theoretical description of this experiment places only one restriction on the function used to fit the attenuation curve; i.e., it has an inverse Laplace transform. There are several practical considerations, however, which lead one to believe that certain types of functions would be more suitable than others. One such consideration is that the neutron spectrum from a reactor is continuous. This eliminates a function which fits attenuation data quite readily; a sum of pure exponential terms. To say that a pure exponential has an inverse Laplace transform one must use the $\mathcal{L}^{-1}(1) = \delta(t)$ as shown by Fich (1951). Applying the translation operation then gives $\mathcal{L}^{-1}(e^{-P_2 s}) = \delta(t-P_2)$, i.e. the reactor spectrum would be represented by delta functions spaced along the energy axis.

A second consideration is that the F(s) should be defined at $s = 0$ (zero wedge thickness). In the present analysis the thickness does not go to zero, although that data point was taken. With a finite beam width, there is a discontinuity between the point $s = 0$ and the point at which the leading edge of the wedge has crossed the beam. With proper modification of the experiment, however, the zero thickness point could be approached very closely.

An example of a function which is not defined at $s = 0$ is $C(s) = \sum P_1/sP_2$. When four to nine such terms were used in a least squares program with the values of $P_2$ fixed in the range of values .25 to 4, the $P_1$'s would converge rapidly to give a curve which fit the data quite well. The resulting $\phi(E)$, however, was not really close to the expected shape of a reactor spectrum. It had negative regions in it and would indeed often be negative in the region where the Maxwellian should have been at its maximum value. The one advantage of this function was that $\phi(E)$ was positive in the region from .1 to 100 ev. and had reasonable slope and magnitude. The spectra resulting from using these terms are shown in the results section.

A third practical consideration is that the function should be such that the initial parameter estimates need not be exactly the correct answer. That is to say, the system described by the least squares program in the computer must have a finite area of stability (the larger the better). This was a criterion which caused the rejection of many functions. An example of one such function is $C(s) = \sum P_1 e^{-P_2 s}/(s+P_3)$. A sum of three such terms was fitted to an 800 point set of data which was not corrected for the constant portion of the total wedge cross section. The function would not converge unless several parameters were held constant during a computer run. The $\phi(E)$ which resulted from this function was not unlike the expected reactor flux. It was found, however, that in making the correction for the constant scattering cross section that one of the assumptions allowing the inversion of the function had been violated. Multiplying the C(s) by $e^{bs}$ gave one of the terms a positive exponent, disallowing the use of the translation operation. An attempt was made at correcting the data by $e^{bs}$ and fitting the resulting curve. This particular function could not be forced to fit the corrected data with all of its exponential coefficients of <u>s</u> less than zero.

An attempt was made to derive the form of terms which would yield a $\phi(E)$ which would vary as $(1/v)^k$ over a limited range of energy. The mathematics involved in the derivation could not be carried out but by setting up the equations it was seen that the function, $C(s) = \sum P_1 e^{-P_2 s}/(s+P_3)^{P4}$, which has the inverse Laplace transform of $f(1/v) = \sum P_1 e^{-P3(1/v-P_2)}(1/v-P_2)^{P4-1}U(1/v-P_2)$ would have great versatility as to type of flux curve which it could generate. Of course, convergence with a function such as this with so many fitted parameters, is very difficult to obtain with the Los Alamos Least Squares Program as shown by McWilliams(1962).

A final attempt at a function that would both fit the curve and give a reasonable spectrum was, $C(s) = \sum P_1/(s+P_2)^{P_3}$, where the $P_2$'s and $P_3$'s were given fixed values. Again the attenuation curve was fit very well but the flux was not the classic reactor spectrum.

## CHAPTER V

## RESULTS

The curves shown in this chapter all result from data taken with the sintered aluminum - boron carbide wedge. Attenuation curves are shown plotting every fifth point (Figs. 5 through 9). The line through the points is the least squares fit of the function:

$$C(s) = \sum_{n=1}^{6} \frac{P_{2n-1}}{s^{P_{2n}}} \tag{11}$$

where the $P_2n$ are given the values .5, 1, 1.5, 2, 3, 4, and the $P_{2n-1}$ are calculated by the program.

Large statistical fluctuations are seen in the regions where only a few hundred counts are stored in a channel. The shapes of the curves change quite noticeably as various foils are placed on the beam. The .005" cadmium foil produces a much flatter attenuation curve than the .005" indium foil, which is not much different than the beam with no foil placed in it.

A portion of the spectra for the cadmium foil and the no foil data sets are shown (Fig. 10). These spectra were generated by a subroutine to the least squares program which inverted the F(s) and solved equation (8) for sixty values of neutron energy. The no foil spectrum is compared with the ones for the indium foil and aluminum-boron carbide slug (Fig. 11).

The spectra shown for the two runs with cadmium in the beam are very si ilar in shape and vary in magnitude by about a factor of three, the amount by which the reactor power was increased between the two measurements. The slopes of the cadmium spectra are greater than 1/E at the lower end of the range (.1 ev) and slightly less than 1/E at the 10 ev point.

The no foil and indium foil spectra both have a slope greater than 1/E in this range, where theory would indicate a 1/E slope as shown by West (1963).

A plot of three of the spectra resulting fitting the data to:

$$C(s) = \sum_{n=1}^{6} \frac{P_{3n-1}}{(s+P_{3n-2})} P_{3n} \tag{12}$$

is shown (Fig. 12) where the parameters in the denominator were assigned arbitrary values in order to obtain a fitted function i.e., to avoid having the calculation diverge. The resulting function fit the data well, but the neutron spectra were not of the expected shape, and were negative in portions of the range of energy which the wedge was designed to describe best (.1 to 100 ev).

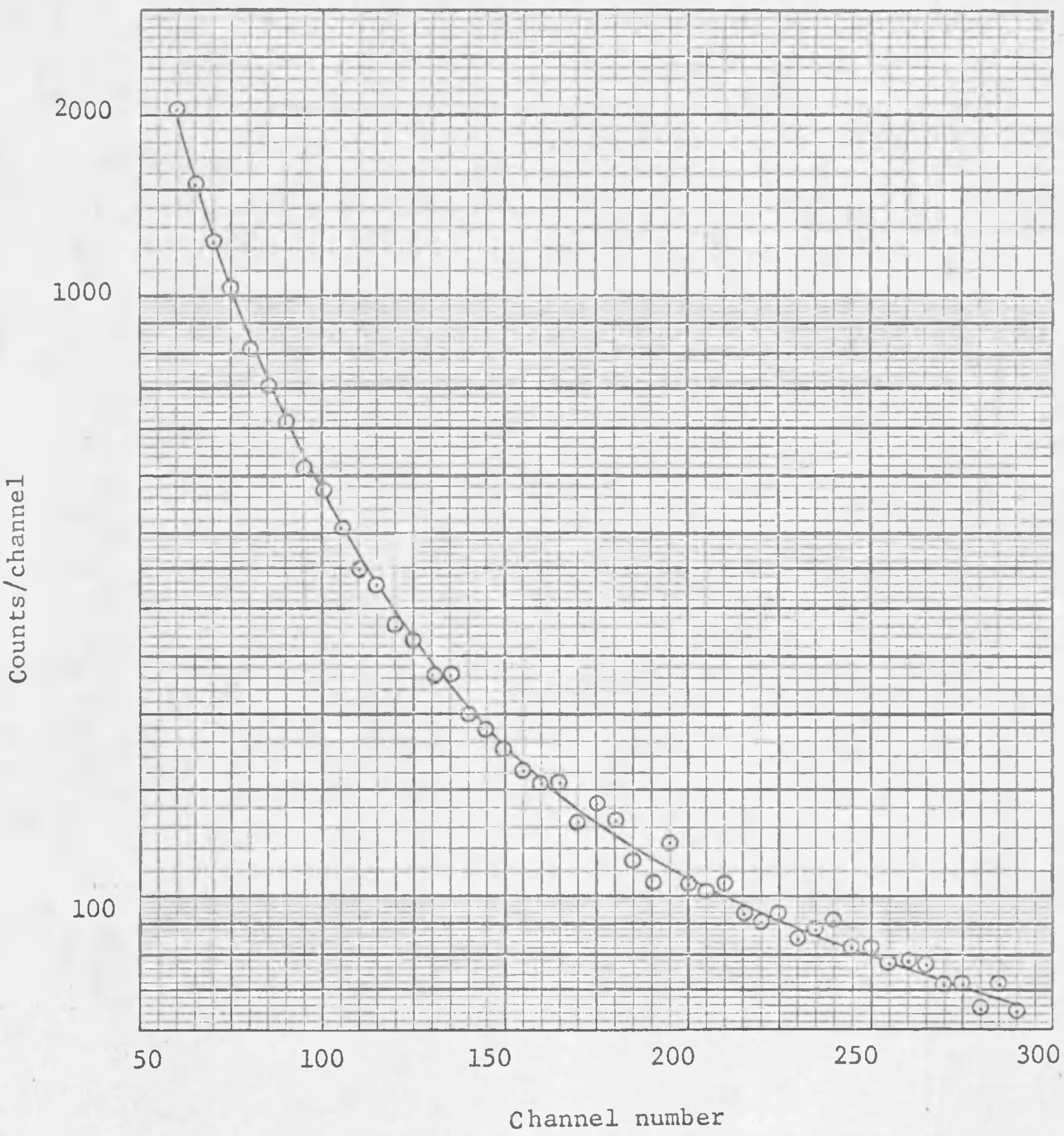


Figure 5. Attenuation curve for TRIGA neutron spectrum. The curve fitted by least squares to

$$f(s) = \sum_{n=1}^{6} \frac{P_{2n-1}}{s^{P_{2n}}} \quad \text{where } s \propto \text{channel number.}$$

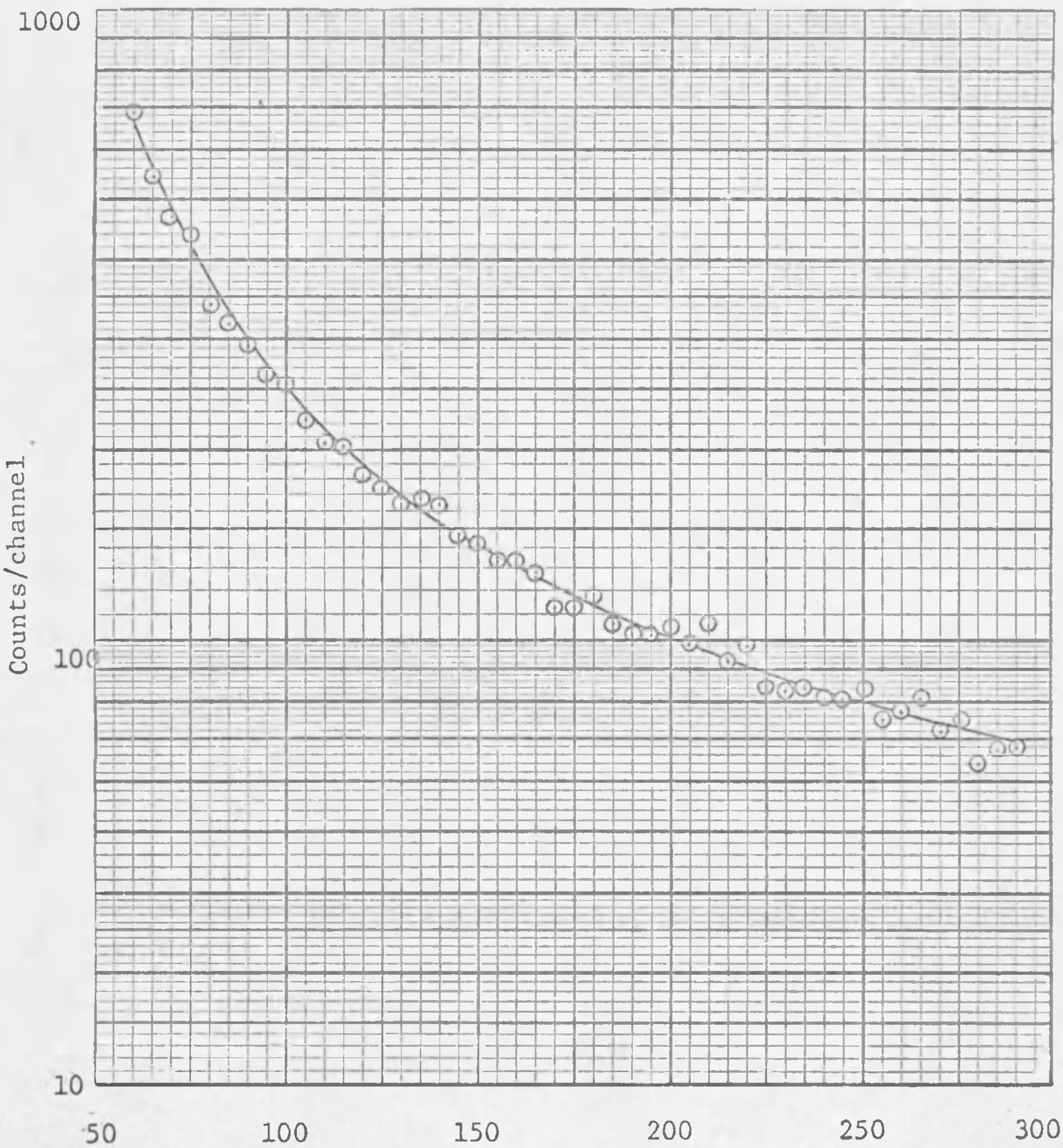


Figure 6. Attenuation curve with same spectrum as Figure 5 except an addition of .005" Cd in the beam. Also using the same fitting function.

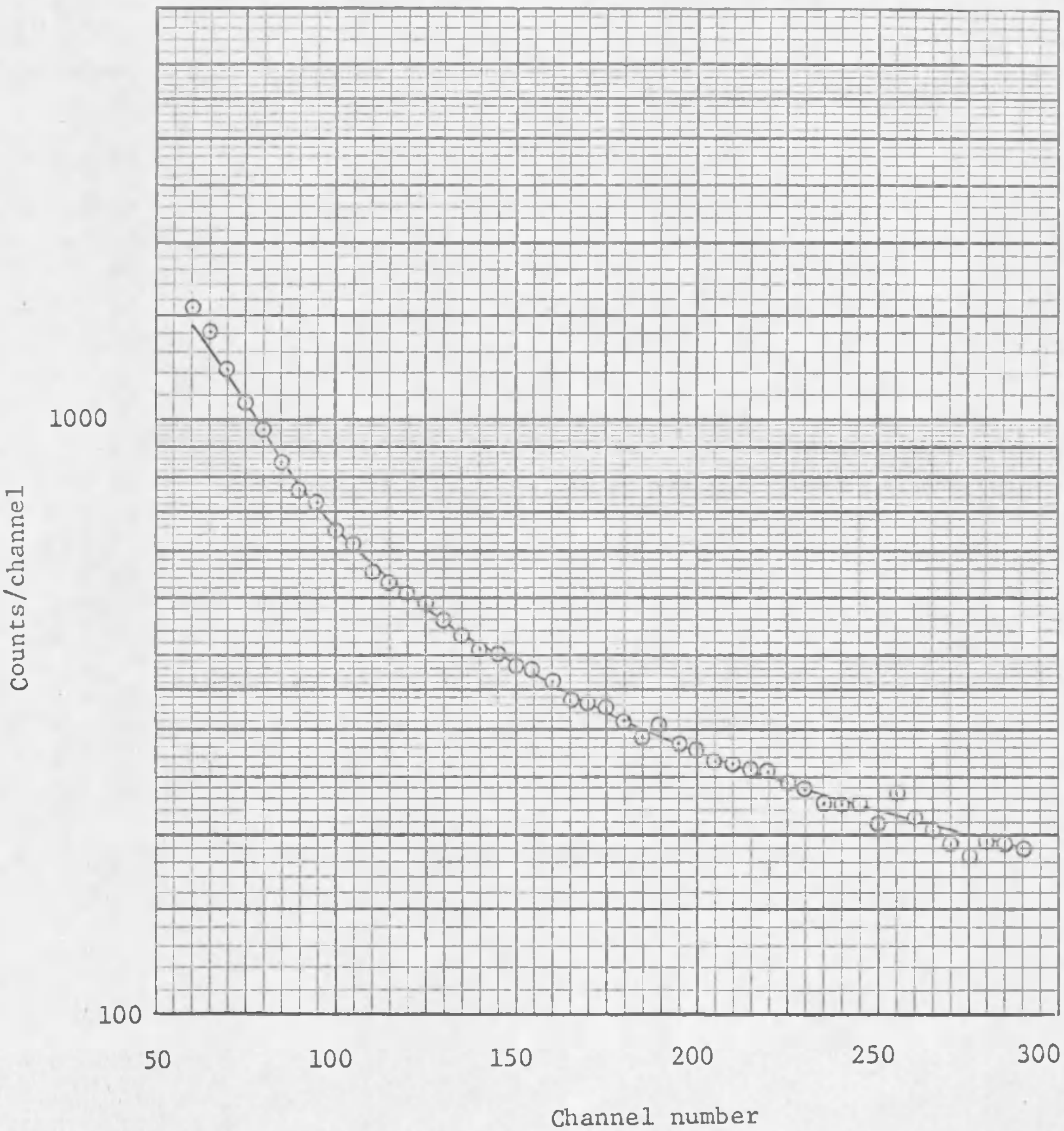


Figure 7. Attenuation curve same as Figure 6 (Cd. foil) except reactor flux three times higher.

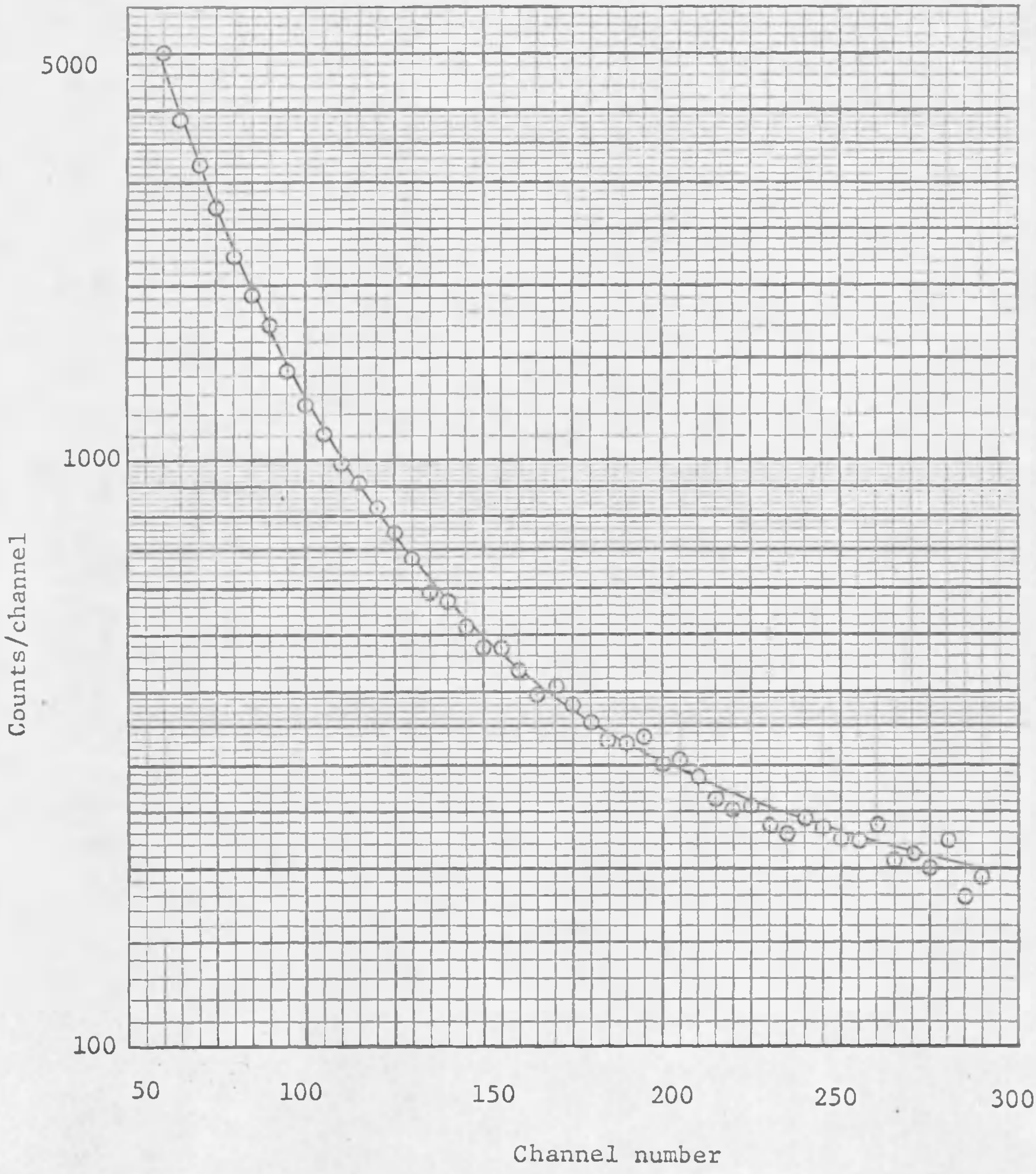


Figure 8. Attenuation Curve using .005" In. in beam.
Same reactor flux and fitting function as Figure 7.

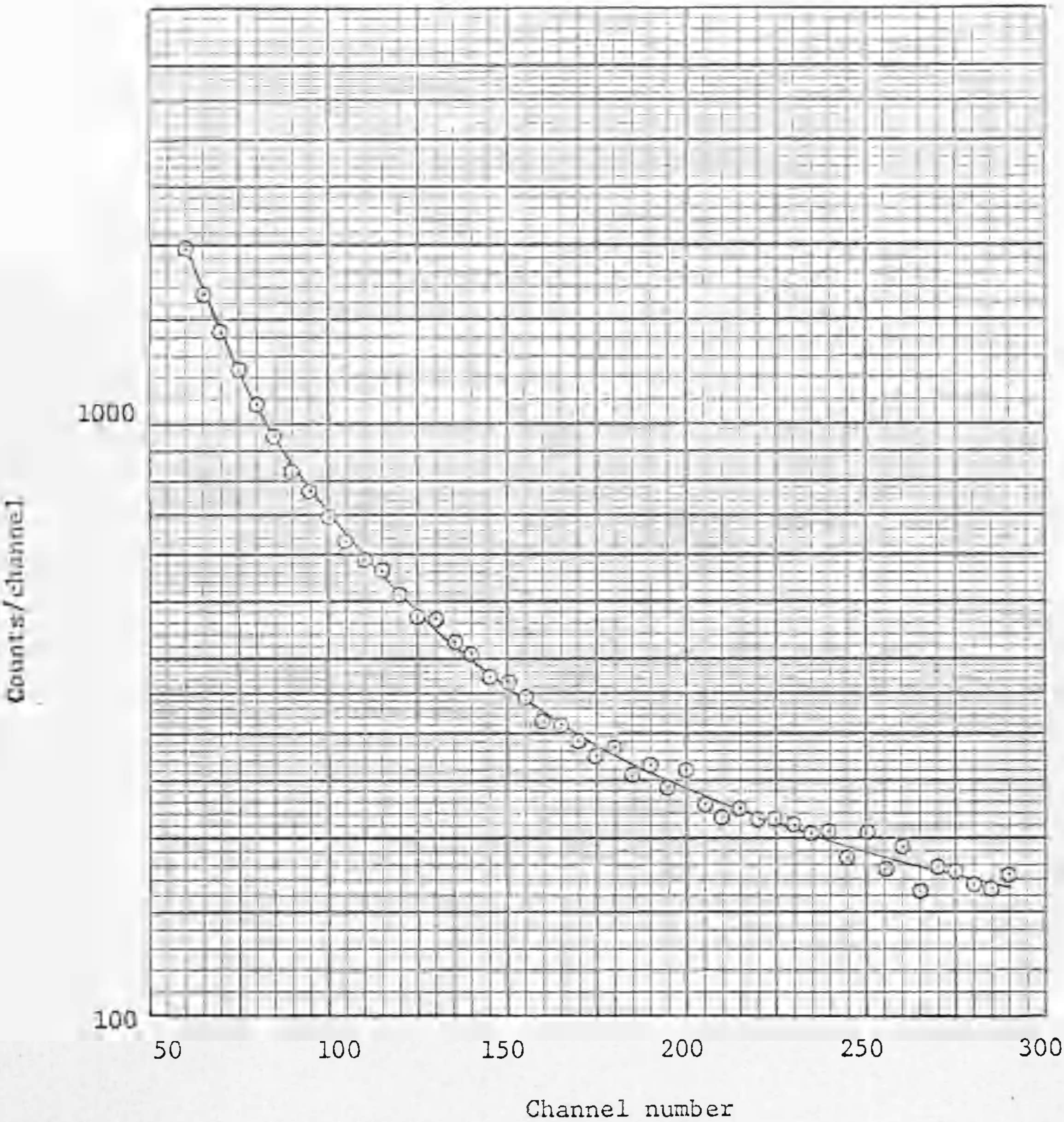


Figure 9. Attenuation curve using .462 $g/cm^2$ of 1:6 mixture of $B_4C$: Al. Same fitting function and flux as previous two figures.

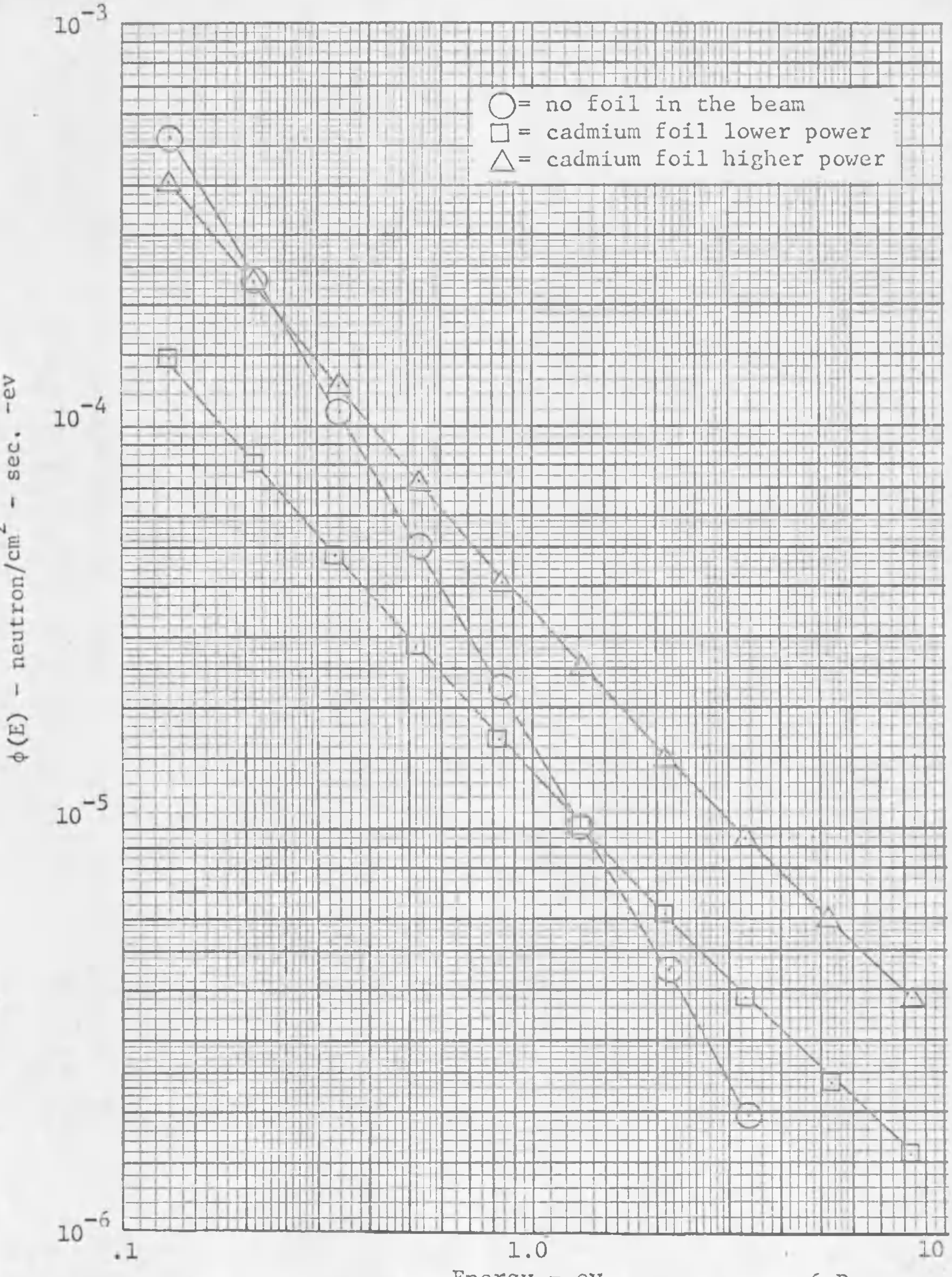


Figure 10. Spectra resulting from fitting data to $C(s)e^{bs} = \sum_{n=1}^{6} \frac{P_{2n-1}}{s^{P_{2n}}}$

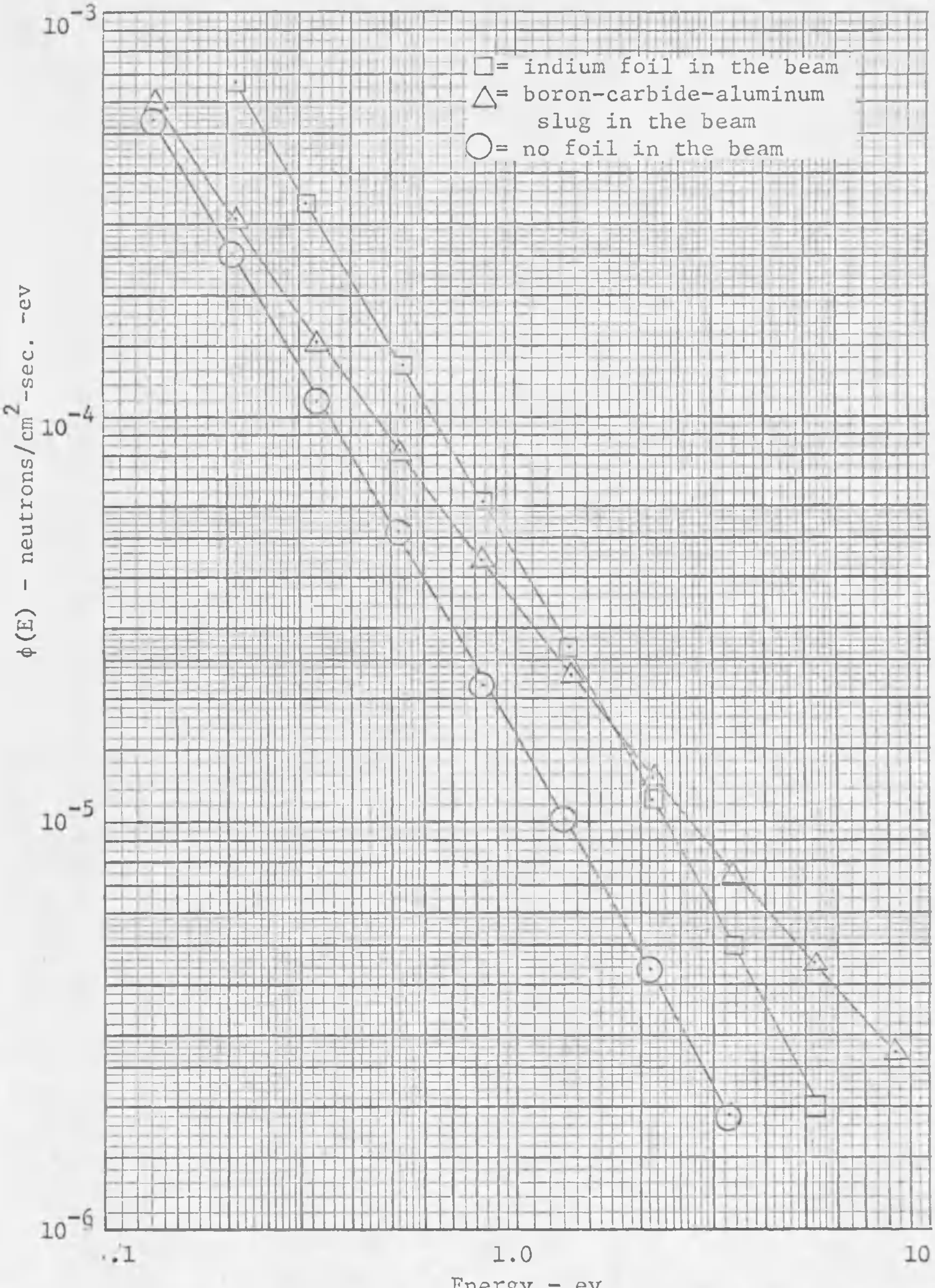


Figure 11. Spectra resulting from fitting data to $C(s)e^{bs} = \sum_{n=1}^{6} \frac{P_{2n-1}}{S^{P_{2n}}}$

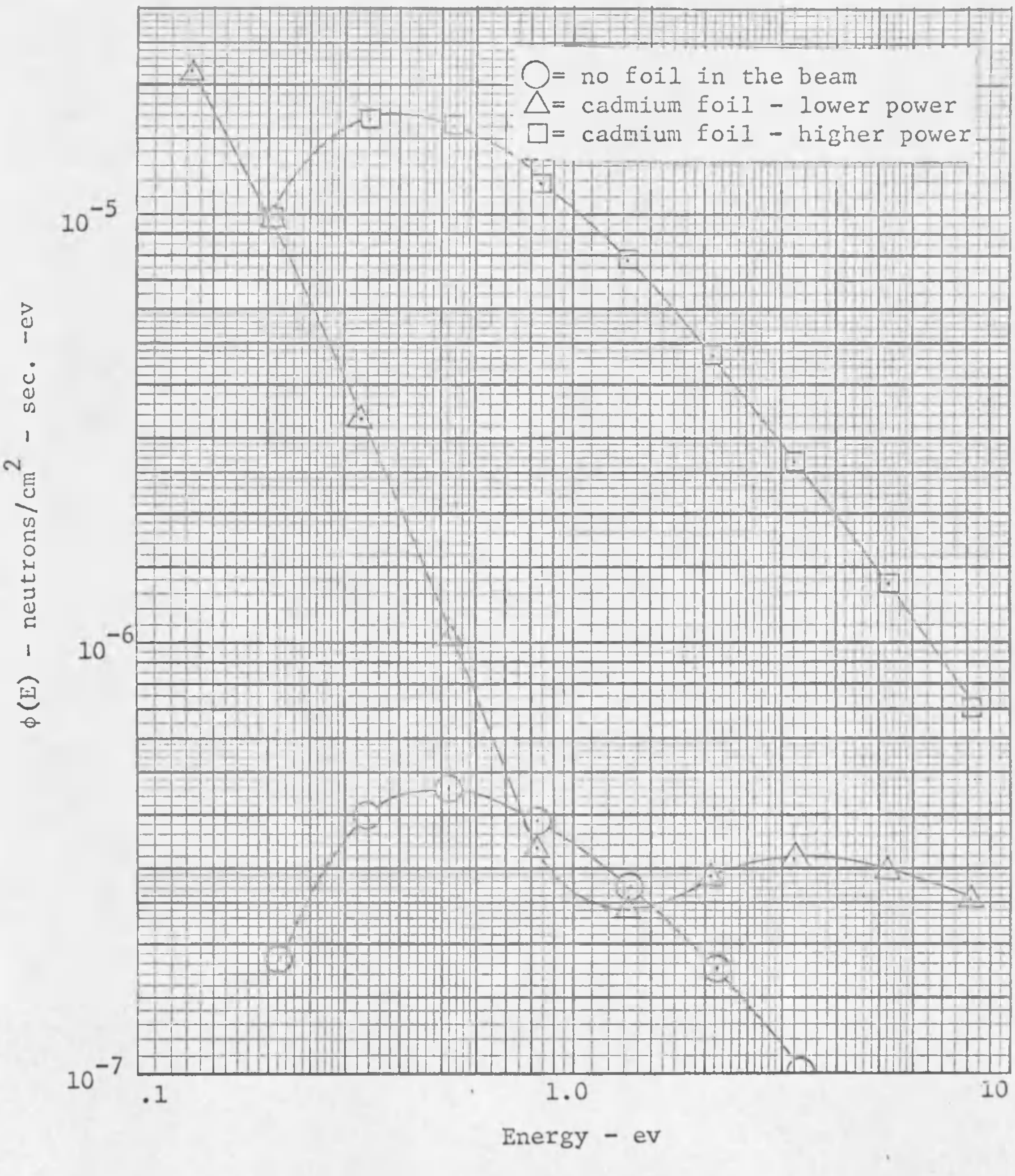


Figure 12. Spectra resulting from fitting data to

$$C(s)\, e^{bs} = \sum_{n=1}^{6} \frac{P_{3n-1}}{(s + P_{3n-2})} P_{3n}$$

## CHAPTER VI

## DISCUSSION AND CONCLUSIONS

Our initial purpose for studying this method of neutron spectrum analysis was to find out if it would work, not having found specific reference to it in a preliminary literature search. We came upon the article by Kuzminov at about the half way point in the study. The spectra which he displayed in that paper were for the BR-2 and BR-5 reactors. The article seemed quite general and gave no details of the analysis, i.e., nothing concerning fitting the attenuation curve or taking its inverse. In fact their representation of the extraction cross section of the "filter" (n-hexane) must have been a somewhat complicated function in itself. Attempts to trace their work in the literature failed. It seems that they must have hit upon a better means of analysis than was used in this paper. Some other methods which could be used to analyse the data follow.

It is quite possible that Kuzminov et al. fit their attenuation data without the use of a computer to one of the more complicated(and descriptive) terms mentioned in the analysis section of this paper. They were not trying to describe their spectra in great detail and therefore would not have needed to use more than a couple of terms.

Another alternative is to use a numerical method for computing inverse Laplace transforms. Schmittroth and Clayton (1959) describe a method. Their method is primarily designed to invert a complicated function whose inverse is not given in tables of transforms, but perhaps it could be incorporated into a method to invert data.

The data taken during the last part of this work seem to be adequate for the purpose of searching for proper fitting functions. There was a bit of deviation from the expected monotonic decrease in count rate with increase in wedge thickness. This was attributed to statistical fluctuations of the source and a little roughness in the wedge drive. The wedge itself was qualitatively uniform in boron density, although it was porous and hydroscopic (3% water by weight). A better wedge could probably be made by hot press techniques, as mentioned in an earlier section.

Assuming that the analysis was perfected to a point where a number of the more complicated fitting terms could be used, it would be necessary to change the data acquisition portion of the experiment to increase the resolution and range of the method. The energy range is determined in the low region by how "thin" the wedge can be made. The upper limit is set either by the point at which the wedge material loses its 1/v characteristics or by the maximum thickness of the wedge. For the present study, these limits should be .5 ev. to 100 ev.

This range should be capable of being vastly increased by using a gas ($BF_3$ or $He^3$) as the attenuator, in which case the gas pressure would be used to vary the blackness of the filter. Such a filter could decrease the lower limit to near 0 ev and increase the upper limit to a few hundred kev.

The resolution should be determined by: the fineness of the increments of absorber thickness, i.e. number of data points in a given range of absorber thickness, the statistical quality of the data, and the type and number of curve fitting terms used in the analysis of the data. Dead time considerations limit the counting rate so that an increase in the number of data points results in a proportional increase in the time required to take data of the same statistical quality. Increasing the number of data points and fitting terms also result in a proportional increase in computer time. The number of data points was reduced from 800 to 46 as it became apparent that it would be necessary to look for appropriate fitting terms and not primarily to measure the resolution of the method.

A careful analysis of experimental errors does not seem appropriate here, as the data analysis is not in a finished form. Had the resultant spectra been closer to the theoretical spectra, estimations of the errors due to statistical fluctuations, wedge drive nonlinearity, and wedge cross section uncertainty would have been combined to give the error associated with each data point. Performing the analysis on the data plus and minus its associated error would have then given an idea of the effect of such errors on the results.

Conclusions:

The data fitting functions used in this paper, equations 11 and 12:

$$C(s)\ e^{bs} = \sum_{n=1}^{6} \frac{P_{2n-1}}{s^{P_{2n}}}$$

and

$$C(s)\ e^{bs} = \sum_{n=1}^{6} \frac{P_{3n-2}}{(s + P_{3n-1})^{P_{3n}}}$$

are not the most appropriate functions, especially when arbitrary values must be assigned to some of the parameters to allow the computer calculations to converge. If the function:

$$C(s)\ e^{bs} = \sum_{n=1}^{N} \frac{P_{4n-3}\ e^{-P_{4n-2}s}}{(s + P_{4n-1})^{P_{4n}}}$$

could be fitted to the data one term at a time with all of the parameters free to vary, it is thought that a fairly complicated spectrum could be charted.

The increased number of data points obtained by continuously varying the attenuator thickness will add very little to the method until the analysis (curve fitting technique) is improved.

## LIST OF REFERENCES


Blosser, T.V., "Design of Two Neutron Responding Instruments: an Isotropically Responding Fast-Neutron Dosimeter and a Spectrometer for the Energy Range from Thermal to 10 kev", ORNL-3714, Oak Ridge National Laboratory, (1964).

Cooper, W.E., A Slow Neutron Chopper, M.S. Thesis, University of Arizona, (1963).

Dancoff, S.M., Kubitschek, H., Lichtenberger, H.V., Monk, G.D., Nobles, R.G., "Activation Cross Section by Boron Absorption", AECD-1781, AECD-1945, United States Atomic Energy Commission, (1948).

Fermi, E., Anderson, H.L., and Marshall, L. "Production of Low Energy Neutrons by Filtering through Graphite", MDDC-54, United States Atomic Energy Commission, (Declassified 1946).

Fich, S., Transient Analysis in Electrical Engineering, Prentice-Hall, Inc., (1951).

Kuzminov, B.D., "A Spectrometer for the Continuous Neutron Spectrum", Instruments and Experimental Techniques, 5, No. 2, pp. 253-54, (1962).

Lichtenberger, H.V., "Metallurgical Laboratory Reports", CP-3081, CP-2436, CP-2638, CP-3195, United States Atomic Energy Commission, (1948).

McWilliams, P. "The Solution of the General Least Squares Problem with Special Reference to High Speed Computers", LA-2367 AD., Los Alamos Scientific Laboratory, (1962).

Mikesell, R.E., Instruction Manual for the Torrey Pines TRIGA Reactors, GA-7275, General Atomic, (1966).

Schmittroth, L.A., Clayton, R.L., "A Numerical Method for Computing Inverse Laplace Transforms," IDO-16525, Phillips Petroleum Co., (1959).

Stehn, J.R., Goldberg, M.D., Magurno, B.A., and Wiener-Chasman, R. "Neutron Cross Sections", BNL=325, 2nd Edition, Brookhaven National Laboratory, (1964).

West, G.B., "Calculated Fluxes and Cross Sections for TRIGA Reactors", GA-4361, General Atomic, (1963).