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NEUTRON FLUX AND TISSUE DOSE STUDIES WITH FISSION THRESHOLD DETECTORS

PART I. EXPERIMENTAL MEASUREMENTS

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PART I. EXPERIMENTAL MEASUREMENTS

INTRODUCTION

The Radiation Instruments Branch, Division of Biology and Medicine, U.S.A.E.C., requested that the Health Physics Division of Oak Ridge National Laboratory make neutron measurements in connection with the Nevada test series Upshot/Knothole, Spring 1953. These neutron measurements would help in defining the tissue dose (energy dissipated per gram of average wet tissue) at different distances from the explosion. Such measurements were made and complete results reported in Project 24.2 report to the Test Director. The purpose of this report is to give a description of the methods and apparatus developed for the measurements.

Most of the neutron detection at previous tests had been done by gold (thermal neutron detector) and sulfur (threshold at about 2.5 Mev), therefore little was known about the number of neutrons with energies between thermal and 2.5 Mev. Since fission spectrum neutrons are largely in this energy region, we considered it important to have a detector that would be sensitive to these neutrons. Even though the tissue dose (Fig. 1) increases with neutron energy, it was anticipated that it would be significant in the intermediate (thermal to 2.5 Mev) energy region. It is noted here that the tissue dose* curve in Fig. 1 is a first collision curve, that is, in case of fast neutrons the curve when multiplied by the spectrum of neutrons actually entering a gram of tissue would give the energy dissipated in that gram. In the case of thermal neutrons the meaning of a first collision calculation is not so clear, since approximately half of the first collision dose

* The dose is given in rep units corresponding to an absorbed dose of 93 ergs per gram of soft tissue.

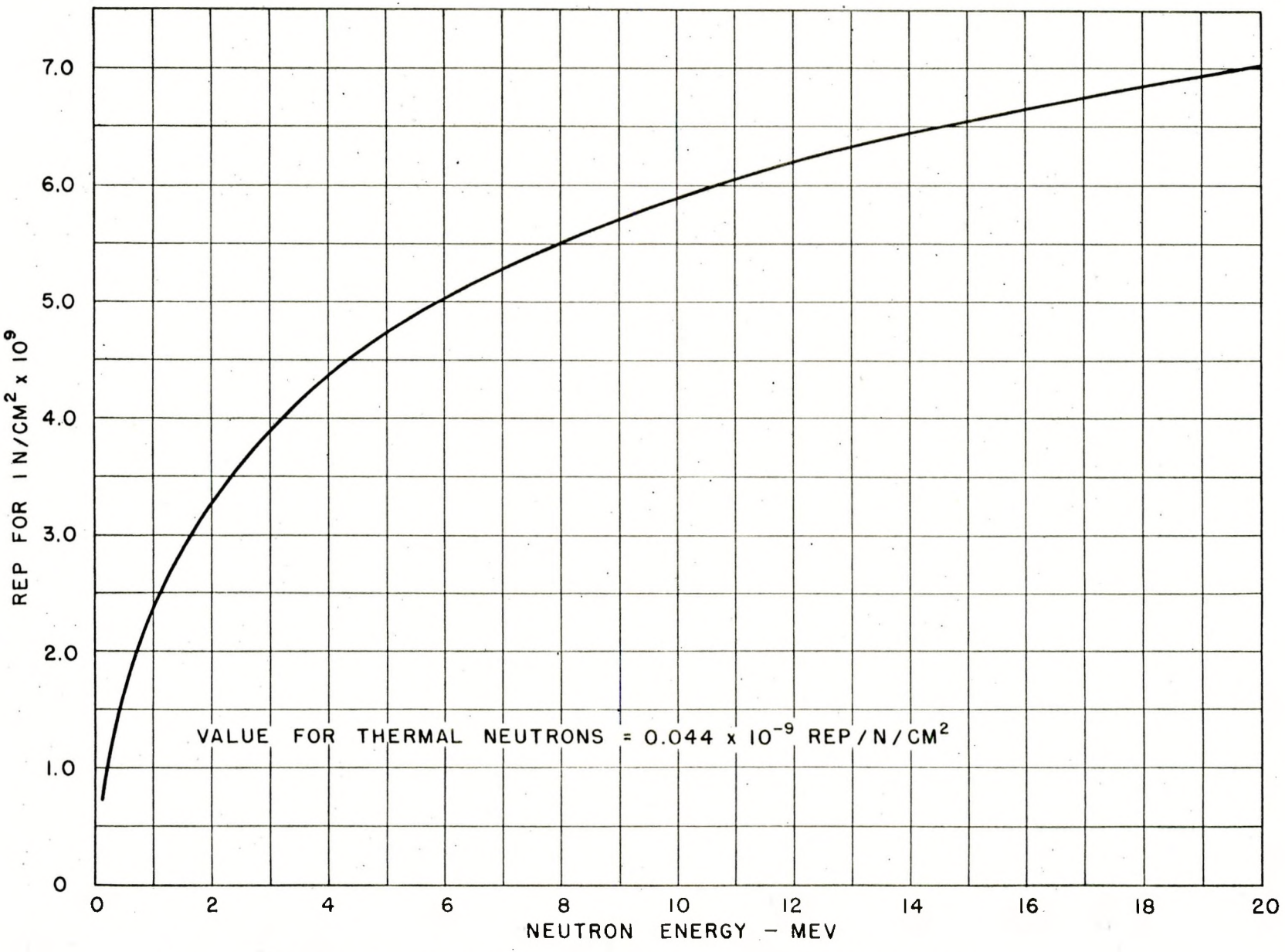

Fig. 1. First Collision Tissue Dose Curve, Using 1 rep = 95 ergs/gram Tissue.

would be due to the $H^1(n, \gamma)H^2$ reaction and the other half due to the $N^{14}(n, p)C^{14}$ reaction. Since protons are absorbed and not scattered out of the differential element where they are formed, the energy from the N^{14} capture would be proportional to the thermal neutron flux entering the element. The energy contribution of the $H^1(n, \gamma)H^2$ reaction, to the differential element, would depend not only on the thermal neutron flux entering the element but would depend in part on the gamma scattering from neighboring elements. Perhaps the best way to determine the thermal neutron dose would be by measuring the thermal flux incident to a body and use the type of calculations used by Snyder(1) to determine the energy distribution for both reactions separately. In any case, for simplicity, we adopt a first collision dose for thermal neutrons and mean that the first collision dose is the energy that would be received per gram of a small element of tissue, not surrounded by other elements of tissue, but being large enough to bring the secondary electrons from the 2.1 Mev gamma ray into equilibrium.

Our objective now reduces itself to dividing the neutron spectrum into sufficient groups so that the curve in Fig. 1 can be used to determine the first collision tissue dose.

ENERGY RESPONSE OF Pu^{239}, Np^{237} AND U^{238}

In Fig. 2(a) we see that the fission cross section of Pu^{239}(2) is nearly constant from 0.03 Mev to 14.0 Mev. Thus, a measure of the number of fissions would indicate the total neutrons/cm^2 in this region. Pu^{239} has a high thermal cross section (700 barns) which can be used to advantage in calibration of the number of fissions. During exposure to fast neutrons the thermal neutrons may be removed by B^{10} shielding of the Pu^{239}. Shown also in Fig. 2(a) is the fission cross

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1. Snyder, W. S., NUCLEONICS 6, No. 2, p. 46, February 1950.
2. Neutron Cross Sections, BNL-170.

section of Np^{237}[3] which has a threshold at approximately .4 Mev and is fairly flat above 1 Mev to 6 Mev. Another useful reaction is the fission of U^{238}[3] which is nearly constant from its threshold at 1.5 Mev to 6.0 Mev. Thus the use of these three fission detectors in conjunction with gold and sulfur permits the spectrum to be resolved into five energy groups.

If the fission cross section of Pu^{239} is multiplied by the absorption factor $e^{-\sigma N}$ where σ is the cross section of normal boron, and N is the number of boron atoms per cm^2, and if $N = 2.4 \times 10^{23}$ atoms/cm^2 the curve in Fig. 2(b) is obtained. A sphere 1-1/2" radius filled with B_4C (70% boron) pulverized to give a density of 1.59 grams/cm^3 gives the equivalent absorption. The responses of the threshold detectors have been approximated by step-functions which are shown in Fig. 2(b). Also shown for comparison is the response of the $S(n, p)P^{32}$ reaction (from AECU-2040).

SOURCES AND CONTAINERS

The Pu sources[4] consisted of 3/4" discs of solid metallic Pu, each containing approximately one gram of Pu. These were plated with Ni to reduce the alpha hazard. Uranium samples[5] were prepared in much the same manner. The one Np sample[6] (30 mg) that was available was prepared by deposition on a platinum disk. All samples were sealed in lucite containers which could be placed at the center of the boron container, Fig. 3.

COUNTING SYSTEM

Since Pu emits several low energy gamma rays, the gammas from the fission products of Pu are counted with a NaI scintillation counter, the detector set so that it is sensitive to gamma rays with energy approximately greater than 0.9 Mev.

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3. Supplement No. 1 to Neutron Cross Sections, BNL-170A.
4. We are indebted to Dr. Jane Hall, Los Alamos Scientific Laboratory, for the preparation of these sources.
5. Supplied by B. Harmatz, Oak Ridge National Laboratory.
6. Supplied and prepared by George Parker, Oak Ridge National Laboratory.

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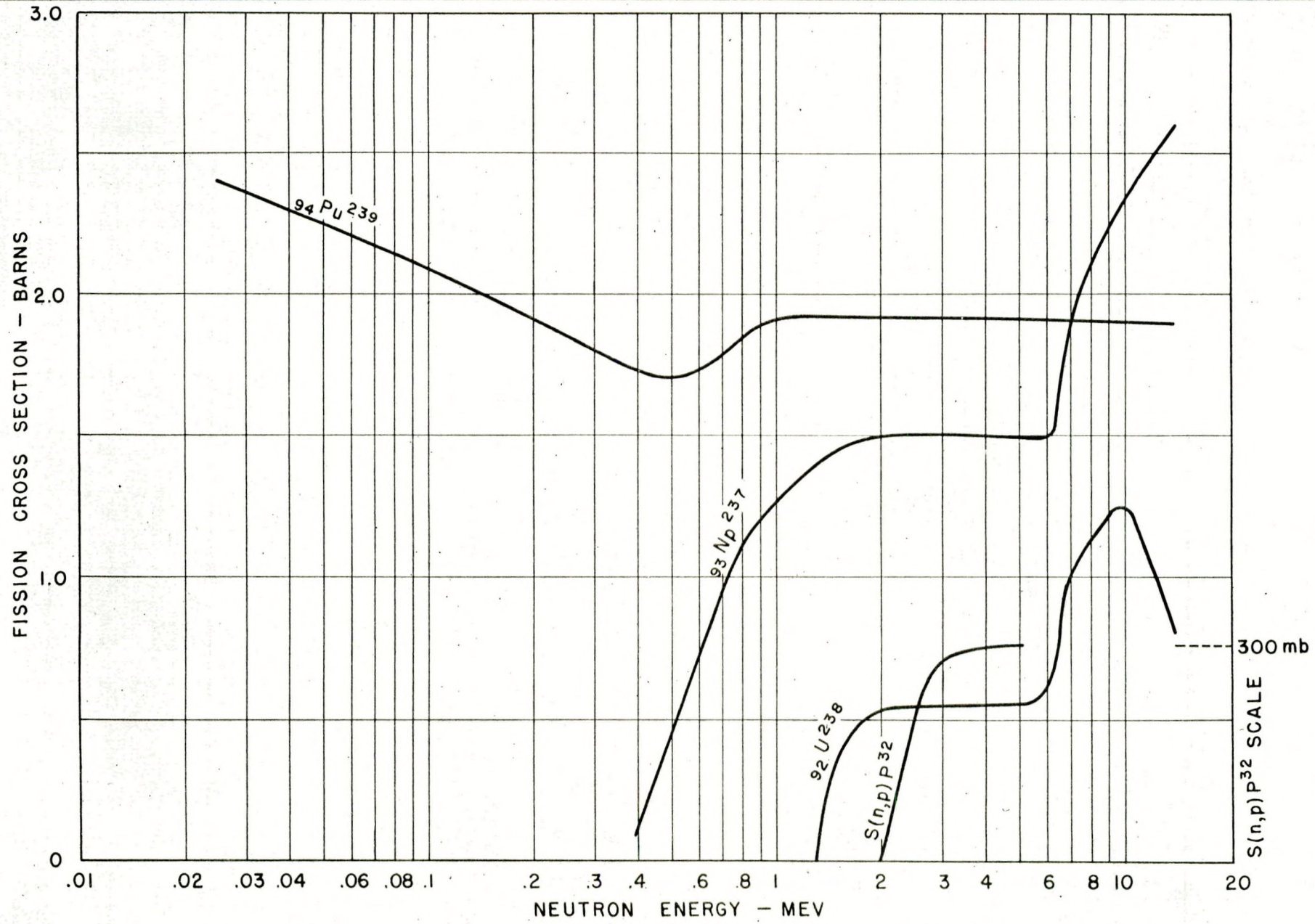

Fig. 2a. Fission Cross Sections of Pu^{239}, Np^{237}, and U^{238}.

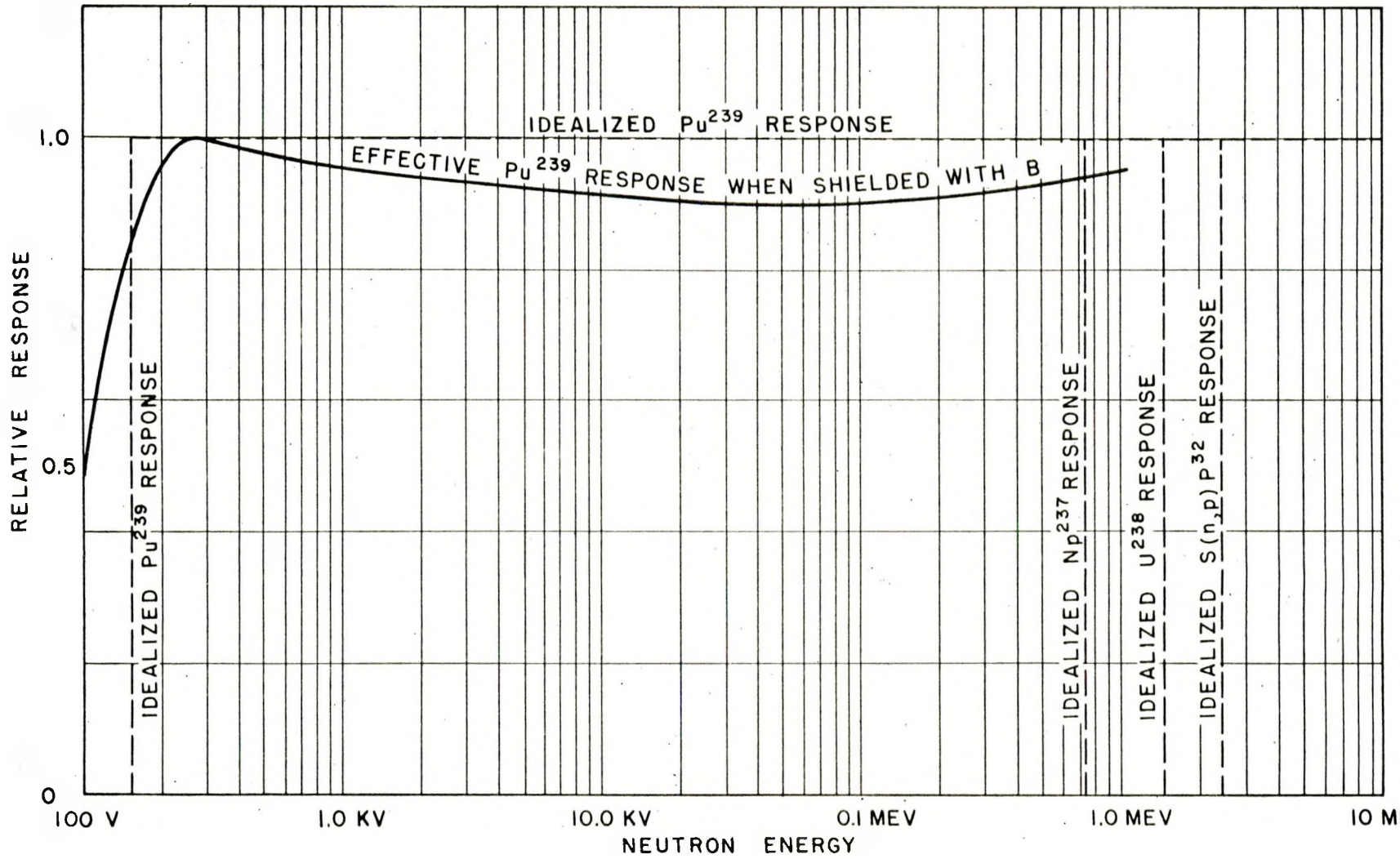

Fig. 2b. Response Curves and Step Function Approximations of the Threshold Detectors.

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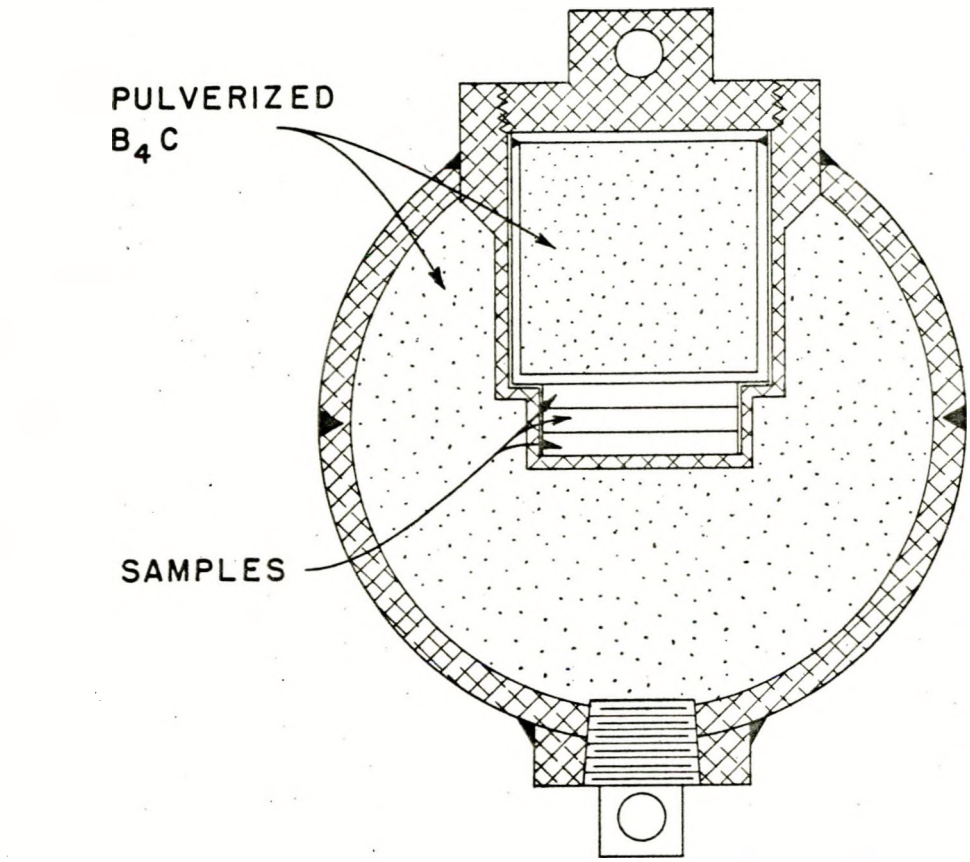

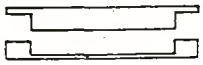

SAMPLE HOLDER

SCALE

Fig. 3. Boron Container for Exposing Foils to Fast Neutrons.

A diagram of the NaI crystal, with lead shield to decrease the number of low energy gammas striking the crystal, is shown in Fig. 4. The entire assembly is shielded with two inches of lead (not shown in the diagram). The output of the photomultiplier is fed through a preamplifier to a linear amplifier and integral pulse height selector. With the bias energy at approximately 0.9 Mev the background with no source present is about 40 counts/min and the one gram Pu sample (before neutron irradiation) adds another 160 counts/min, making a total effective background of 200 counts/min.

CALIBRATION OF COUNTER

As mentioned previously, the large fission cross section of Pu^{239} in the thermal region makes the calibration of the entire system of threshold detectors very simple. The procedure is to irradiate a Pu sample for a short time to a known thermal flux. This sample is .1 gram and 3/4" in diameter; hence, thin enough so that the attenuation of these thermal neutrons incident normally is less than 5%. If a one gram foil (with boron shield to eliminate the thermal flux) is exposed to fast neutrons the ratio $\frac{\text{counts/n/cm}^2\text{ (thermal)}}{\text{counts/n/cm}^2\text{ (fast)}} = \frac{\sigma_{Th}}{\sigma_f} \times \frac{.1\text{ gm}}{1\text{ gm}} = \frac{700}{2} = \frac{1}{1} = 35.0$, holds at any given time after irradiation. The Pu curve shown in Fig. 5 was, thus, constructed by following the decay of the 0.1 gram Pu calibration sample. The Np and U curves are obtained from the Pu curve by correcting for the difference in cross section, i.e.,

$$\frac{\text{counts/n/cm}^2\text{ Pu}}{\text{counts/n/cm}^2\text{ Np}} = \frac{2.0}{1.5} = 1.33$$

and

$$\frac{\text{counts/n/cm}^2\text{ Pu}}{\text{counts/n/cm}^2\text{ U}} = \frac{2.0}{.43} = 4.65$$

at the same time after irradiation and for equal masses.

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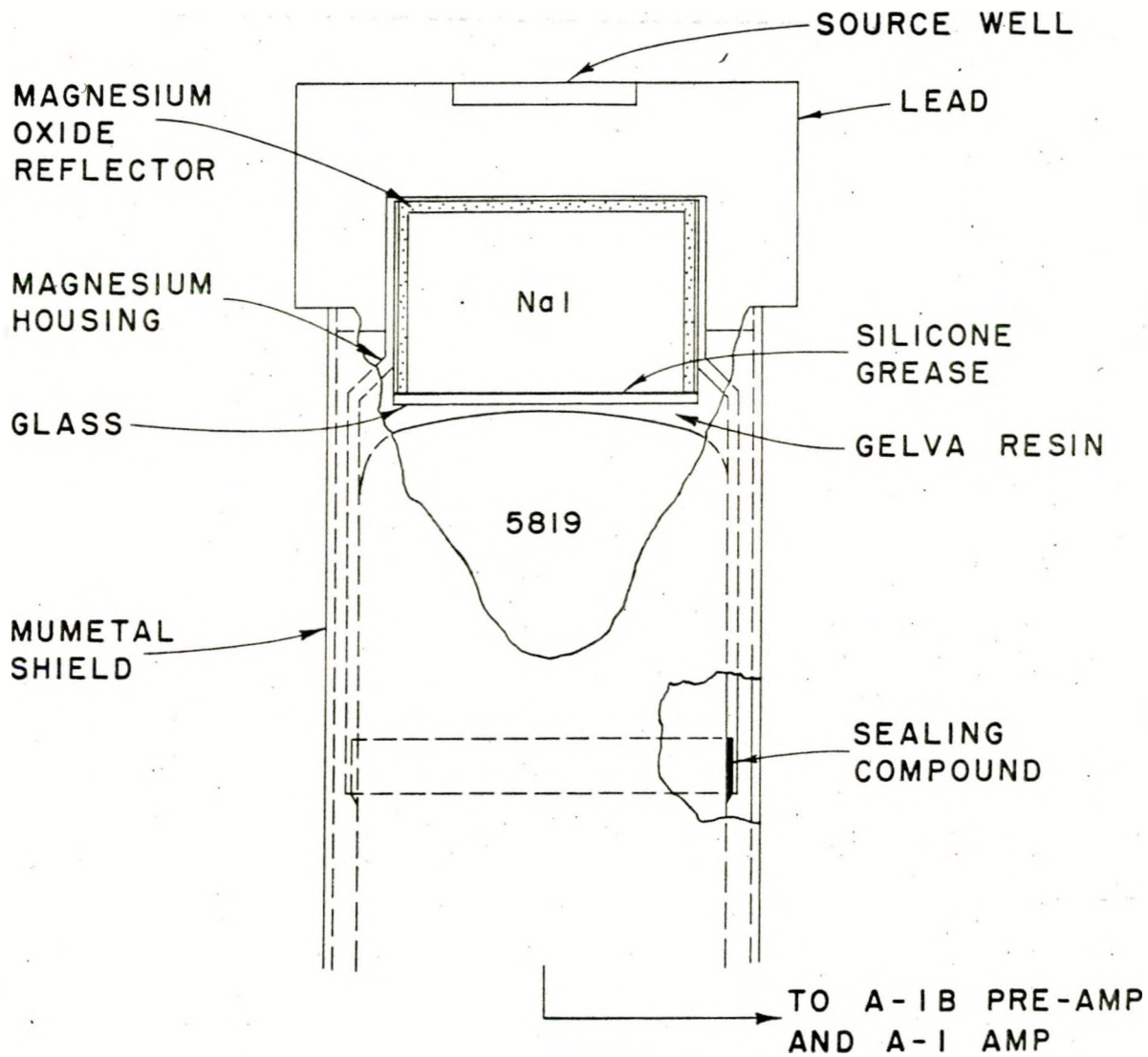

Fig. 4. Diagram of the Gamma-Ray Scintillation Detector.

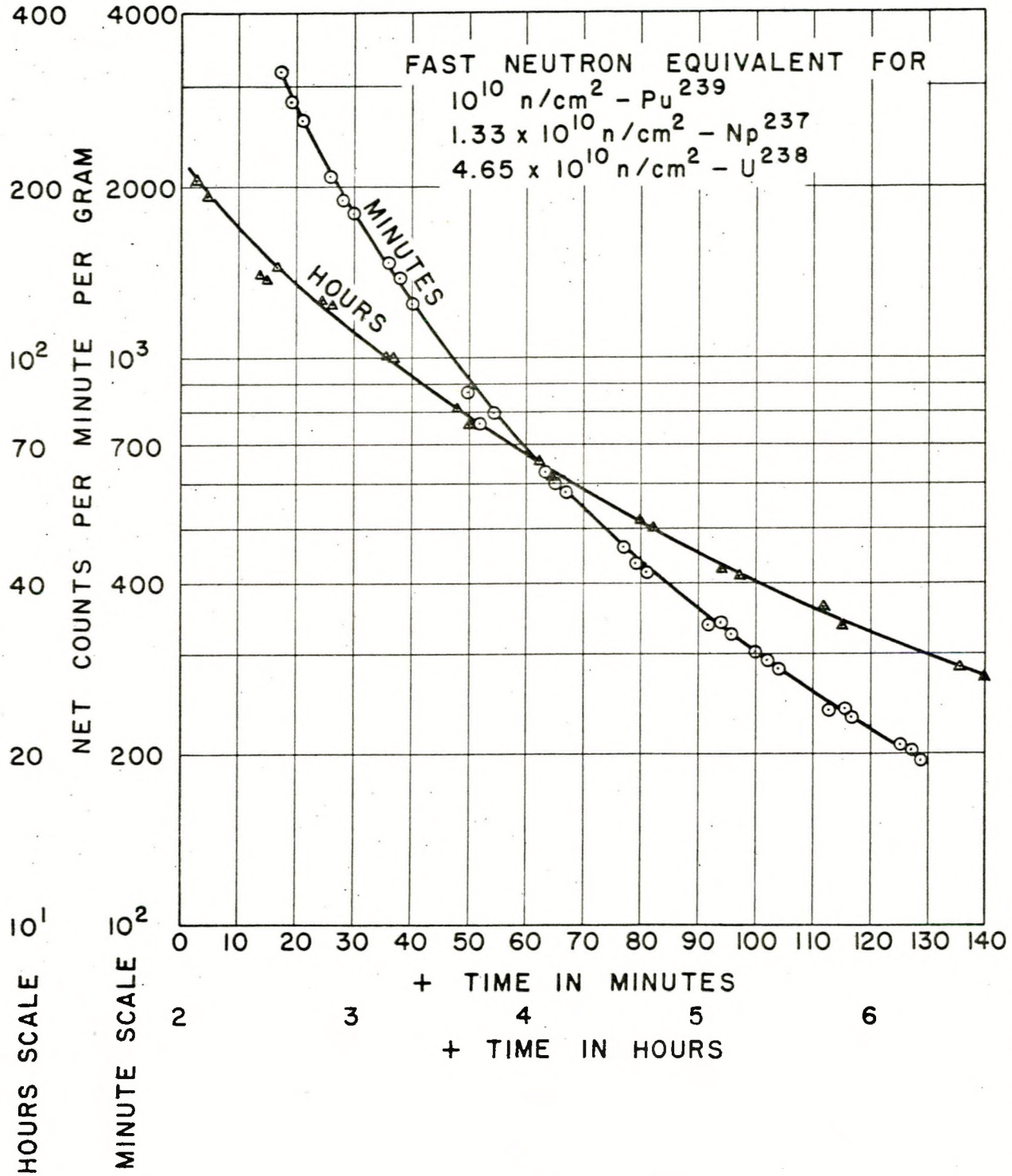

Fig. 5. Scintillation Count Rate vs Time for Indicated Fast Neutron Flux.

In the case of Np^{237} there is one difficulty; namely, that the reaction $Np^{237} + {}_0n^1 \longrightarrow Np^{238}$ is competitive with the fission reaction. Np^{238} has a 1.0 Mev gamma with a half-life of 2.1 days. One might eliminate this gamma by using a higher bias when counting Np. In any case the number of fission (N_f) can be determined from the counts vs. time (N_t) by obtaining the best fit to the equation

$$N_t = N_f \times C_t + N_a e^{-.693t/50}$$

where C_t is the experimental count rate at time, t, for fission of Pu; N_a is the count rate at time zero due to the thermal neutron activation, where t is given in hours.

A very important experimental detail is that the entire counting system must be standardized frequently, and this may be done by using a gamma source at a known distance above the detector. For example, if 0.25 millicuries Co^{60} is placed at 40.0 cm above the lead shield on the NaI crystal, the count rate is 440 counts/64 in one minute with energy bias set at 0.9 Mev.

TYPICAL FIELD RESULTS

Typical results obtained for one station and one shot are shown in Fig. 6. The histogram plot shows the number of neutrons in each energy interval, where it is noted that the predominate number is between the gold and sulfur thresholds. Rep figures in the second column, obtained by multiplying the number of neutrons in each interval by the curve in Fig. 1, for the mean energy of the interval, show that the major part of the tissue dose is detected by Pu, Np, and U.

ACKNOWLEDGEMENT

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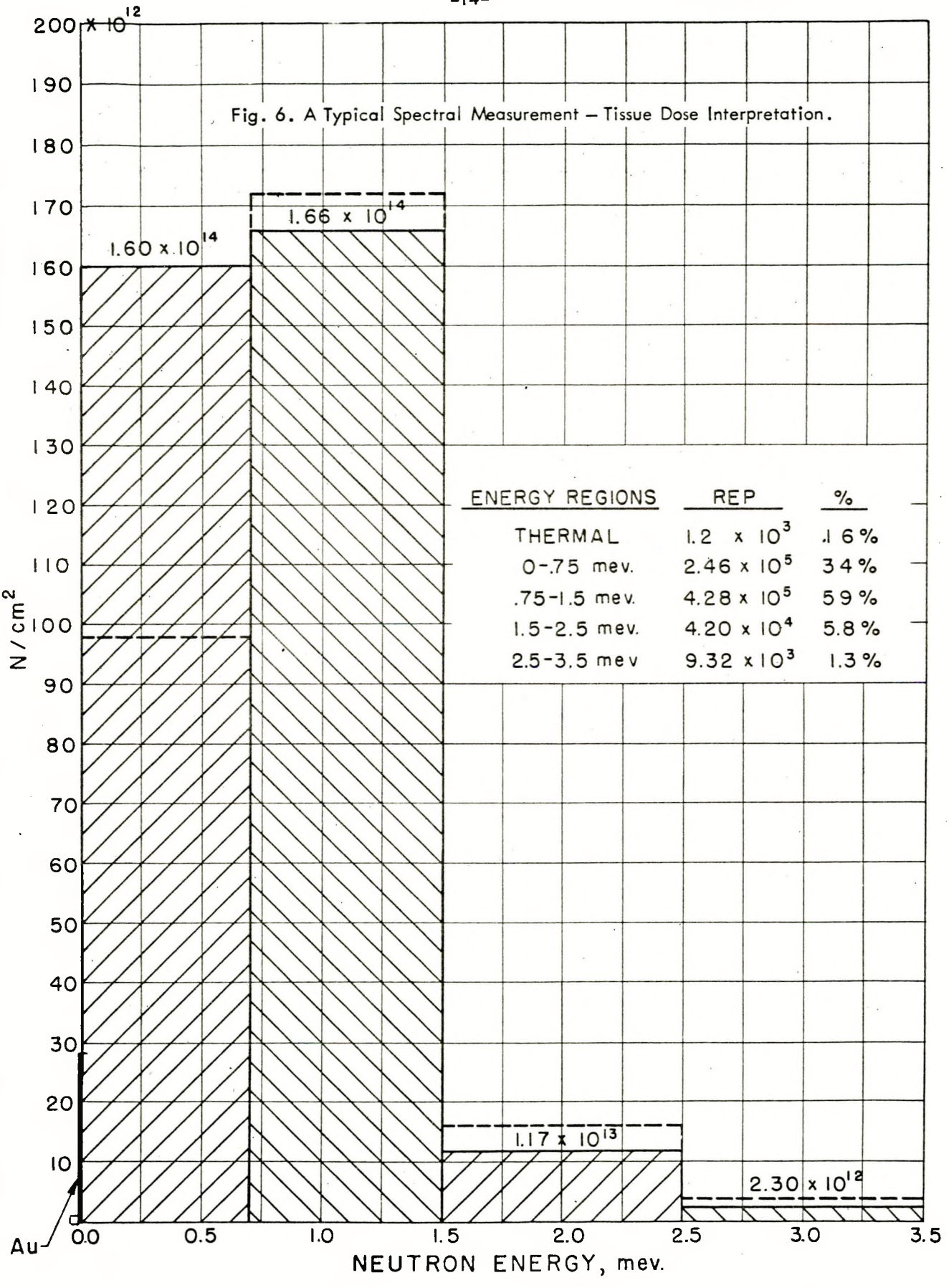

Fig. 6. A Typical Spectral Measurement – Tissue Dose Interpretation.

ENERGY REGIONS	REP	%
THERMAL	1.2×10^3	.1 6%
0-.75 mev.	2.46×10^5	34%
.75-1.5 mev.	4.28×10^5	59%
1.5-2.5 mev.	4.20×10^4	5.8%
2.5-3.5 mev	9.32×10^3	1.3%