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PRELIMINARY MEASUREMENT OF THE $^{235}U(n,f)$ CROSS SECTION UP TO 750 MeV

by

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ABSTRACT

The recently commissioned high energy white neutron source was used to measure the neutron fission cross section of ^{235}U in the energy range from 0.6 to 750 MeV. The shape of the cross section from 0.6 to 20 MeV agrees well with previously reported data. Absolute values for the $^{235}U(n,f)$ cross section from 20 MeV to 750 MeV were obtained by normalizing the measured shape of the cross section to data in the 10 to 20 MeV range.

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1. INTRODUCTION

The $^{235}U(n,f)$ cross section has been measured up to approximately 20 MeV incident neutron energy[1]. Between 20 and 25.7 MeV a few data points have been reported by Pankratov[2]. We report here the results of measurements that extend the data for this reaction up to approximately 750 MeV incident neutrons using the recently constructed high energy white neutron source at the Weapons Neutron Research (WNR) Target-4 (see Ref. 3). This reaction is particularly important for experiments at WNR since it would provide a convenient method of monitoring the neutron flux during the course of experiments. These results should be considered preliminary in the sense that a significantly better experiment is being planned for the next run cycle.

To determine the absolute fission cross section above 20 MeV, it is necessary to know the neutron flux incident on the fission foils as a function of neutron energy. The following procedure was used to calculate the neutron flux. The absolute flux between 0.6 and 20 MeV was obtained by measuring the yield of the ^{235}U fission chamber and using the known ^{235}U fission cross section. The relative shape of the neutron flux between 10 and 800 MeV was determined from the time-of-flight yield obtained with a thin plastic scintillator and a calculated efficiency for that detector. The relative flux, obtained with the plastic scintillator, was normalized to the absolute measurement obtained using the fission chamber in the overlap region between 10 to 20 MeV. This procedure resulted in "absolute" values for the neutron flux from the white source at 15° from 0.6 to 750 MeV. The main uncertainty in the neutron flux measured this way comes from the calculated values for the energy dependence of the relative efficiency of the plastic scintillator. The values of the ^{235}U cross section we present here are based on this flux and therefore should be considered preliminary.

In section 2 we describe the experimental set-up, and in section 3 we describe the data analysis and results. In section 4 we compare the results with similar measurements in nearby nuclei.

2. EXPERIMENTAL SET-UP

High energy neutrons were produced by focusing the pulsed 800 MeV proton beam from the Los Alamos Meson Physics Facility (LAMPF) accelerator on a 7.5-cm long, 3-cm diam. tungsten target. Two flight paths, one located at 15° left (15L) and one located at 15° right (15R) with respect to the incident proton beam, were used in this measurement. The fission chamber was used at both the 15R detector station at a distance of 15.0 m and at the 15L detector station at a distance of 42.3 m. The plastic scintillator was located in the 15L detector station at a distance of 42.3 m.

2.1 The Fission Chamber

The fission chamber consisted of two 14.6 cm diam. foils each containing 340 $\mu g/cm^2$ of uranium enriched to 91.1% in ^{235}U. The chamber was filled with approximately 2 atm of Argon-Methane (P-10) gas. A diagram of the chamber is shown in fig. 1. In this configuration, the response of the detector to gamma-rays is suppressed. The signals from the two active foils were summed together, amplified, and connected to a constant fraction timing discriminator (CFTD). The low energy threshold was set in the valley between the fission fragments and the low energy pulses from alpha decay. The output of the CFTD was used to start a time to amplitude converter (TAC) which was stopped by a signal derived from the incident proton beam pulse. The range of the TAC was 2 μs; for the 15.0 m flight path this allowed us to measure the fission cross section down to a neutron energy of 0.6 MeV.

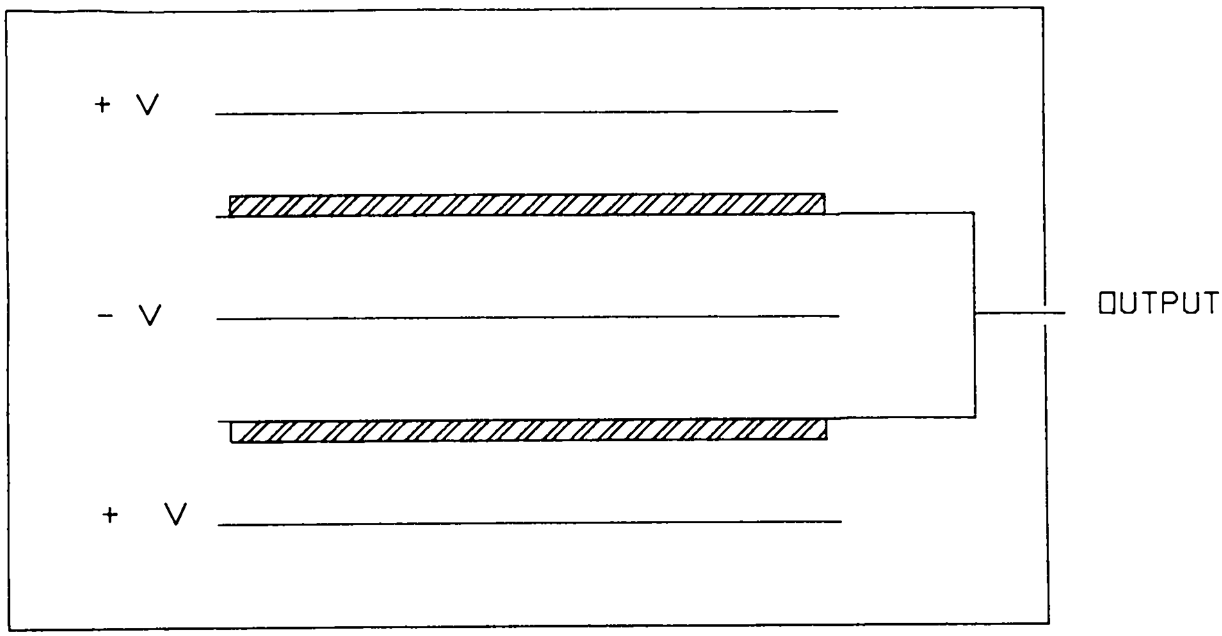

Fig. 1. Diagram of the fission chamber used in measuring the neutron flux at Target-4 and the ^{235}U fission cross section. The uranium deposits are shown in the cross hatched region.

The beam spot size at the detector location on the 15R flight path was approximately 2.54 cm wide and 5.1 cm high. The beam spot on the 15L flight path was 2.4 cm high and 5.0 cm wide. The exact neutron beam location and size were obtained by exposing a polaroid film. A typical time-of-flight spectrum for the fission chamber is shown in fig. 2.

2.2 The Plastic Scintillator

The plastic scintillator used to measure the neutron flux was made of NE-102 ($CH^{1.1}$). The size of the detector was 5.1 cm x 5.1 cm x 0.16 cm thick. It was coupled through an RCA 8575 phototube via a light guide. The anode signal was sent through a CFTD to start a TAC which was set to a 2 μsec full scale range. The threshold of the plastic scintillation detector was set at 0.12 MeV with a 3.5 MeV ^{106}Ru electron source. The stop signal of the TAC was obtained from a beam pick-off located ahead of the neutron production target. A typical time-of-flight spectrum is shown in fig. 3. At a 42.3 m flight path we expect a neutron resolution of approximately 0.7 MeV for 100 MeV neutrons and about 10 MeV for 500 MeV neutrons, assuming 1 nsec time resolution.

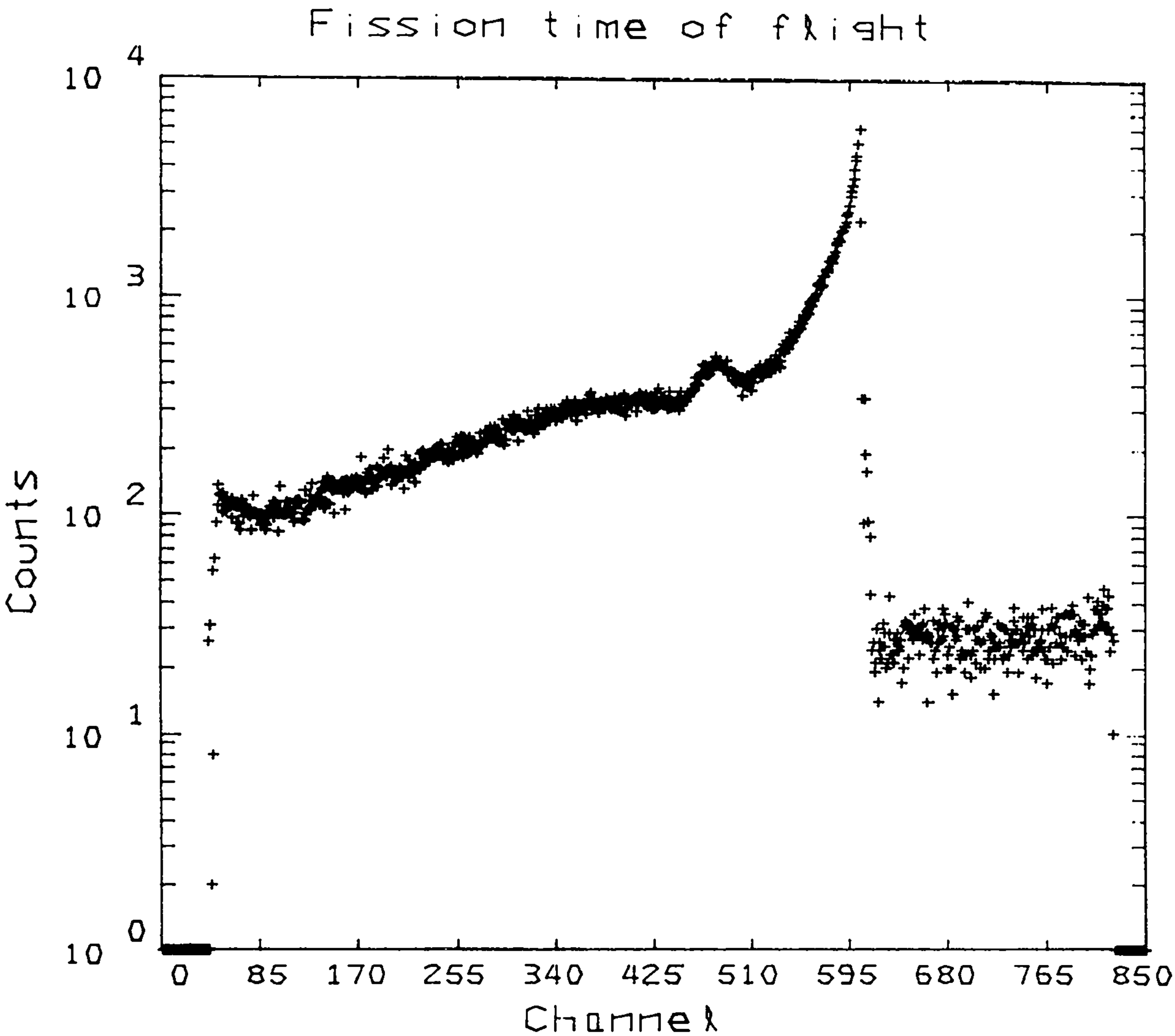

Fig. 2. Typical time-of-flight spectrum for the fission chamber.

3. DATA ANALYSIS AND RESULTS

The shape of the neutron flux above 20 MeV was obtained from the plastic scintillator data using the following formula:

$$N_n(E) = \frac{D(E)}{\Delta E_n A_n(E) \epsilon(E) \Omega} \quad .$$

$N_n(E)$ is the number of neutrons/MeV/sr detected in the run. $D(E)$ is the number of counts measured at neutron energy E in the neutron energy range ΔE_n. $A_n(E)$ is the neutron attenuation in the 42.3 m of air, and Ω is the solid

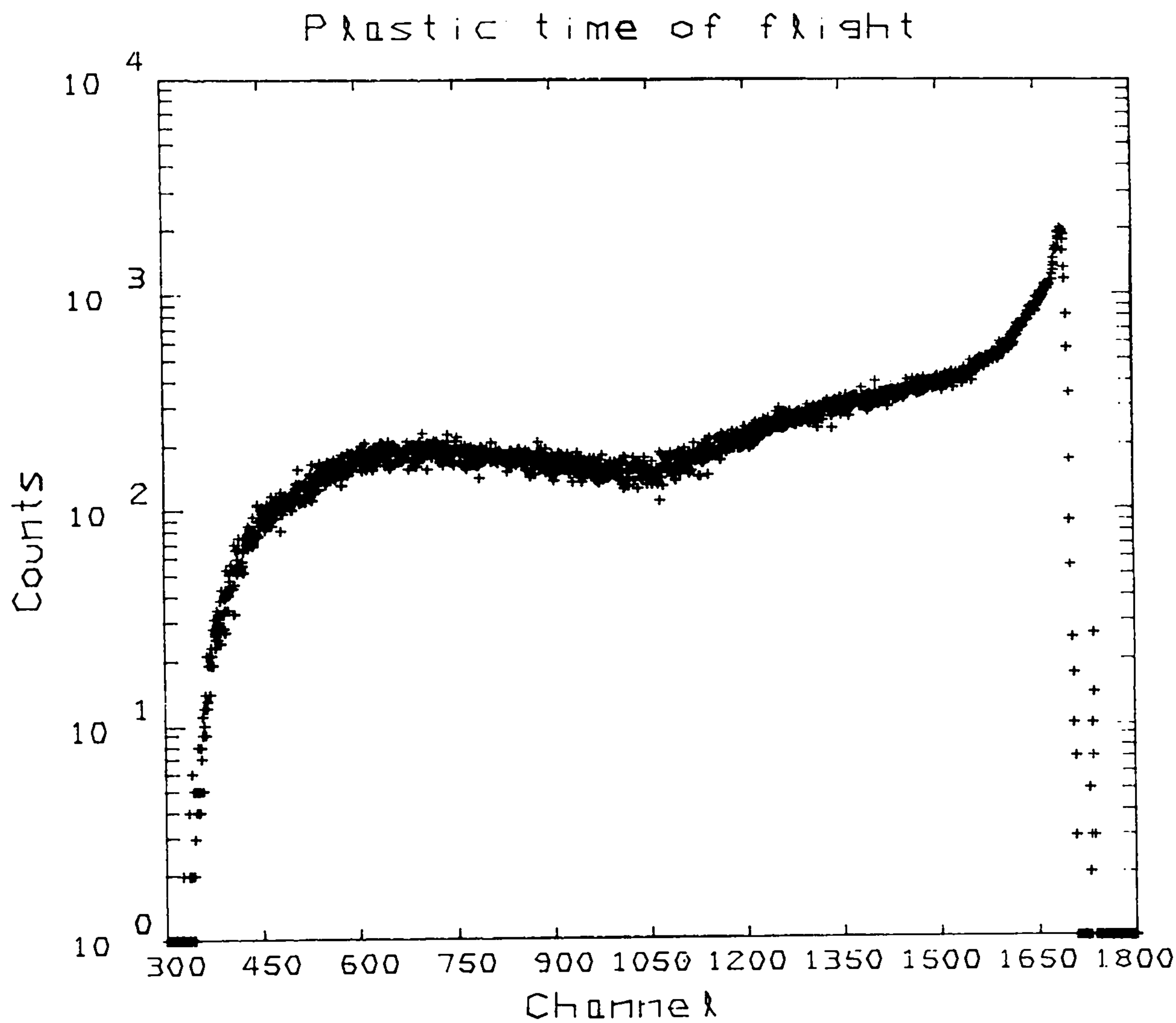

Fig. 3. Typical time-of-flight spectrum for the plastic scintillation detector.

angle of the neutron beam as defined by the collimation system. This solid angle was calculated using a ray-tracing Monte-Carlo code for the geometry of the flight path.

The efficiency of the scintillator, $\epsilon(E)$, was calculated using the modified Monte Carlo code STANTON[4] of Cecil *et al.*[5]. The modifications to the code include the measured values for the n-C cross-sections up to 1.5 GeV compiled by Del Guerra[6]. Below 10 MeV, the detector efficiency is dominated by the well known n-p total cross section. Thus, the shape of the calculated neutron flux in this energy region may readily be compared with the absolute neutron flux measured with the neutron fission chamber. Fig. 4 shows the ratio of the absolute flux measured using the fission chamber to the relative flux measured using the plastic scintillator in the overlap energy region between 10 and 20 MeV. The average value of the ratio or the normalization is $(1.97 \pm .16) \times 10^{-7}$. Thus, we may assume that the energy dependence of the relative neutron flux obtained with the plastic scintillator is determined to an accuracy of approximately 10%. Fig. 5 shows the combined flux from 0.6 to 750 MeV.

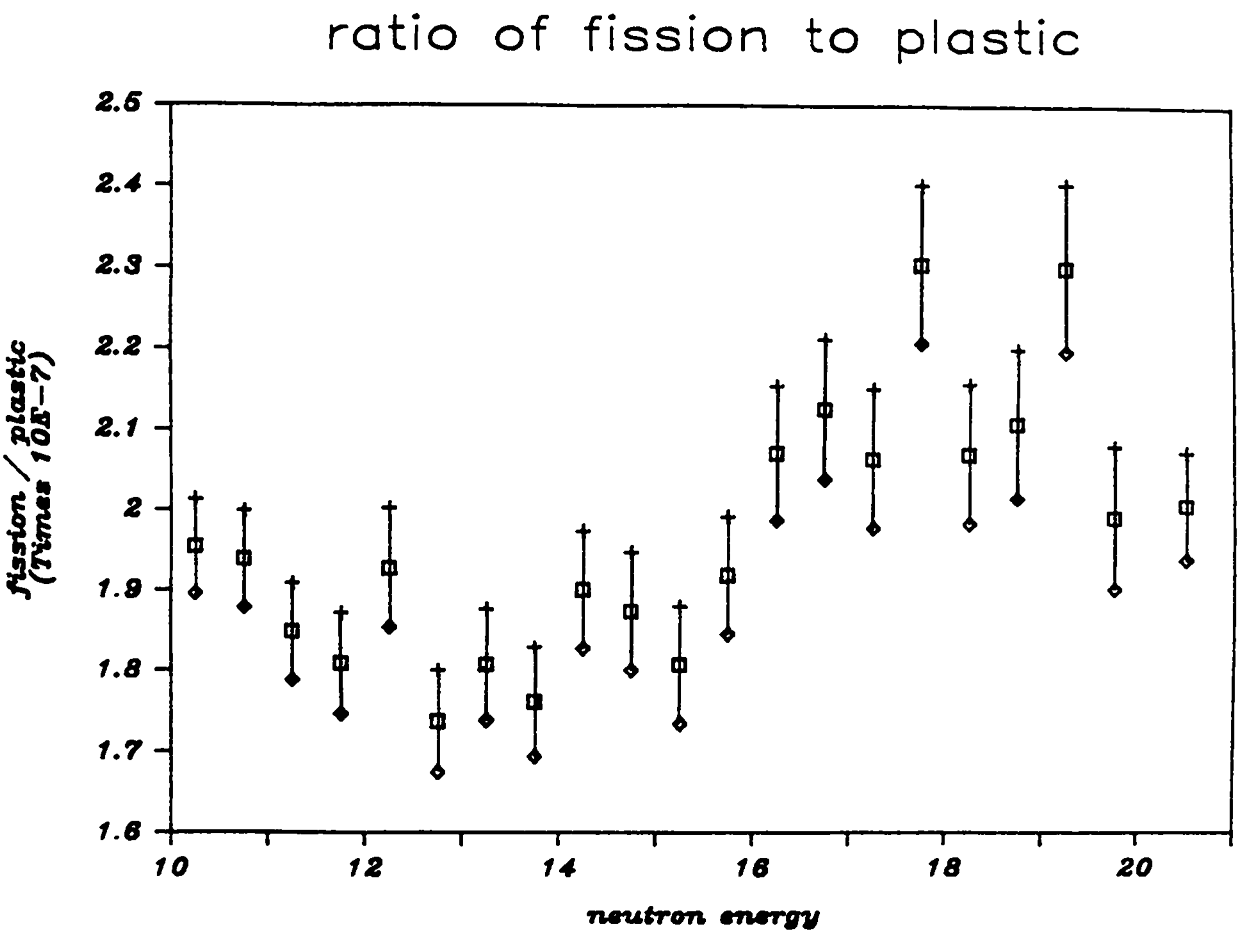

Fig. 4. The ratio of the absolute neutron flux measured using the fission chamber to the relative flux measured using the plastic scintillator in the overlap energy region from 10 to 20 MeV.

To estimate the accuracy of the STANTON efficiency code in the energy range from 125 to 750 MeV, we have calculated the relative efficiency for a rectangular neutron detector (NE-102) 7.5 cm thick and 25.4 cm by 50.8 cm in

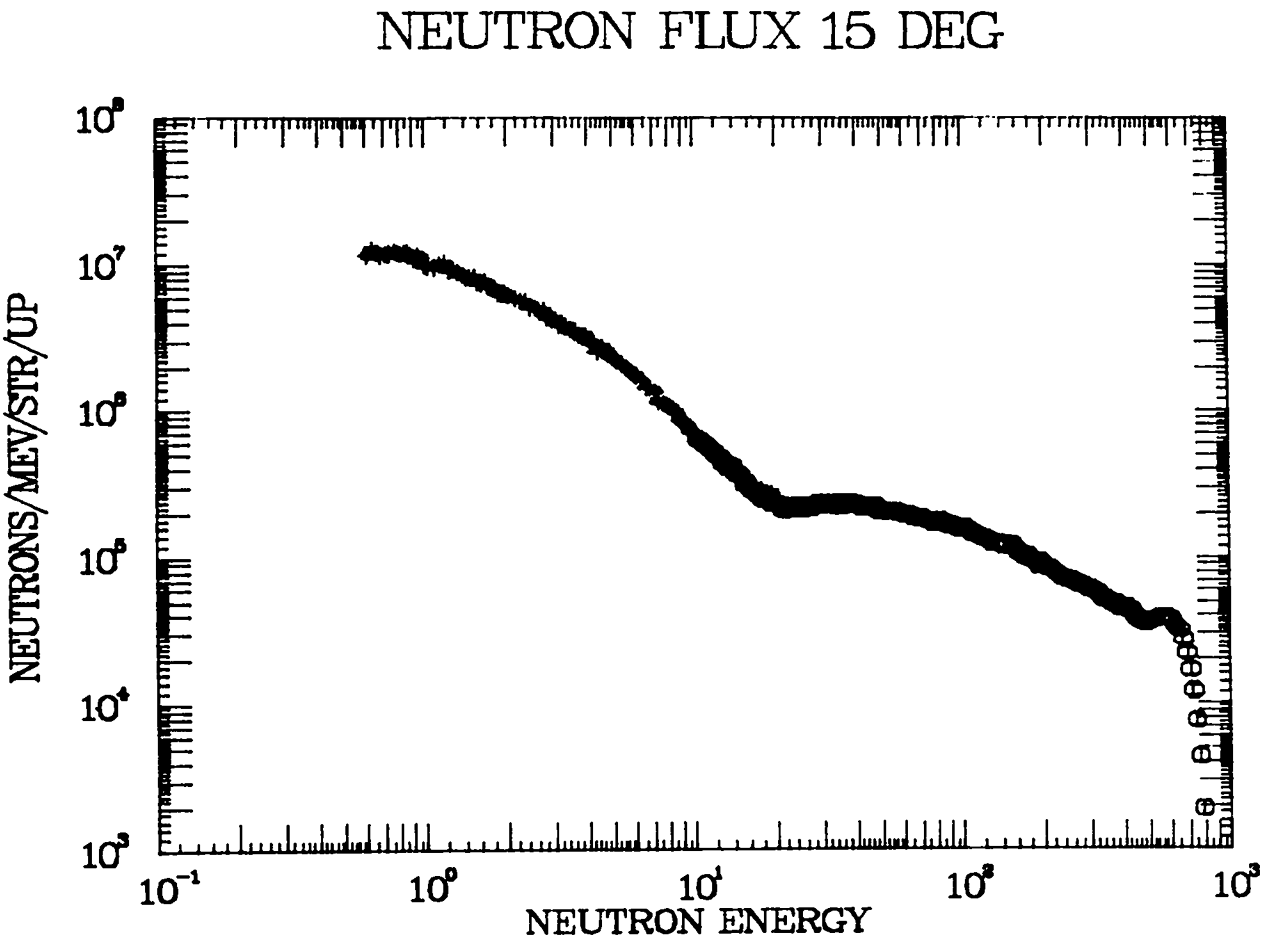

Fig. 5. The neutron flux obtained at the 15° flight paths. These results are based on the known ^{235}U cross section below 20 MeV. The circles are the neutron flux obtained from the plastic scintillator which is normalized to the absolute flux in the overlap energy range from 10 to 20 MeV.

area and compared the calculation with measured values. The measured efficiency has been obtained by comparing the time-of-flight yield for the $^{nat}Pb(p,n)$ reaction to the yield for the same reaction obtained with a spectrometer and normalized to measured NP elastic cross sections[7]. For more details on this method see Ref. 8. Fig. 6 shows the comparison of the two efficiencies. The agreement ranges from better than 15% below 400 MeV to worse than 40% above 700 MeV.

We used this flux (now normalized per μpulse) to calculate the effective fission cross section [$\sigma_{eff}(E)$] for ^{235}U up to 750 MeV using the following relation:

$$\sigma_{eff}(E) = \frac{[D(E) - B]}{\Omega N_a N_{up} \Delta E_n(E) N_n(E)} \quad .$$

D(E) is the number of counts/channel in the fission chamber at neutron energy E. $\Delta E_n(E)$ is the width of the channel in MeV. B is the number of background counts/channel obtained from the data region just prior to the beam pulse. N_a is the number of target atoms/cm^2. $N_n(E)$ is the neutron flux at energy E per

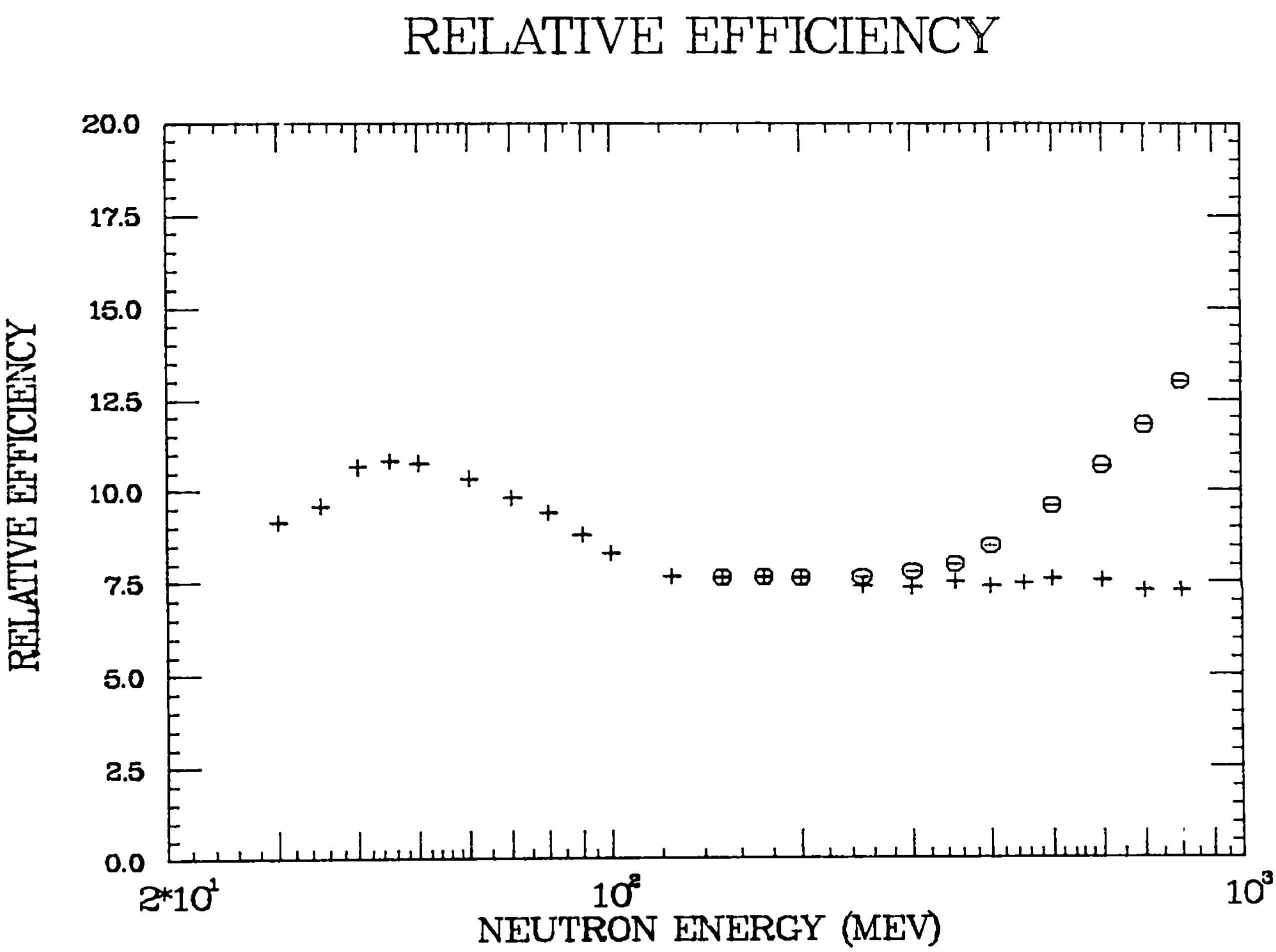

Fig. 6. Measured and calculated (Stanton Code) relative efficiency of a 25.4 cm by 50.8 cm and 7.5 cm deep NE-102 neutron detector. Crosses represent STANTON calculated efficiencies while circles indicate empirical values.

micropulse as described above. The solid angle as described above is Ω. N_{up} is the number of micropulses obtained in the run. Since the foil contained approximately 10% ^{238}U, the ^{235}U fission cross section is related to the measured effective cross section by the following relation:

$$\sigma_f(E) = 1.09\ \sigma_{eff}(E) - 0.094\ \sigma_{238U}(E) \quad .$$

Values for the ^{238}U cross section were obtained from the ENDF evaluation. The ^{235}U fission cross section calculated in this manner, binned into appropriate energy bins, is shown in fig. 7.

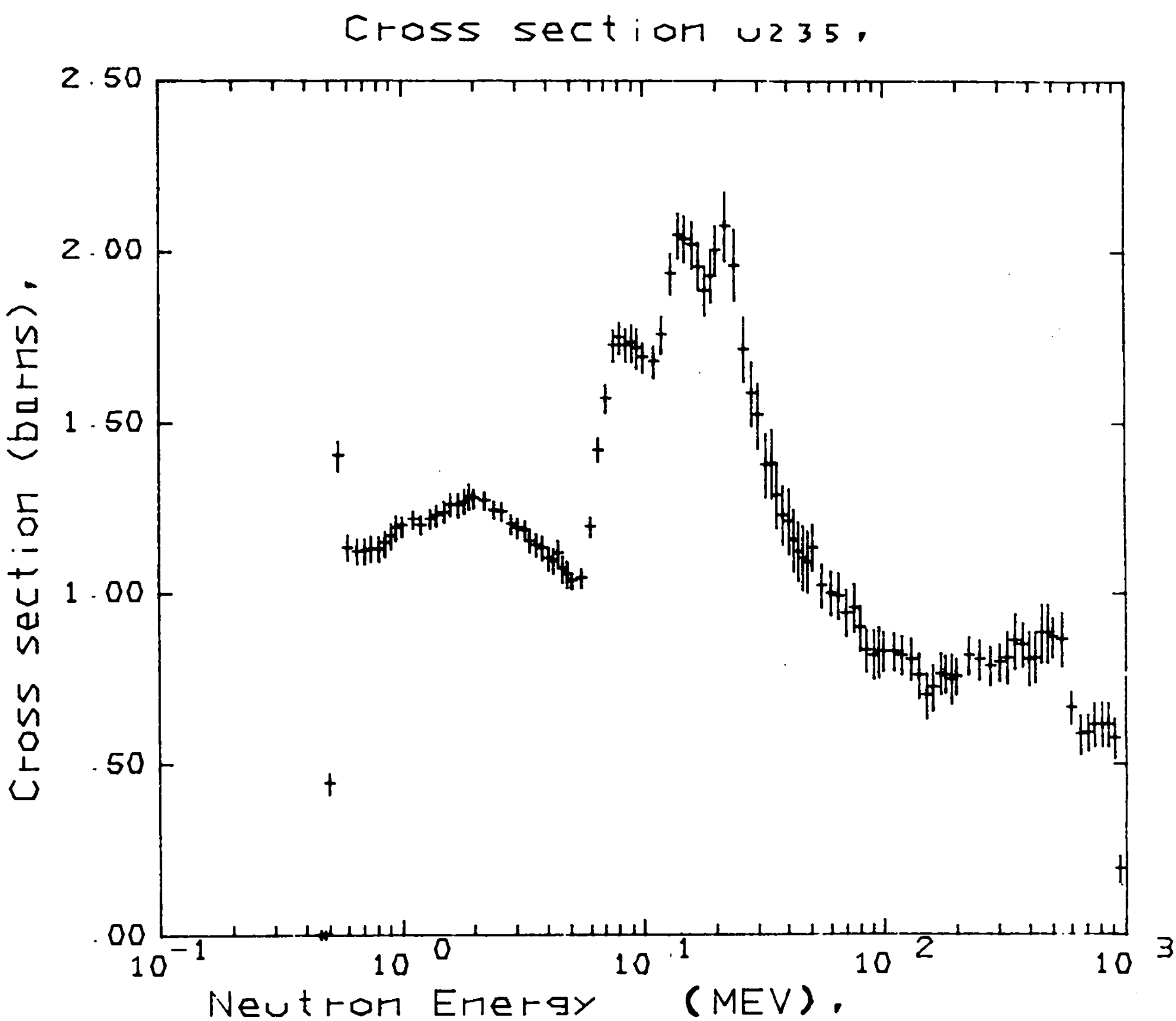

Fig. 7. Measured $^{235}U(n,f)$ cross section from 0.6 to 750 MeV neutron energy.

4. DISCUSSION

In fig. 8 we compare the present $^{235}U(n,f)$ data with other $^{235}U(n,f)$ data[9] (X's) and $^{235}U(p,f)$ data[10] (O's).

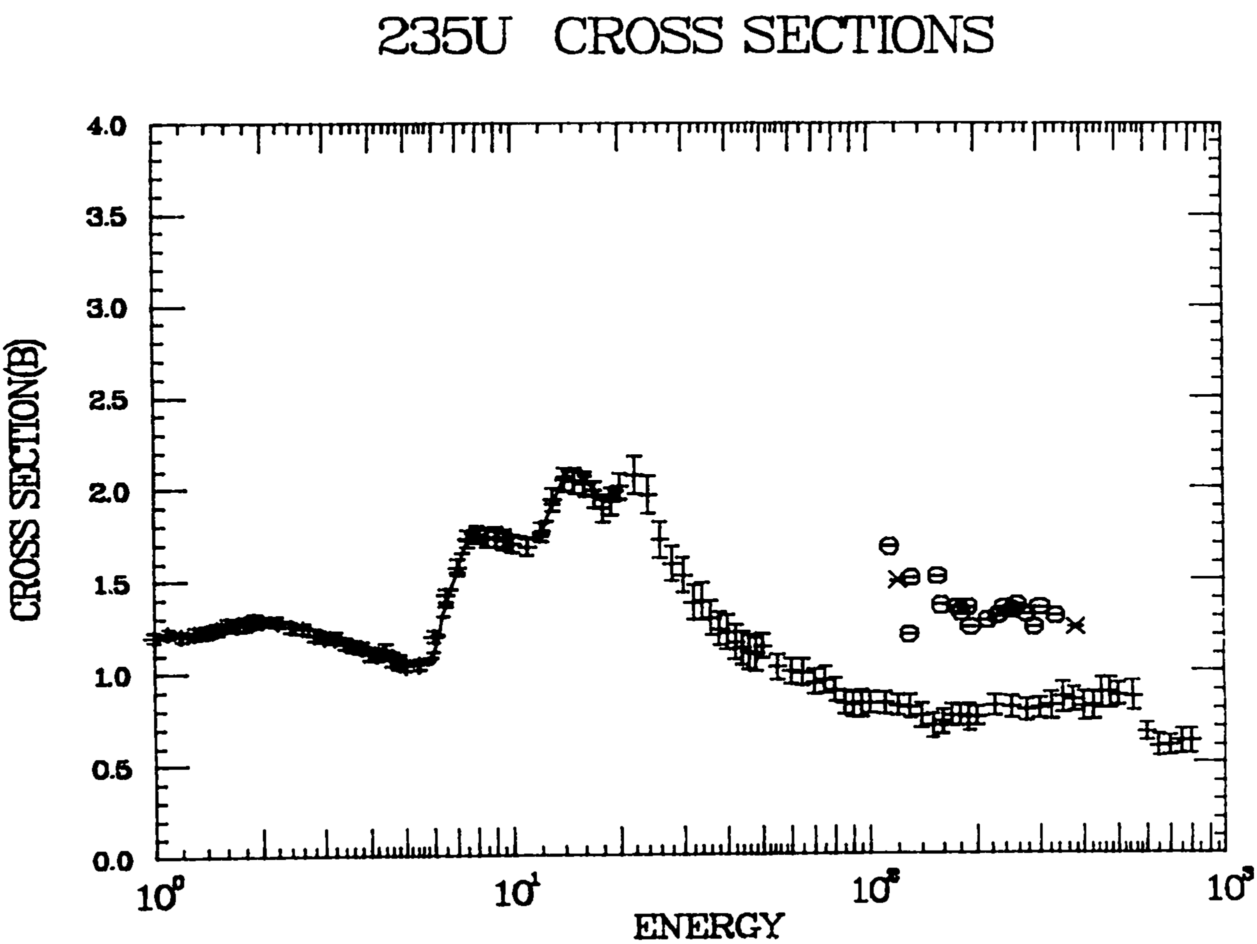

Fig. 8. The (n,f), (p,f) cross section for ^{235}U. The X's are (n,f) data from Ref. 9. The line is from the ENDF evaluation. The O's are (p,f) data from Ref. 10. The present experiment is shown as data points with errors.

In fig. 9 we present similar data for ^{238}U. The (n,f) data up to 36 MeV neutron energy are from Ref. 2 (continuous line) while the data points at 84 MeV, 120 MeV, and 380 MeV (X's) are from Ref. 9. The $^{238}U(p,f)$ data from 10 to 340 MeV protons (O's) are from Ref. 11, from 100 to 340 MeV (+'s) from Ref. 10 and from 0.6 to 2.9 GeV (Δ's) from Ref. 12.

Above 100 MeV the (p,f) cross sections for ^{235}U and ^{238}U have about the same values (1.4 barns). On the other hand, the (n,f) cross sections for

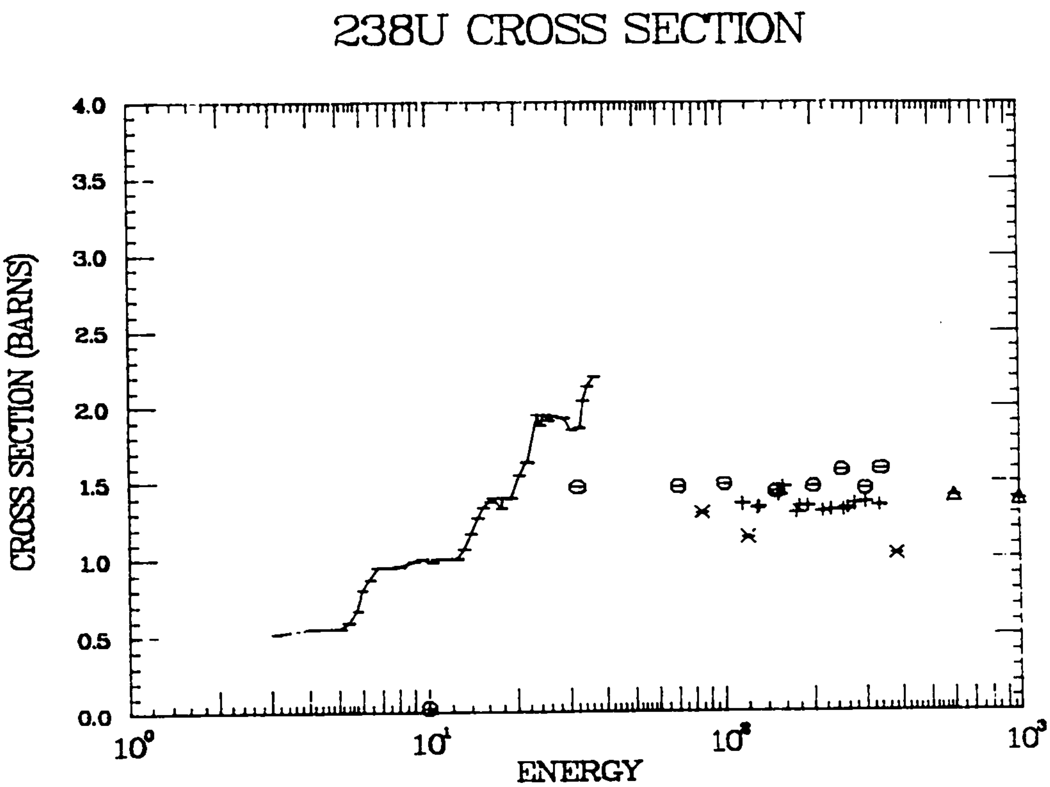

Fig. 9. The (n,f), (p,f) cross section for ^{238}U. The X's are (n,f) data from Ref. 9. The +'s, O's, and Δ's are (p,f) data from Ref. 10, Ref. 11, and Ref. 12 respectively. The line is from Ref. 2.

these isotopes have, within error bars, values of about 1.0 barn in this energy region. Preliminary calculations done by M. Blann* using the evaporation code ALICE/LIVERMORE/85 at 200 MeV incident nucleon energy, indicate that the ratio of the ^{235}U fission width to the total width induced by neutrons is 0.0586 while that induced by protons is 0.0887. The ratio of these two values is 1.5 in good agreement with the experimentally observed ratio.

5. CONCLUSIONS

The results reported here extend the data on the $^{235}U(n,f)$ cross section to 750 MeV. We would like to emphasize that these data were taken in less than 12 hours of beam time and the results are based on the efficiency code

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*Information provided by M. Blann, Physics Division, Lawrence Livermore National Laboratory (June, 1987)

STANTON. The reliance on the efficiency code above 200 MeV is probably our greatest source of uncertainty.

During the next run cycle we plan to measure the efficiency by tagging the detected neutrons to the recoil protons from the H(n,p) reaction. This method will tie the fission cross section to the better known (n,p) cross sections.

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