KFKI-75-21



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*Hungarian Academy of Sciences*

CENTRAL RESEARCH INSTITUTE FOR PHYSICS

BUDAPEST

2017

KFKI-75-21

# MONTE CARLO CALCULATED SPECTRA OF NEUTRONS TRANSMITTED THROUGH AND REFLECTED FROM HOMOGENEOUS POLYETHYLENE SLABS

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February 1975

Work supported by the
International Atomic Energy Agency
under Research Contract No. 1115/RB
and Research Agreement No. 889/Rl/CF

ISBN 963 371 020 0

## ABSTRACT

Transmission and reflection of 14.5 MeV and fission neutrons are calculated for polyethylene shields of thicknesses from 5 to 40 cm. The 48 group spectra are calculated by the Monte Carlo code O5R5S and plotted by the code TRESSPASS. Characteristic quantities of the spectra, average energies, thermal and fast fractions, as well as the transmission or reflection probabilities are also given.



## АННОТАЦИЯ

Даются спектры нейтронов, прошедших через однородные слои полиэтилена или отраженных от них. 48-групповые спектры были вычислены с помощью программы O5RS5S Monte Carlo и вычерчены с помощью программы TRESPASS. Для каждого спектра приведены вероятности и прохождения или отражения, средняя энергия, а также доля тепловых и быстрых нейтронов.



## KIVONAT

Homogén polietilén rétegeken áthaladt, illetve azokról visszavert neutronok spektrumát közöljük. A 48 csopotos spektrumokat az O5R5S Monte Carlo programmal számoltuk és a TRESPASS programmal rajzoltattuk fel. Minden spektrumra megadjuk az áthaladási vagy visszaverődési valószinüséget, az átlagenergiát és a gyors, illetve termikus hányadokat.

## 1. Introduction

In neutron dosimeter evaluation one of the most critical points is the knowledge of the spectrum of the neutrons. As a measurement of the spectrum in every case is practically impossible a compendium of spectra calculated for and/or measured in typical situations (typical shield materials, thicknesses, input spectra and geometries) could well be used.

The International Atomic Energy Agency (IAEA) supports this work in which our Institute participates, on the basis of Research Contract No. 1115/RB and Research Agreement No. 889/R1/CF.

We have developed a special version of the O5R program for the calculation of the spectra of neutrons transmitted through or reflected from different homogeneous slab shields. This code, the O5R5S (Koblinger, 1974) prints and punches out the spectra.
Two other codes, the TRESPASS (Pálfalvi, 1974) and SPECTRANS-2 (Pálfalvi, 1973) plot the computed spectra and calculate some of their characteristic quantities.

Some spectra calculated for water shields were published earlier (Pálfalvi, Koblinger, 1974), in the present report results obtained for polyethylene shields are given.

Although for dosimeter evaluation only the shape of a spectrum is interesting and not the attenuation, all the quantities calculated are presented here as it is hoped that our results can be used in other fields.

## 2. Comments on the calculations

The O5R5S calculates the spectra by Monte Carlo technique using the collision density method, i.e. the transmission and reflection probabilities are determined after each scattering, regarding the incidence of a neutron as the $0^{th}$ scattering. This method results in better statistics in comparison with the analysis of the really escaping neutrons.

The 05R5S prints and punches out the number of the transmitted or reflected neutrons in 49 energy groups. The energy limits and the mean energies for 48 groups are given in Table 1, the $0^{th}$ contains the thermal neutrons. The coefficients of variation are also calculated and edited for every group.

Details of the calculation method are given in the description of the 05R5S code (Koblinger, 1974).

The calculations were performed using the following parameters:

a/ for cross section handling, the energy supergroups of the 05R code were divided into 128 subgroups (for details: see Lux, Koblinger, 1973);

b/ the cutoff energy under which the neutrons are considered as thermal, was 0.5 eV;

c/ for thermal neutrons the non-absorption probability was set to 0.99437, the mean free path length was chosen to be 0.2494 cm. These values were calculated by the code THERMOS (Gadó, 1973).

d/ the scattering angular distribution for the hydrogen was assumed to be isotropic, whereas for the carbon the distribution was approximated by a Legendre expansion of 6 terms. The Legendre coefficients are given for 64 subgroups in every supergroup.

## 3. Comments on plotting

From the 05R5S results code TRESPASS determines the $\varphi(u) = E \times \Phi(E)$ spectra (neutrons per unit lethargy interval) normalized to unit incident neutron.

Table 1

| | STANDARD ENERGY EV | LIMITS OF THE ENERGY GROUPS E EV | E EV | LETHARGY INTERVALS |
|---|---|---|---|---|
| 1 | 2.17010E-01 | 1.88450E-01 | 2.50000E-01 | 0.2830 |
| 2 | 3.53560E-01 | 2.50000E-01 | 5.00000E-01 | 0.6930 |
| 3 | 7.07150E-01 | 5.00000E-01 | 1.00000E 00 | 0.6930 |
| 4 | 1.46630E 00 | 1.00000E 00 | 2.15000E 00 | 0.7660 |
| 5 | 3.16190E 00 | 2.15000E 00 | 4.65000E 00 | 0.7710 |
| 6 | 6.81910E 00 | 4.65000E 00 | 1.00000E 01 | 0.7660 |
| 7 | 1.46630E 01 | 1.00000E 01 | 2.15000E 01 | 0.7660 |
| 8 | 3.16190E 01 | 2.15000E 01 | 4.65000E 01 | 0.7710 |
| 9 | 6.81910E 01 | 4.65000E 01 | 1.00000E 02 | 0.7660 |
| 10 | 1.46630E 02 | 1.00000E 02 | 2.15000E 02 | 0.7660 |
| 11 | 3.16190E 02 | 2.15000E 02 | 4.65000E 02 | 0.7710 |
| 12 | 6.81910E 02 | 4.65000E 02 | 1.00000E 03 | 0.7660 |
| 13 | 1.46630E 03 | 1.00000E 03 | 2.15000E 03 | 0.7660 |
| 14 | 3.16190E 03 | 2.15000E 03 | 4.65000E 03 | 0.7710 |
| 15 | 6.81910E 03 | 4.65000E 03 | 1.00000E 04 | 0.7660 |
| 16 | 1.12200E 04 | 1.00000E 04 | 1.25890E 04 | 0.2300 |
| 17 | 1.41250E 04 | 1.25890E 04 | 1.58480E 04 | 0.2300 |
| 18 | 1.78160E 04 | 1.58480E 04 | 1.99510E 04 | 0.2300 |
| 19 | 2.23850E 04 | 1.99510E 04 | 2.51170E 04 | 0.2300 |
| 20 | 2.81820E 04 | 2.51170E 04 | 3.16200E 04 | 0.2300 |
| 21 | 3.54780E 04 | 3.16200E 04 | 3.98050E 04 | 0.2300 |
| 22 | 4.46630E 04 | 3.98050E 04 | 5.01120E 04 | 0.2300 |
| 23 | 5.62260E 04 | 5.01120E 04 | 6.30860E 04 | 0.2300 |
| 24 | 7.07820E 04 | 6.30860E 04 | 7.94180E 04 | 0.2300 |
| 25 | 8.91170E 04 | 7.94180E 04 | 1.00000E 05 | 0.2300 |
| 26 | 1.12200E 05 | 1.00000E 05 | 1.25890E 05 | 0.2300 |
| 27 | 1.41250E 05 | 1.25890E 05 | 1.58480E 05 | 0.2300 |
| 28 | 1.78160E 05 | 1.58480E 05 | 1.99510E 05 | 0.2300 |
| 29 | 2.23850E 05 | 1.99510E 05 | 2.51170E 05 | 0.2300 |
| 30 | 2.81820E 05 | 2.51170E 05 | 3.16200E 05 | 0.2300 |
| 31 | 3.54780E 05 | 3.16200E 05 | 3.98060E 05 | 0.2300 |
| 32 | 4.46630E 05 | 3.98060E 05 | 5.01120E 05 | 0.2300 |
| 33 | 5.62260E 05 | 5.01120E 05 | 6.30860E 05 | 0.2300 |
| 34 | 7.07820E 05 | 6.30860E 05 | 7.94180E 05 | 0.2300 |
| 35 | 8.91170E 05 | 7.94180E 05 | 1.00000E 06 | 0.2300 |
| 36 | 1.12200E 06 | 1.00000E 06 | 1.25890E 06 | 0.2300 |
| 37 | 1.41250E 06 | 1.25890E 06 | 1.58480E 06 | 0.2300 |
| 38 | 1.78160E 06 | 1.58480E 06 | 1.99510E 06 | 0.2300 |
| 39 | 2.23850E 06 | 1.99510E 06 | 2.51170E 06 | 0.2300 |
| 40 | 2.81820E 06 | 2.51170E 06 | 3.16200E 06 | 0.2300 |
| 41 | 3.54780E 06 | 3.16200E 06 | 3.98060E 06 | 0.2300 |
| 42 | 4.46630E 06 | 3.98060E 06 | 5.01120E 06 | 0.2300 |
| 43 | 5.62260E 06 | 5.01120E 06 | 6.30860E 06 | 0.2300 |
| 44 | 7.07820E 06 | 6.30860E 06 | 7.94180E 06 | 0.2300 |
| 45 | 8.91170E 06 | 7.94180E 06 | 1.00000E 07 | 0.2300 |
| 46 | 1.12200E 07 | 1.00000E 07 | 1.25890E 07 | 0.2300 |
| 47 | 1.41250E 07 | 1.25890E 07 | 1.58480E 07 | 0.2300 |
| 48 | 1.78160E 07 | 1.58480E 07 | 1.99510E 07 | 0.2300 |

It should be noted that by this normalization only

$$\int_{u_{min}}^{u_{max}} \varphi(u)\,du < 1 = \int_{u_{min}}^{u_{max}} \varphi_{inc}(u)\,du$$

is satisfied but for a given group $\varphi(u_k)$ may exceed 1.

Before plotting the spectra the following two transformations are carried out, if necessary:

a/ If monoenergetic incident neutrons are considered the upper limit of the last energy interval is replaced by the source energy as there are no neutrons with energies higher than this value. If the new last energy interval obtained by this method is shorter than one tenth of the original interval the last and penultimate intervals are united.

b/ In the case of plotting of thermal neutrons, their distribution is assumed to be Maxwellian. The peak of the $\varphi(u)$ distribution is at E=1.5 kT (=0.0379 eV) and the differential fluence at this point is

$$\frac{2}{\sqrt{\pi}}\left(\frac{1.5}{e}\right)^{\frac{3}{2}} = 0.463$$

times the fluence of the thermal group.
(The real distribution slightly differs from the Maxwellian but generally neither the location nor the height of the peak is shifted by more than 4-5 per cent, therefore if this minor effect had been taken into account the plotting procedure would have been unnecessarily complicated.)

The code plots the spectra as step functions marking the standard deviation also. The thermal peak is represented by an "X".

## 4. Results and Conclusions

Runs have been carried out for three incident sources:

a/ monoenergetic source of 14.5 MeV, cosine angular distribution,

b/ monoenergetic source of 14.5 MeV, perpendicular incidence,

c/ fission source: the energy distribution is given by the Watt formula (cosine angular distribution).

For all the three cases shield thicknesses of 5, 10, 20 and 40 cm are considered. Thermal neutrons are treated only for 5 and 10 cm thick slabs to save running time, which increases by a factor of two even in the case of thickness of 10 cm and grows rapidly with increasing thickness.

The spectra are given in Figs 1-24.

The statistics for the transmitted neutrons worsen if the thickness increases or the incident energy decreases. For instance, the time spent in computing of the transmission of fission neutrons through 40 cm was more than 4 times higher than that spent for computing neutrons of 14.5 MeV but the statistics are poorer for the fission neutrons (see Fig. 23). It must be mentioned here that the uncertainty decreases if fewer energy groups are used. This effect is illustrated in Fig. 23a, where the mean of 3 flux values is taken.

The following characteristic data are calculated by code SPECTRANS-2:

a/ transmitted or reflected fraction: $N_T/N$ or $N_R/N$, where N is the number of incident neutrons, $N_T$ and $N_R$ are the numbers of neutrons transmitted and reflected, respectively;

b/ average energy:

$$\bar{E} = \sum_{k=0}^{48} N_k E_k \Big/ \sum_{k=0}^{48} N_k ,$$

where $N_k$ is the number of neutrons and $E_k$ is the standard (mean) energy for the $k^{th}$ group (for thermal neutrons, the average energy is $E_o = 0.0402$ eV - calculated from the results of the code THERMOS);

c/ fast neutron fraction of the transmitted or reflected neutrons: $N_F/N_T$ or $N_F/N_R$, ($N_F$ is the number of neutrons with energies higher than 2.5 MeV; considered as fast neutrons);

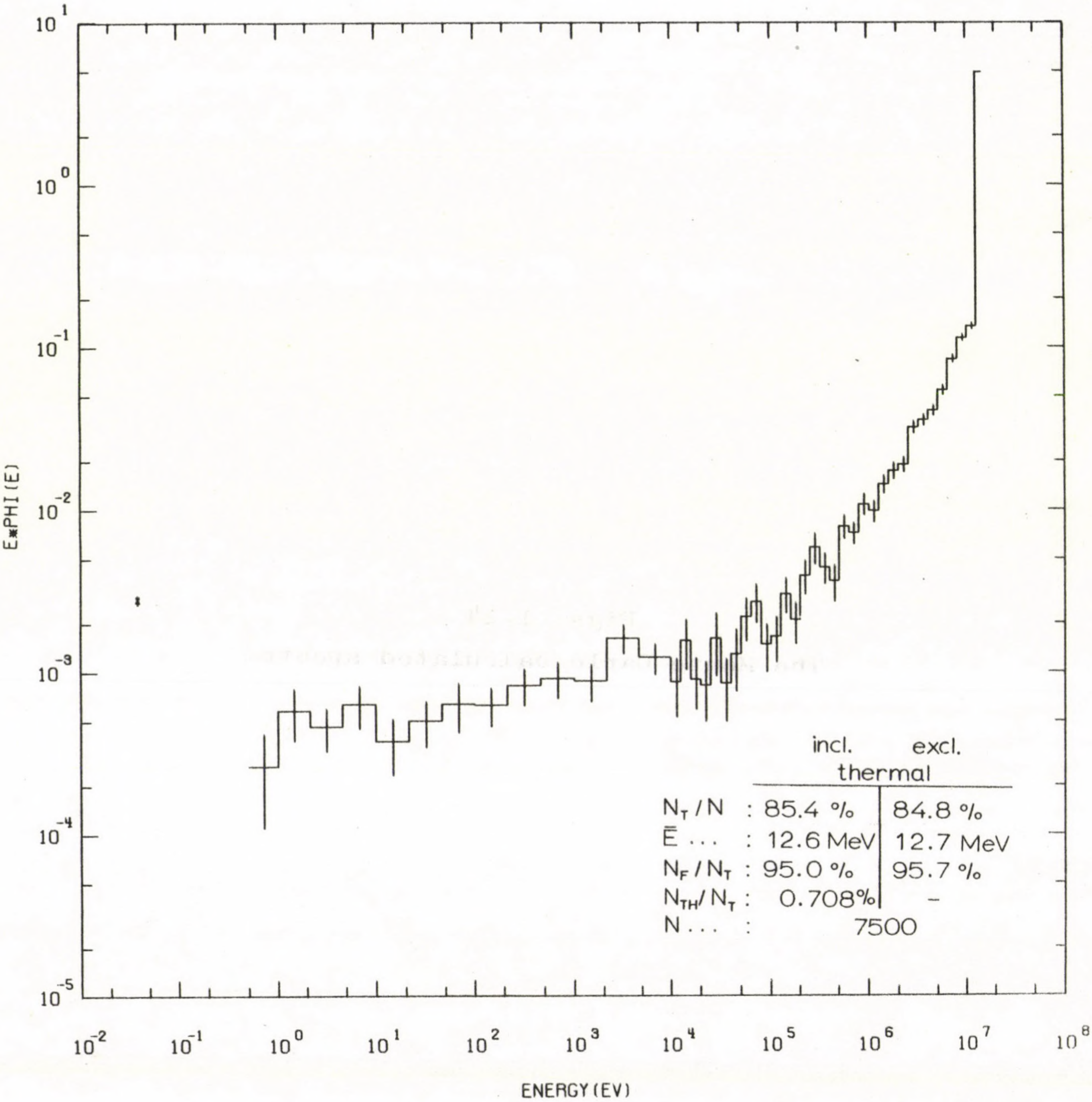
d/ thermal fraction of the transmitted or reflected neutrons: $N_{TH}/N_T$ or $N_{TH}/N_R$, ($N_{TH}$ is the number of the thermal neutrons).

For 5 and 10 cm cases where also thermal neutrons are calculated these data are computed both including and excluding the thermal neutrons. The latter set of values can be used for comparison with data of other thicknesses where thermal neutrons were not calculated.

The characteristic data along with the number of incident neutrons N (which has no physical meaning but is interesting from the point of view of computation) are given in each figure (Figs 1-24). Some of the characteristic data are plotted vs slab thickness in Figs 25-28.

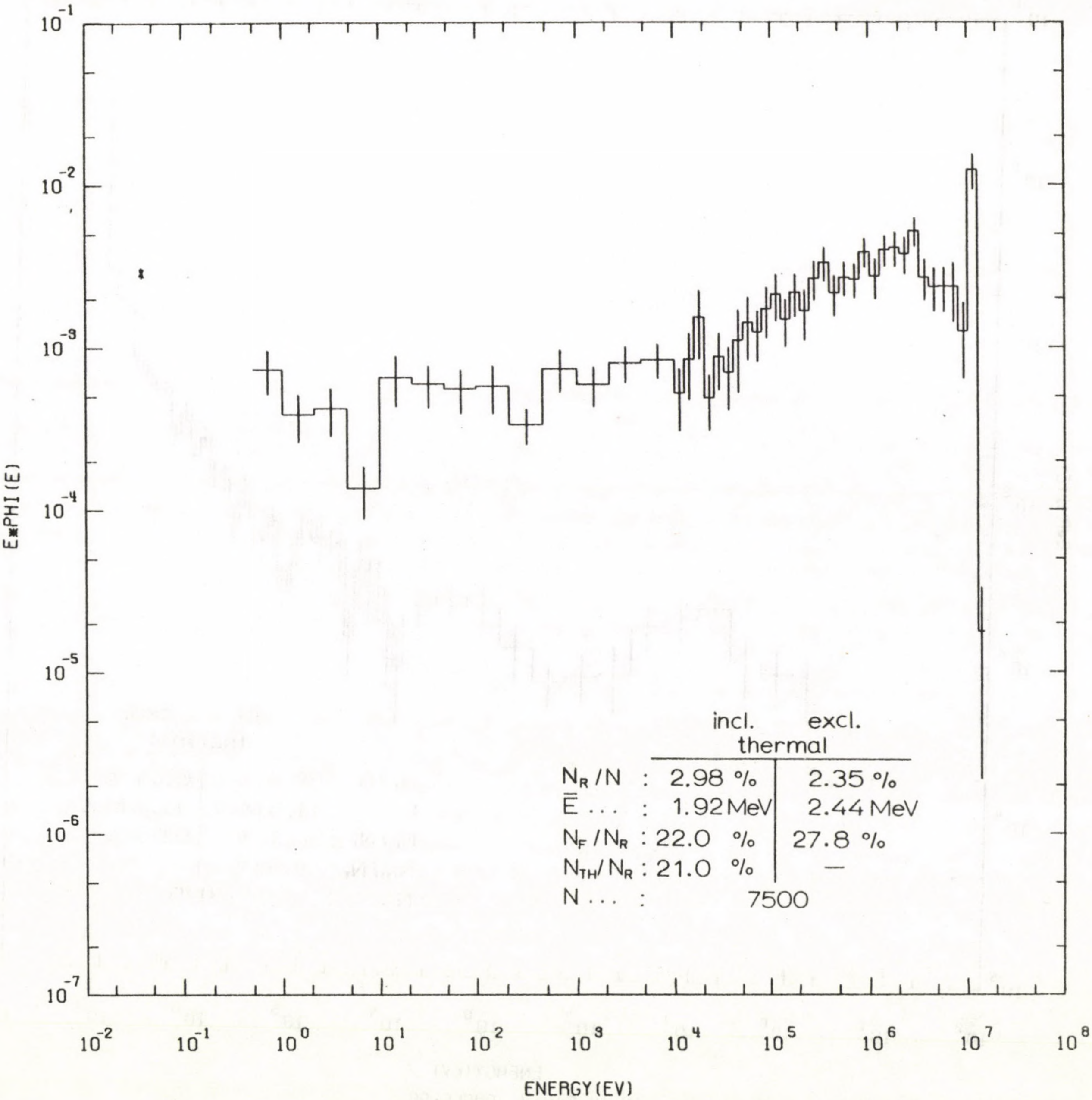
Figs 1-24

The Monte Carlo calculated spectra




TRANS. 5.0 CM PE EIN=14.5 MEV. ANGLE:90

F. 1

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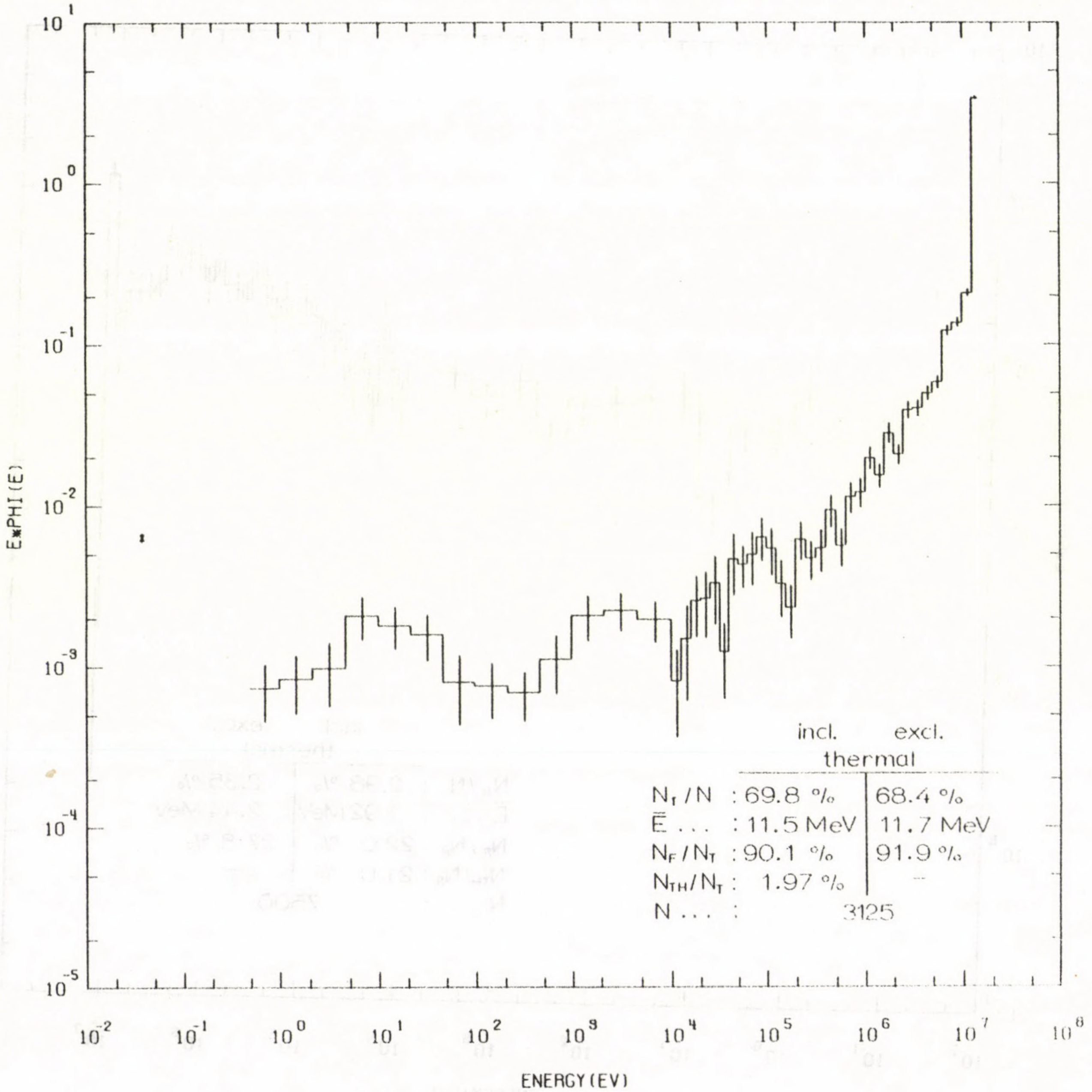
F. 2

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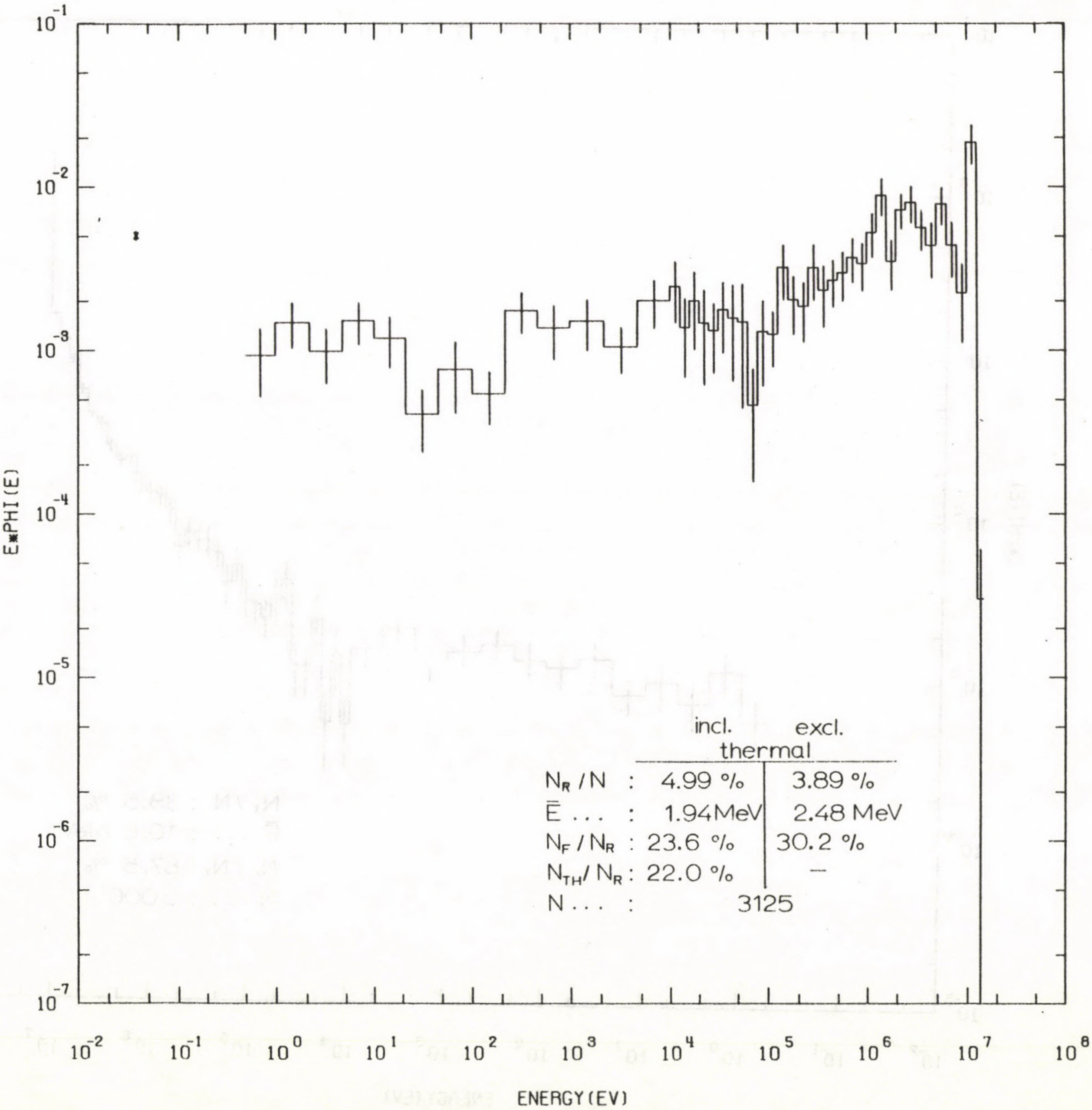
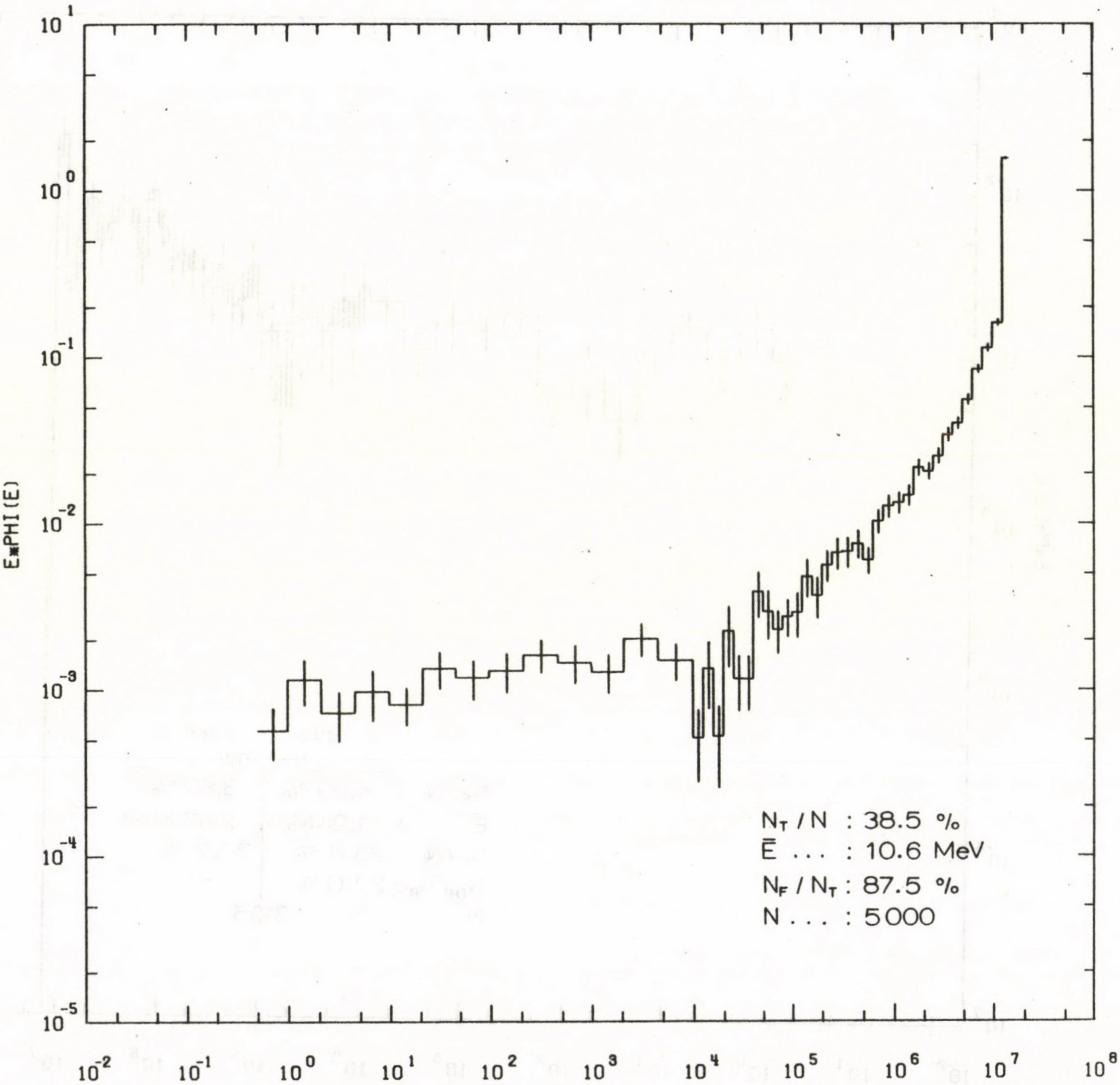
F. 3



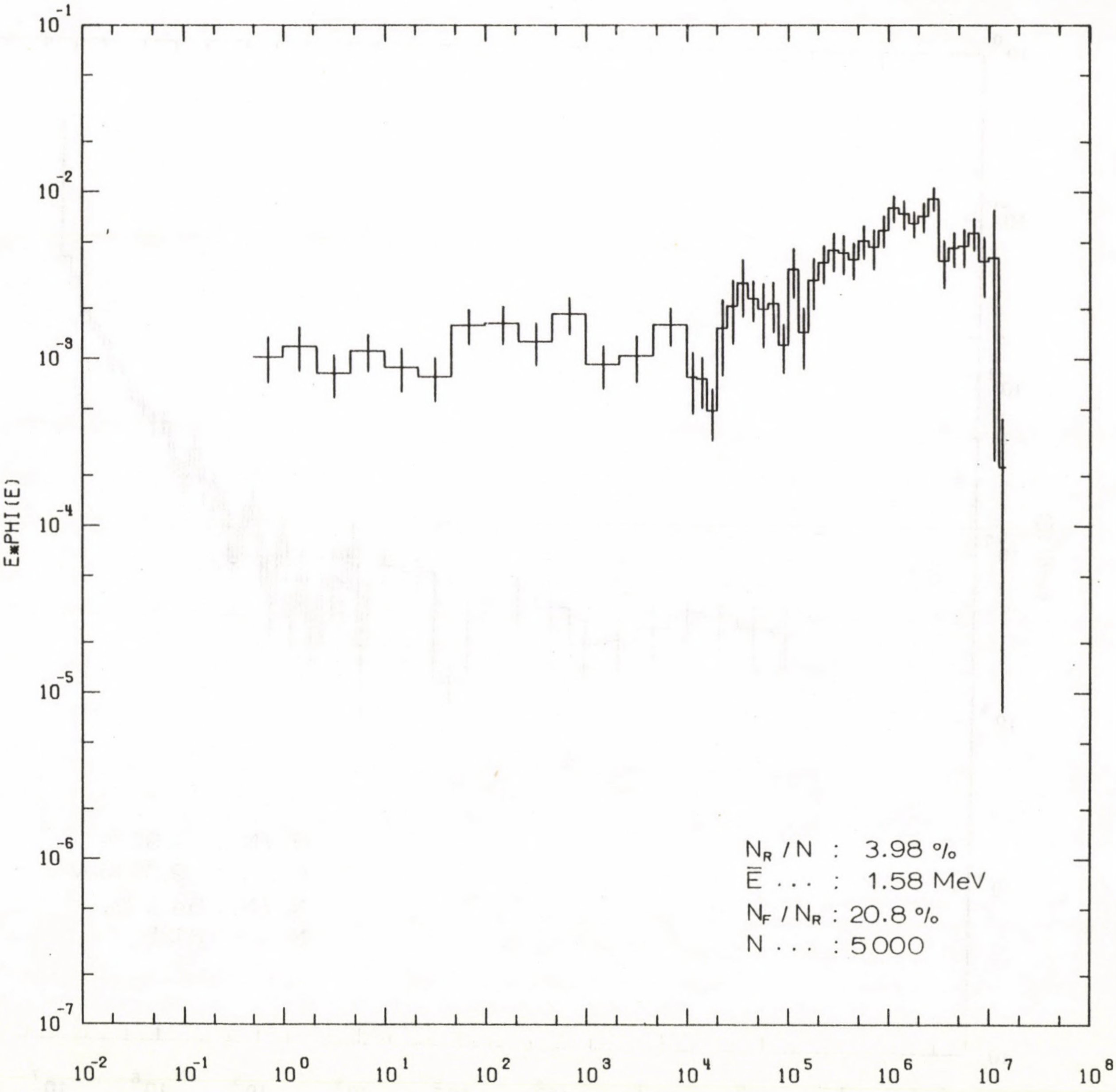

REFL. 10.0 CM PE EIN=14.5 MEV, ANGLE:90

F. 4



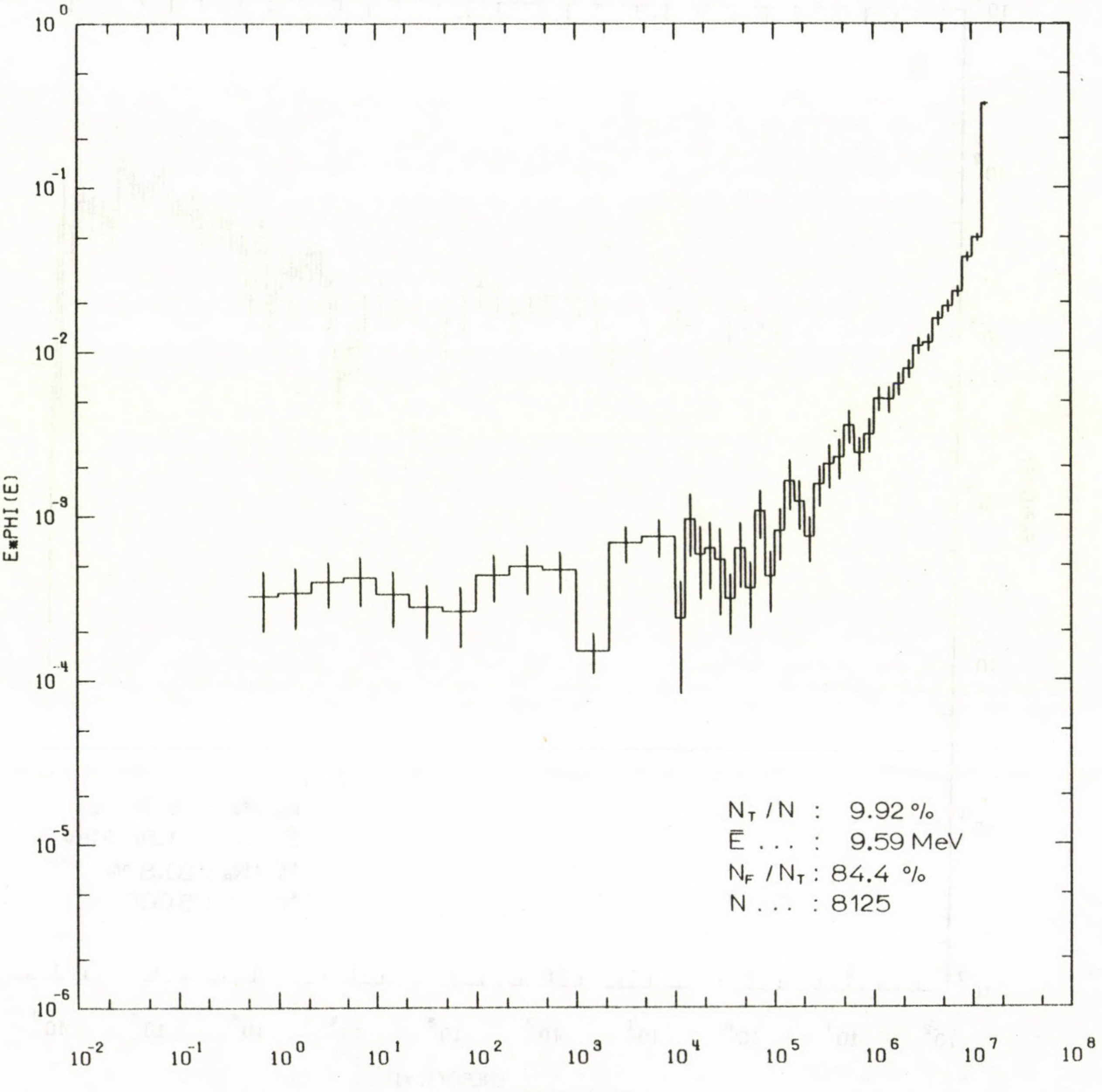

TRANS. 20.0 CM PE EIN=14.5 MEV, ANGLE:90

F. 5



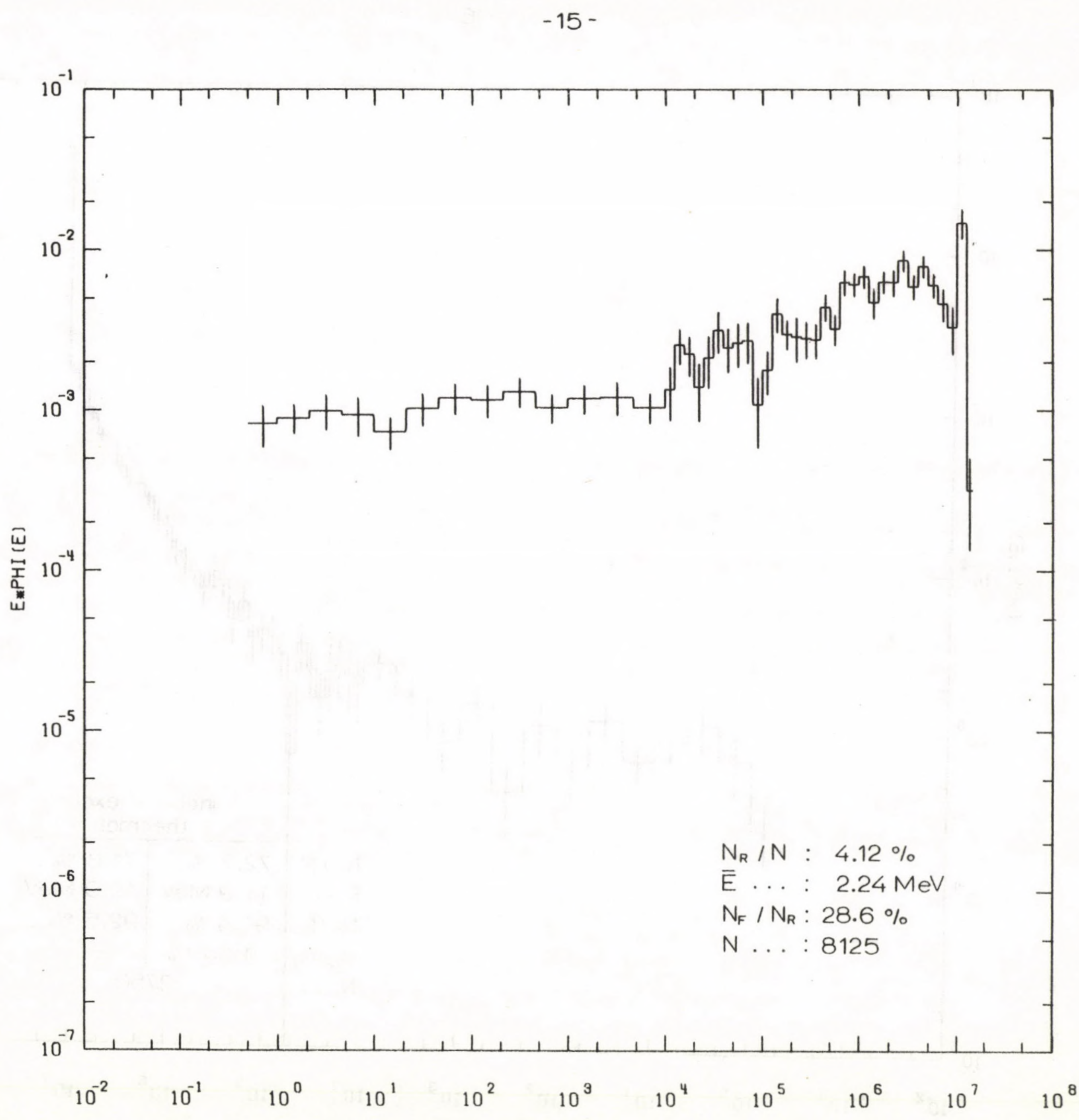

REFL. 20.0 CM PE EIN=14.5 MEV. ANGLE:90

F. 6



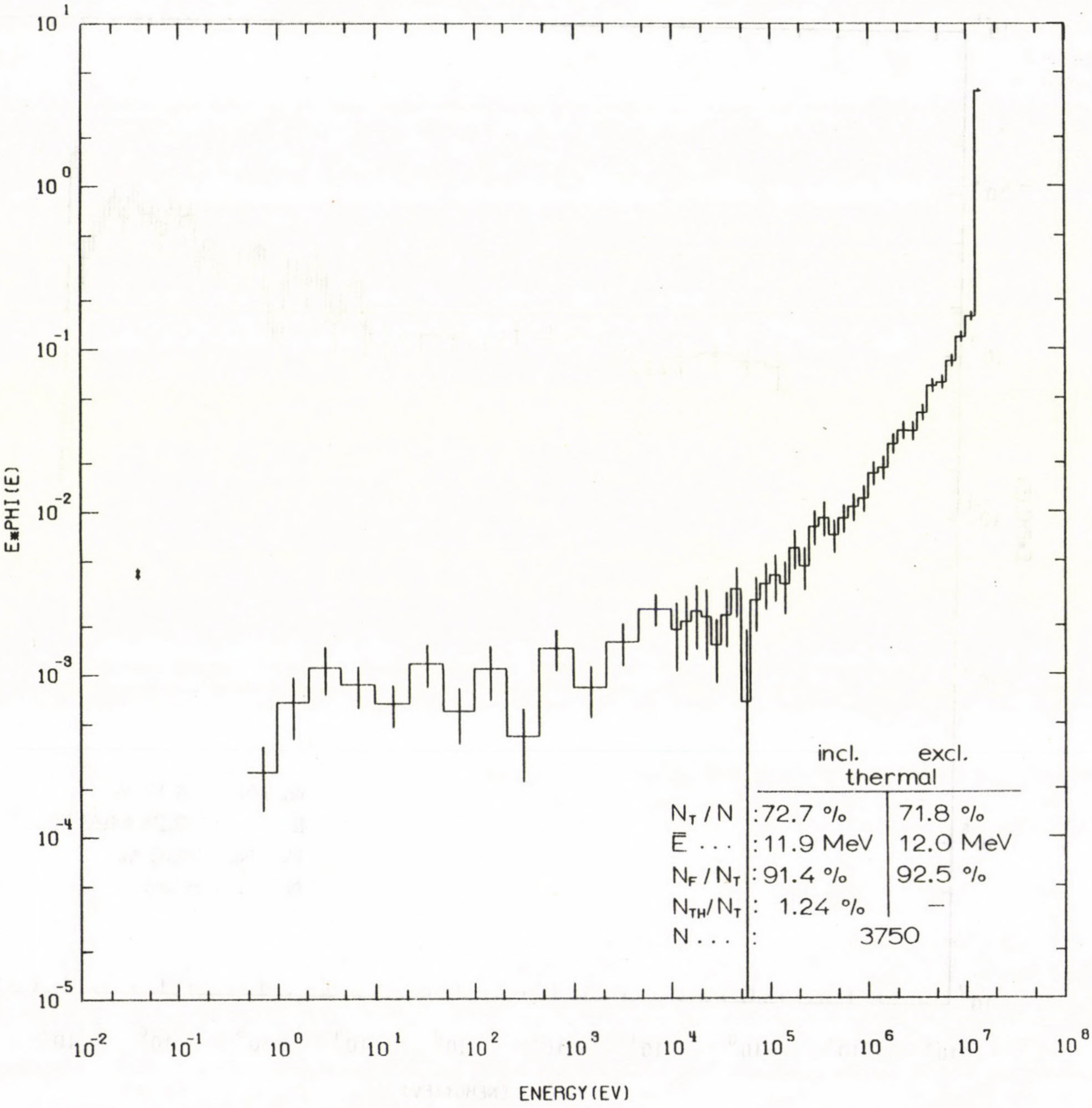

TRANS. 40.0 CM PE EIN=14.5 MEV, ANGLE:90

F. 7



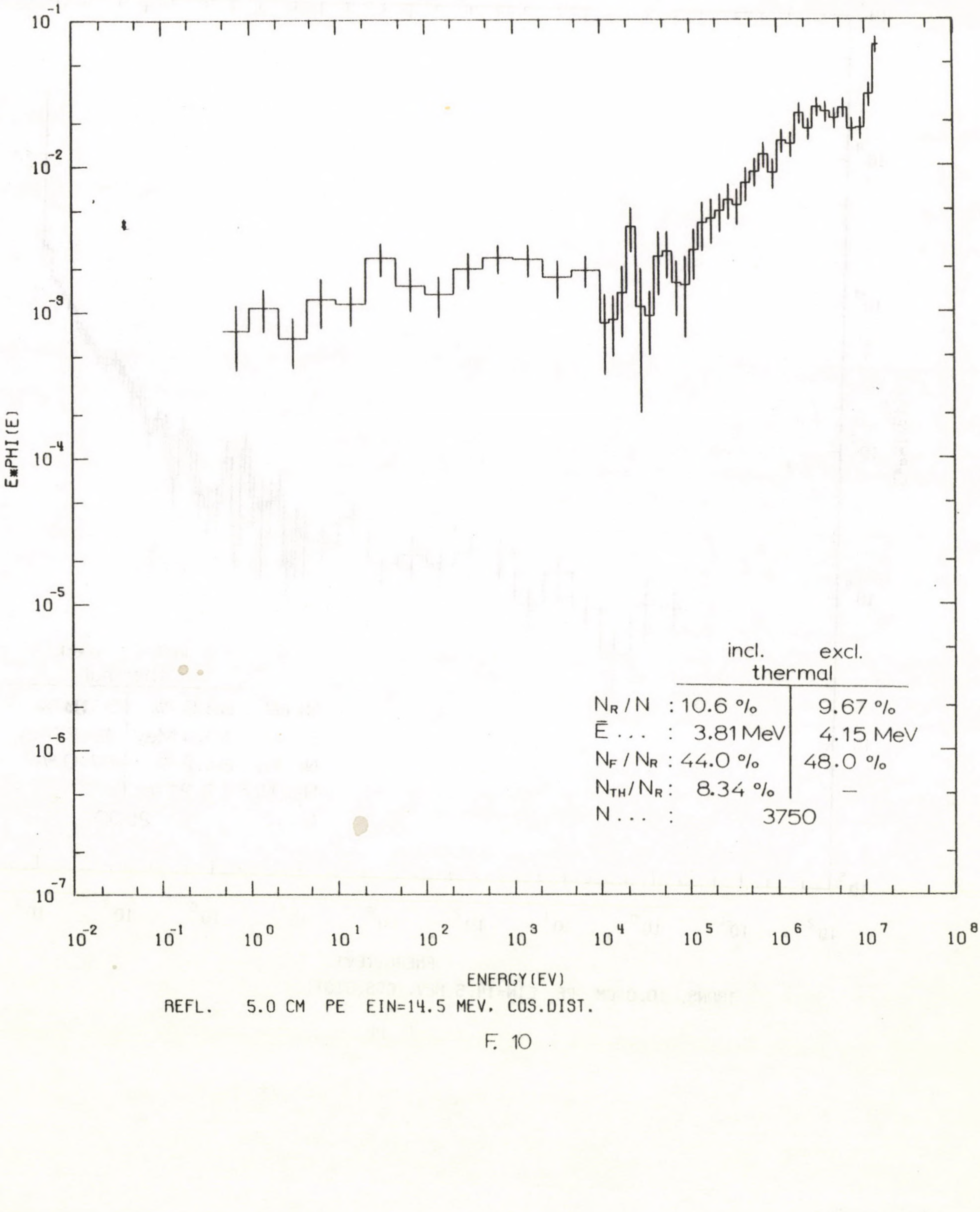

REFL. 40.0 CM PE EIN=14.5 MEV, ANGLE:90

F. 8




TRANS. 5.0 CM PE EIN=14.5 MEV. COS.DIST.

F. 9

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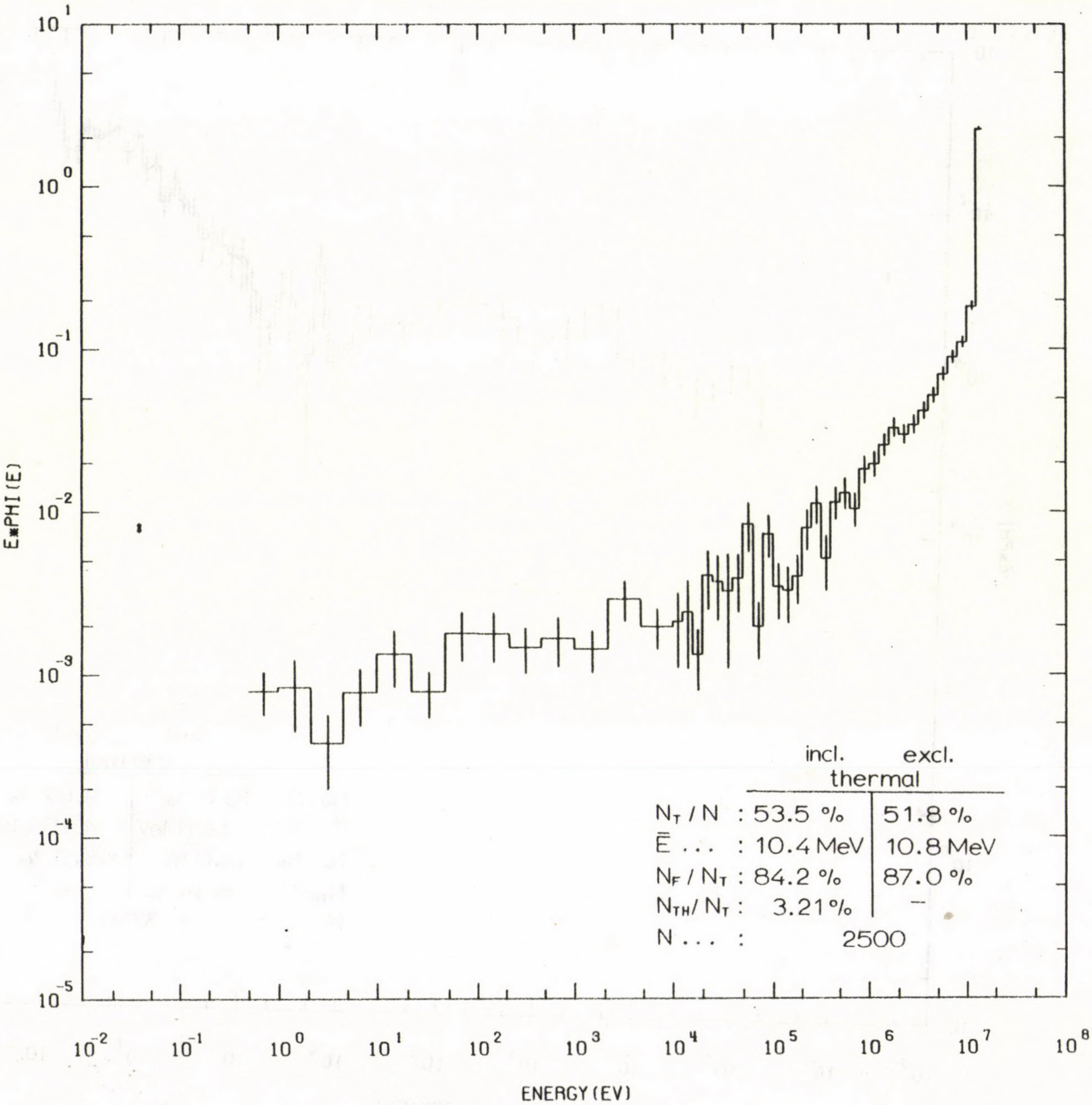
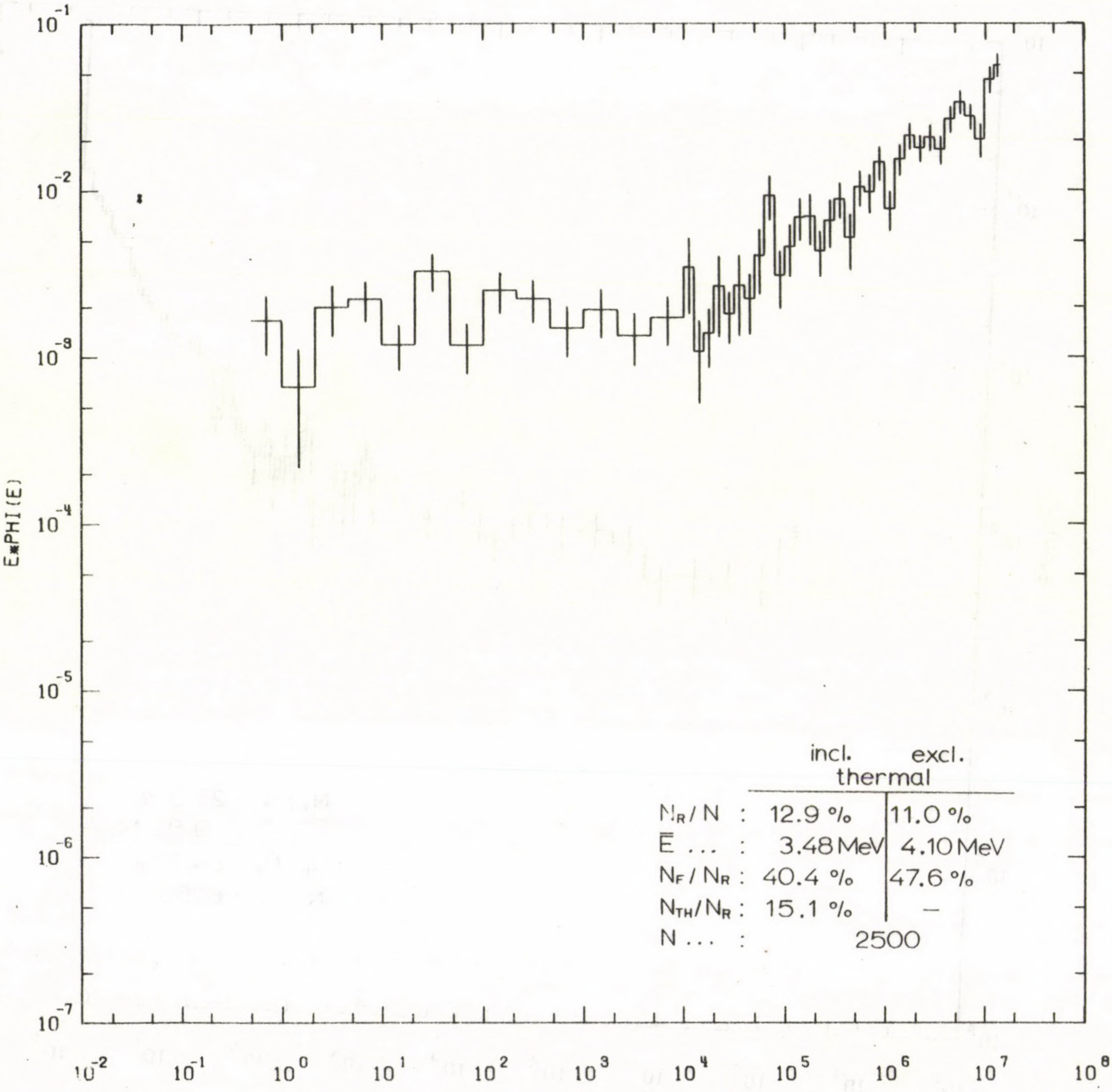
F. 10




TRANS. 10.0 CM PE EIN=14.5 MEV. COS.DIST.

F. 11

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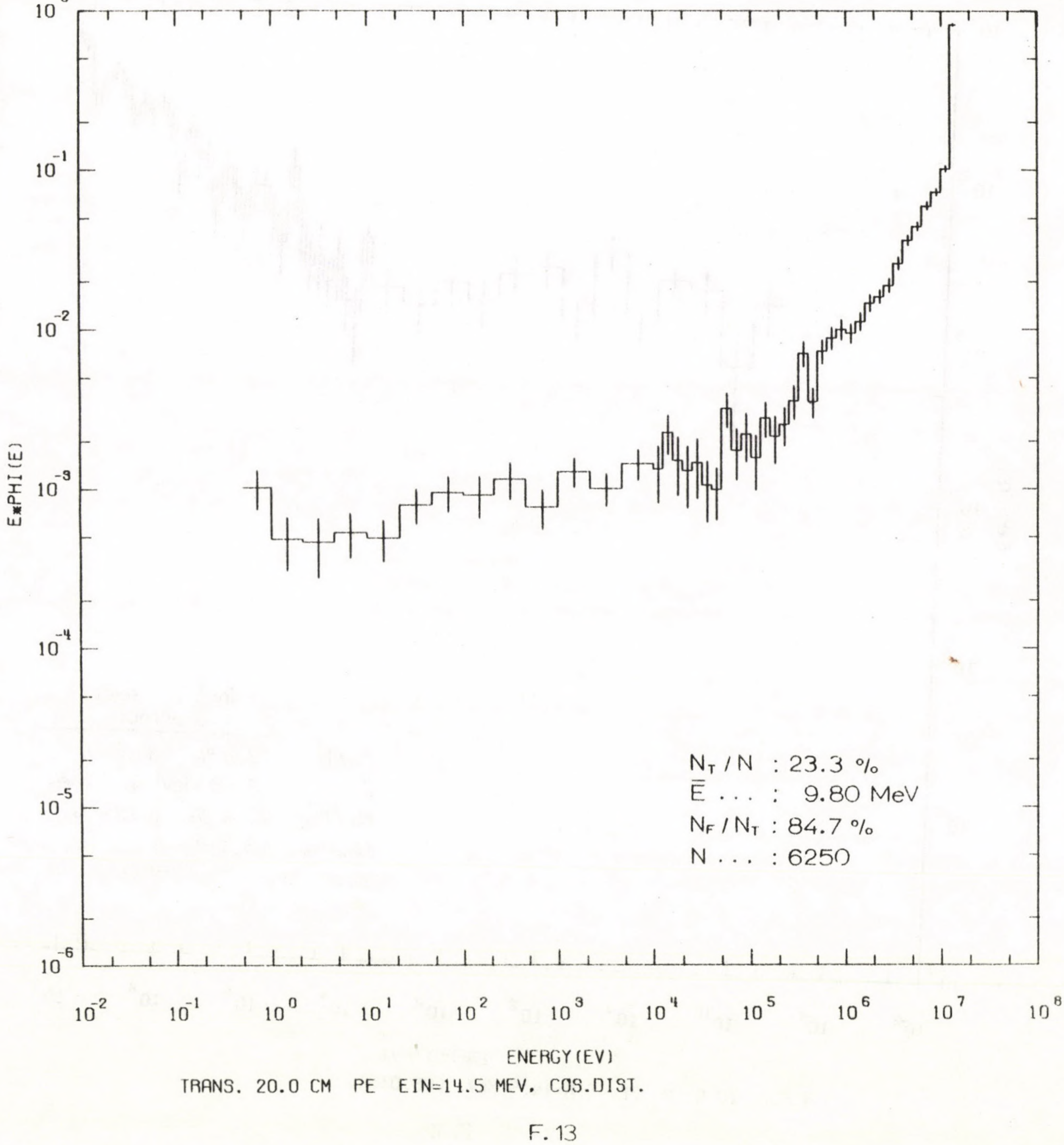
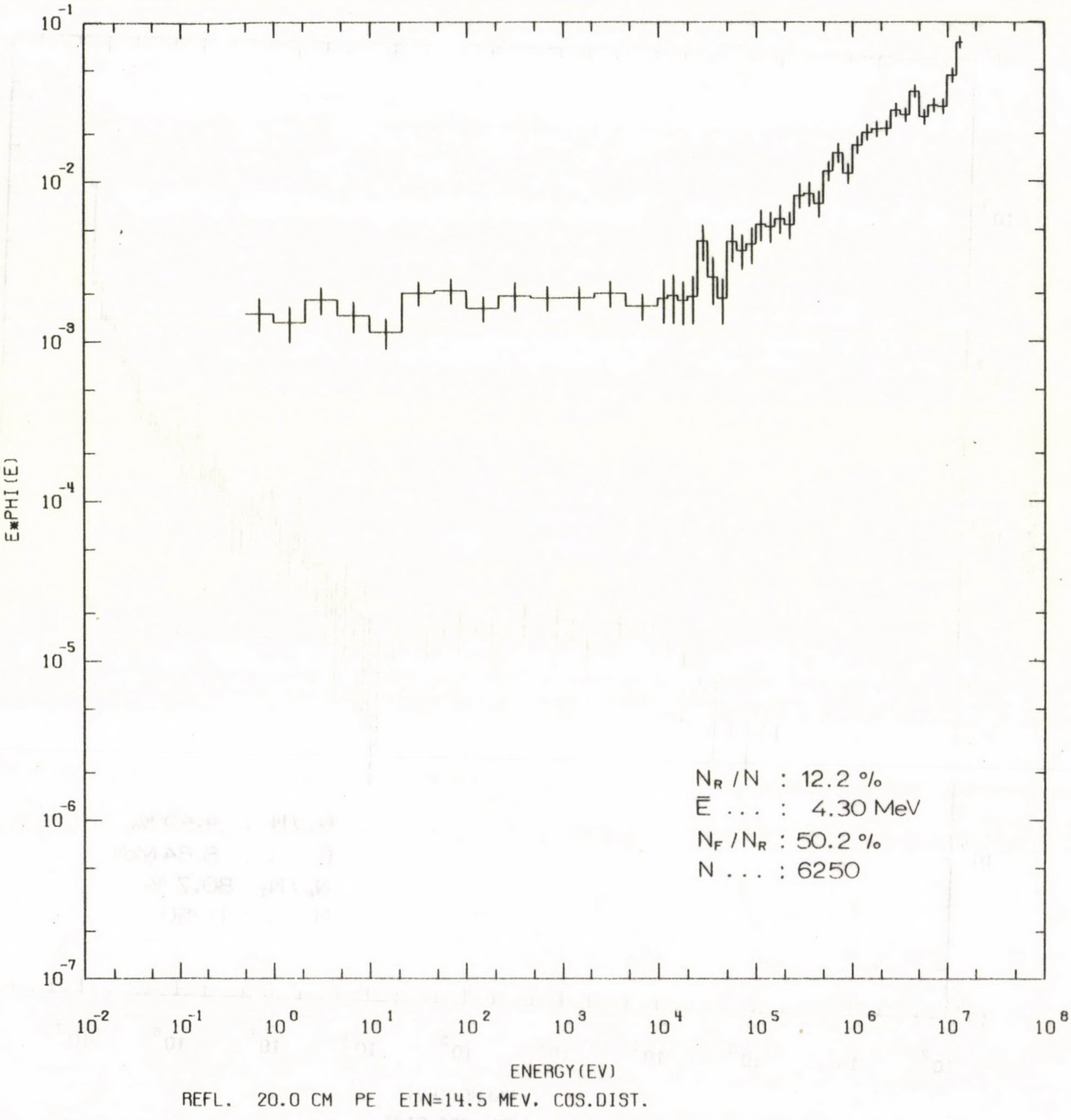
F. 12




TRANS. 20.0 CM PE EIN=14.5 MEV. COS.DIST.

F. 13

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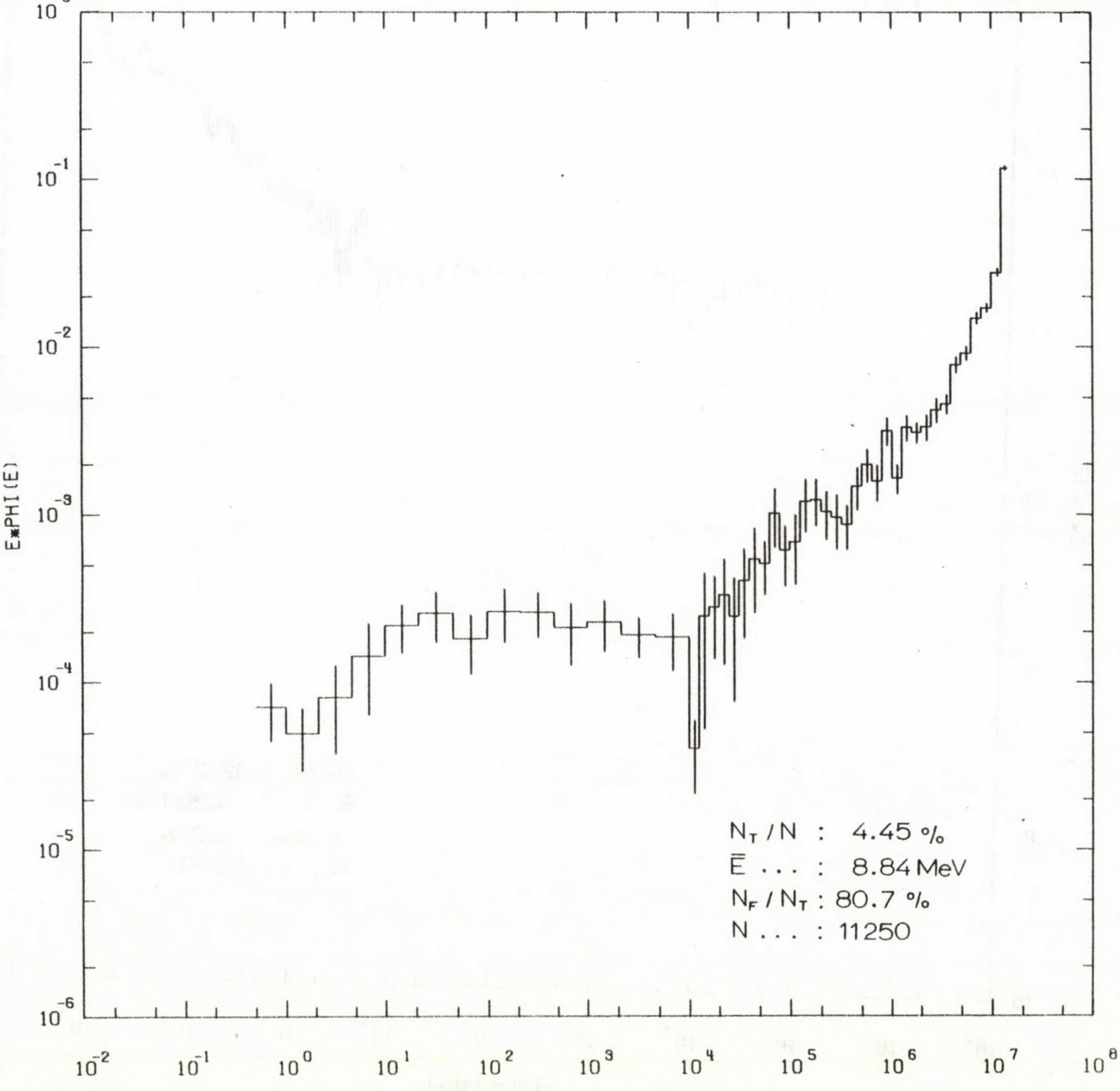
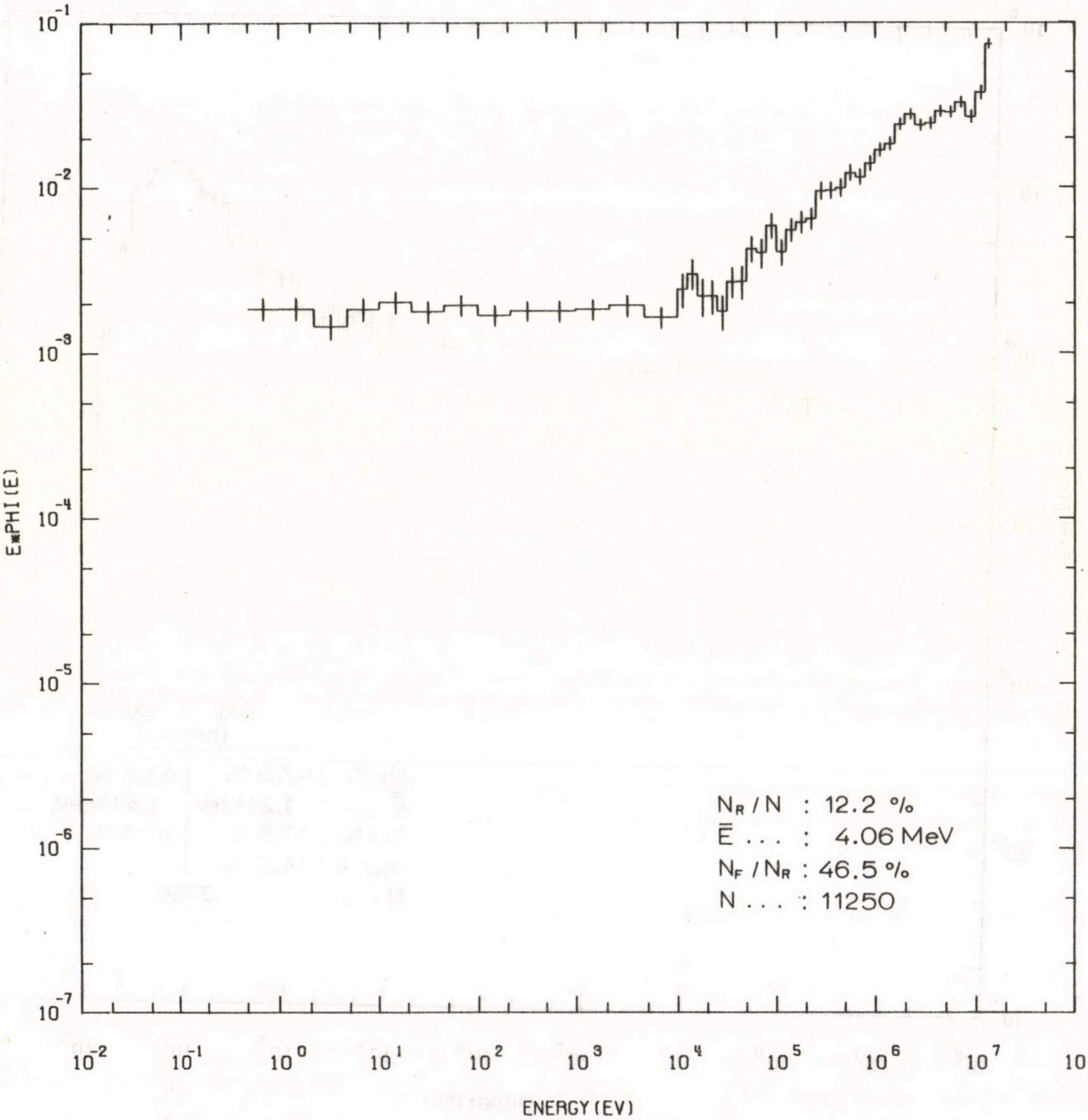
F. 14



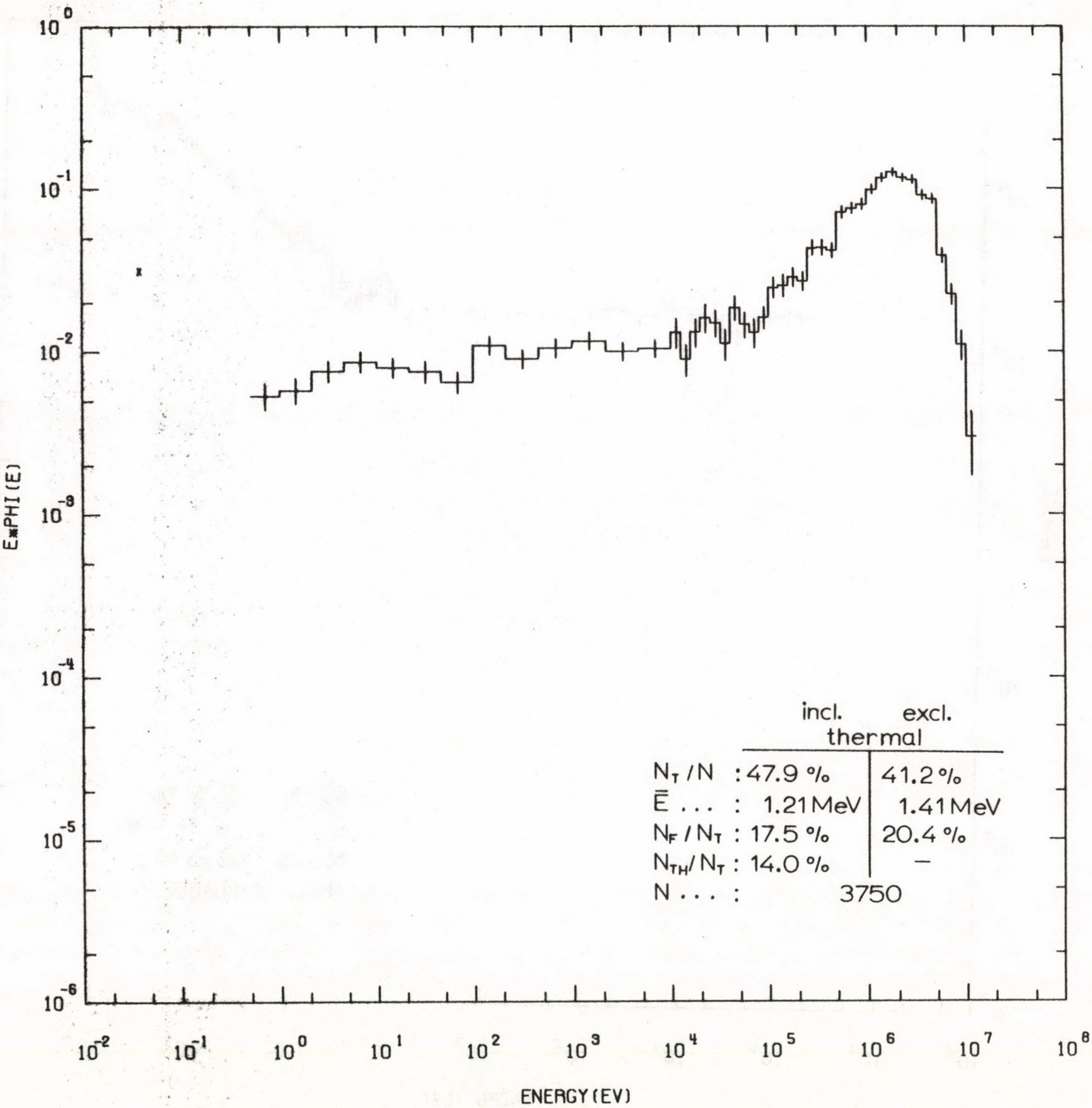

TRANS. 40.0 CM PE EIN=14.5 MEV, COS.DIST.

F. 15




REFL. 40.0 CM PE EIN=14.5 MEV, COS.DIST.

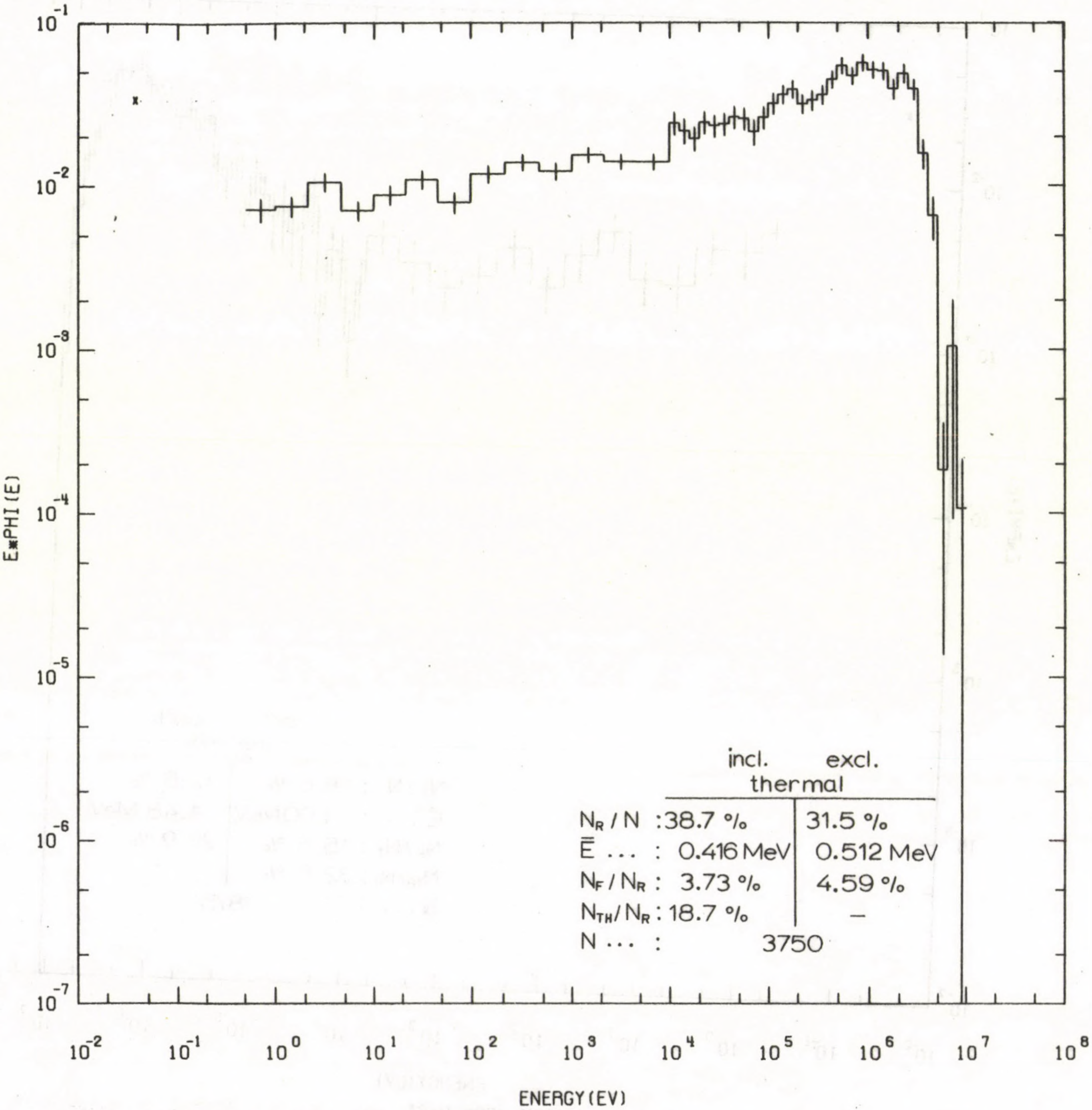
F. 16




| | incl. thermal | excl. thermal |
|---|---|---|
| $N_T/N$ : | 47.9 % | 41.2 % |
| $\bar{E}$ ... : | 1.21 MeV | 1.41 MeV |
| $N_F/N_T$ : | 17.5 % | 20.4 % |
| $N_{TH}/N_T$ : | 14.0 % | – |
| N ... : | 3750 | |

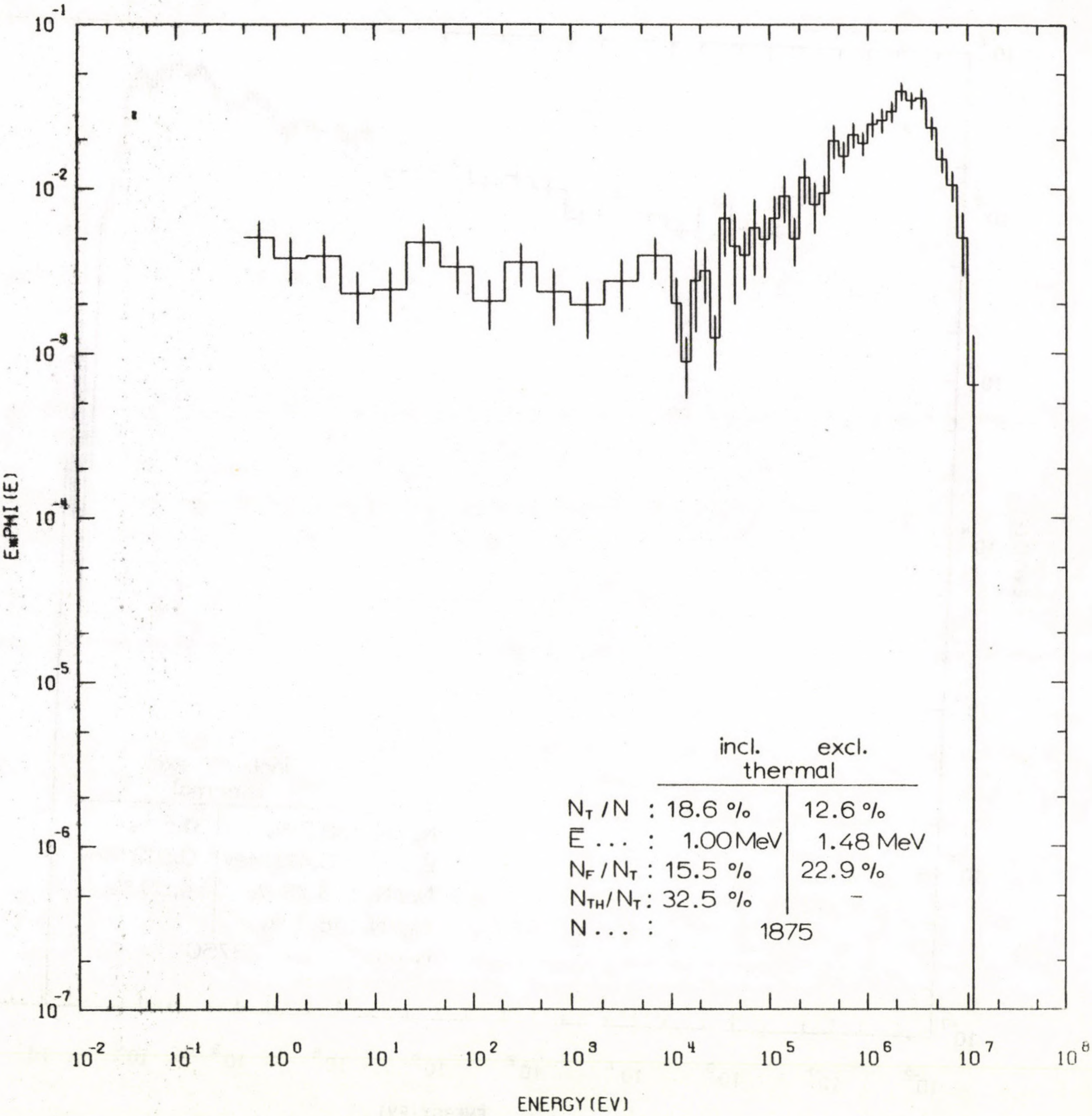
TRANS. 5.0 CM PE FISSION. COS.DIST

F. 17



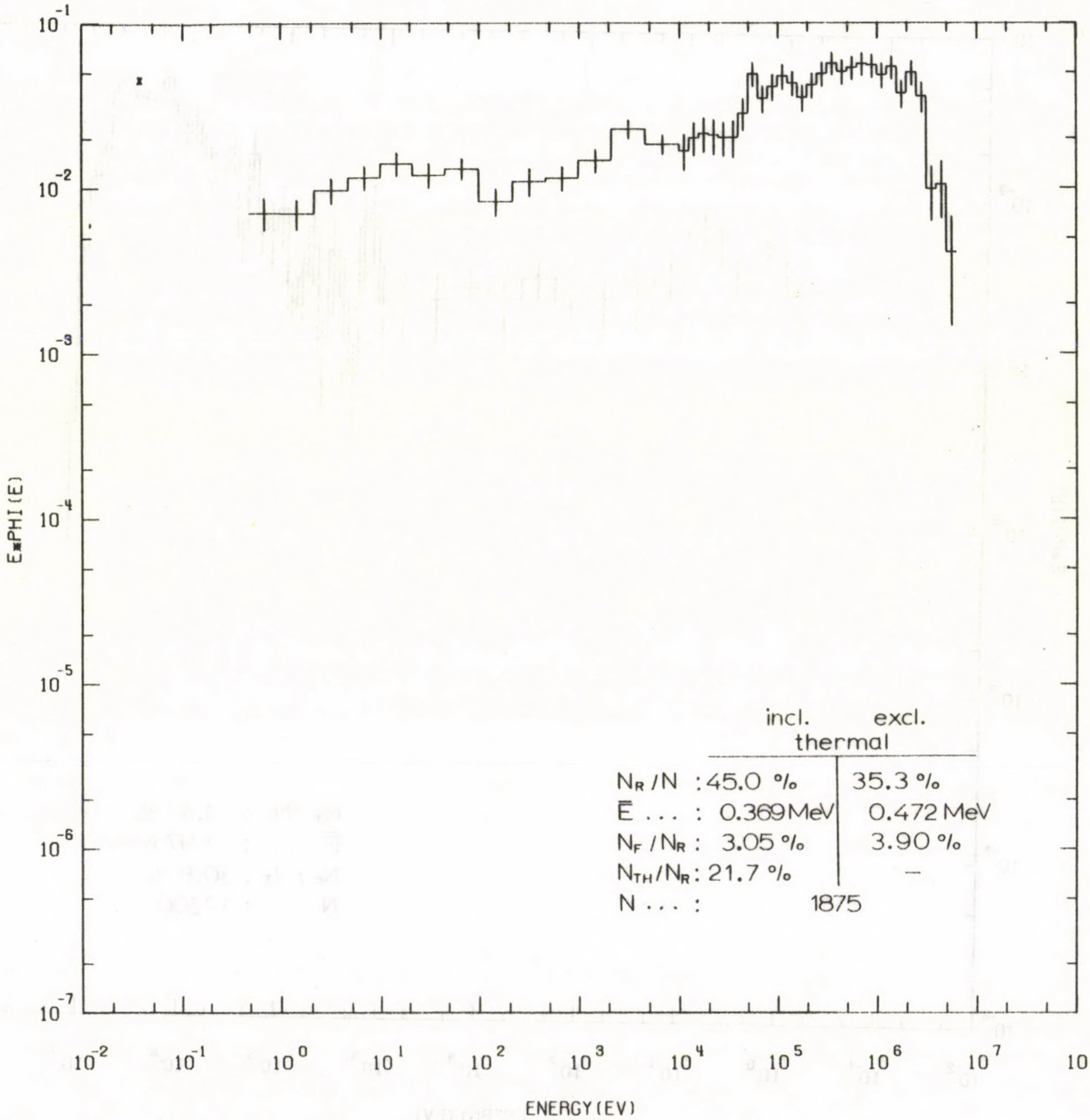

REFL. 5.0 CM PE FISSION. COS.DIST

F. 18



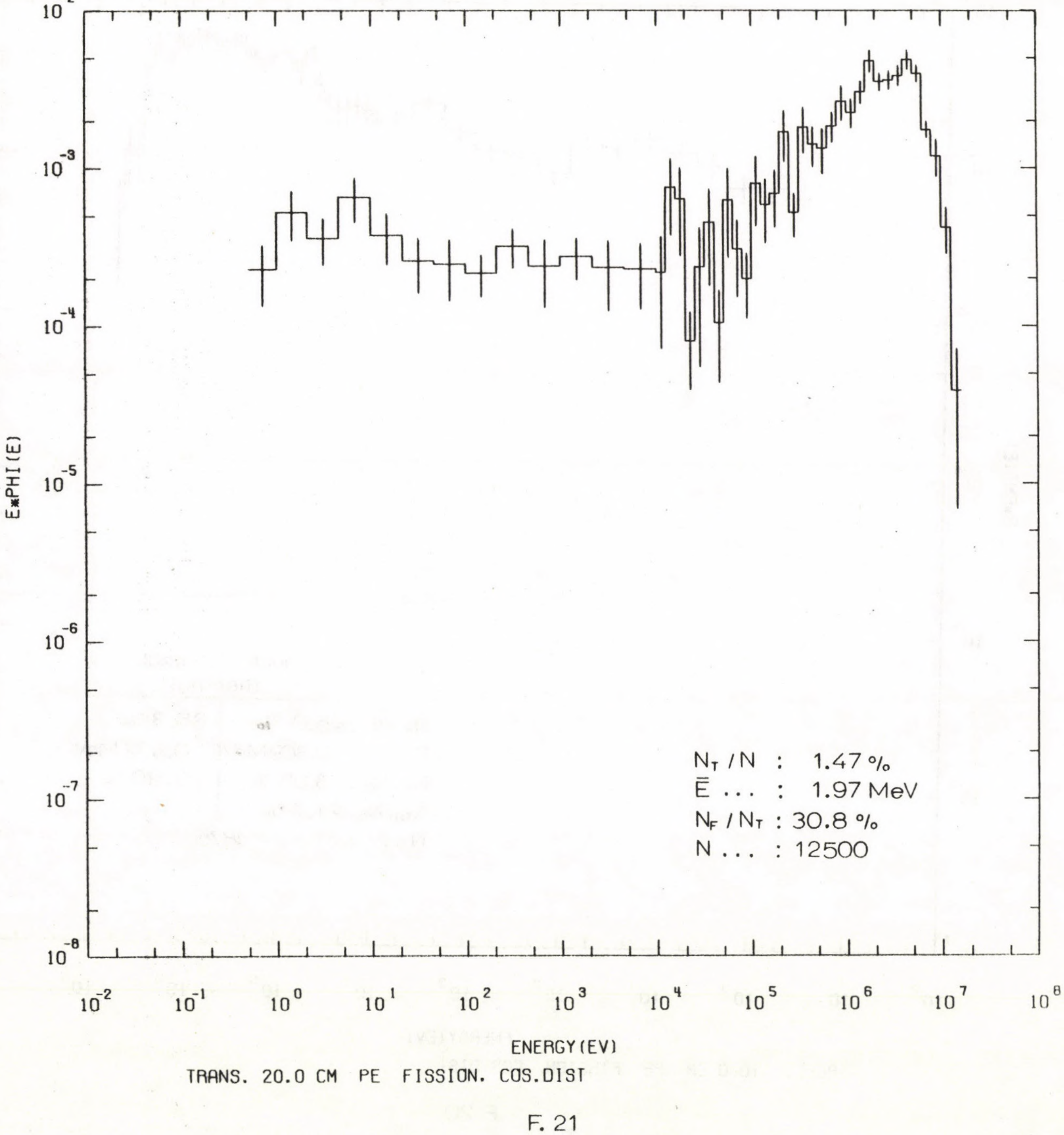

TRANS. 10.0 CM PE FISSION, COS.DIST

F. 19

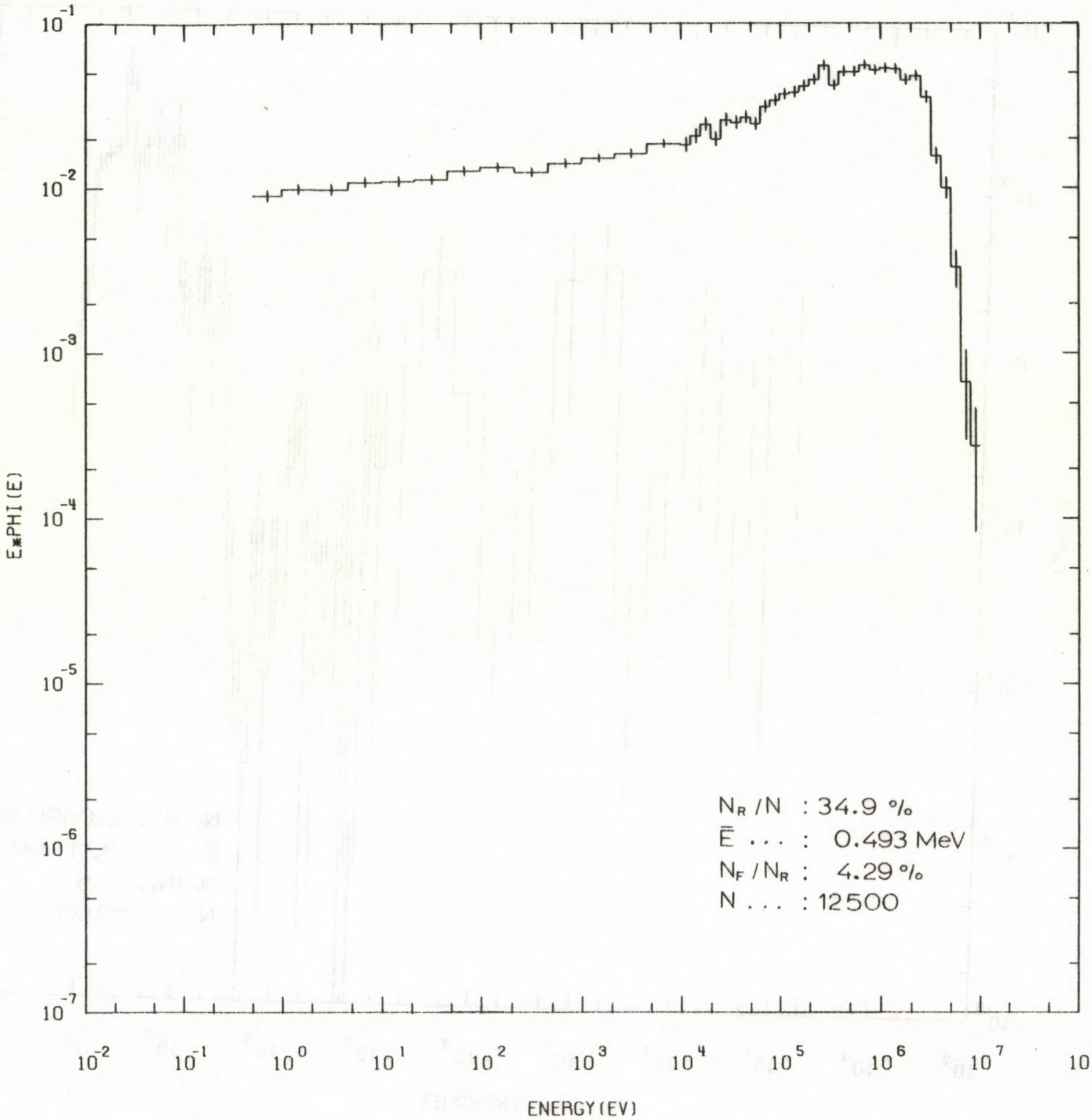



REFL. 10.0 CM PE FISSION. COS.DIST

F. 20

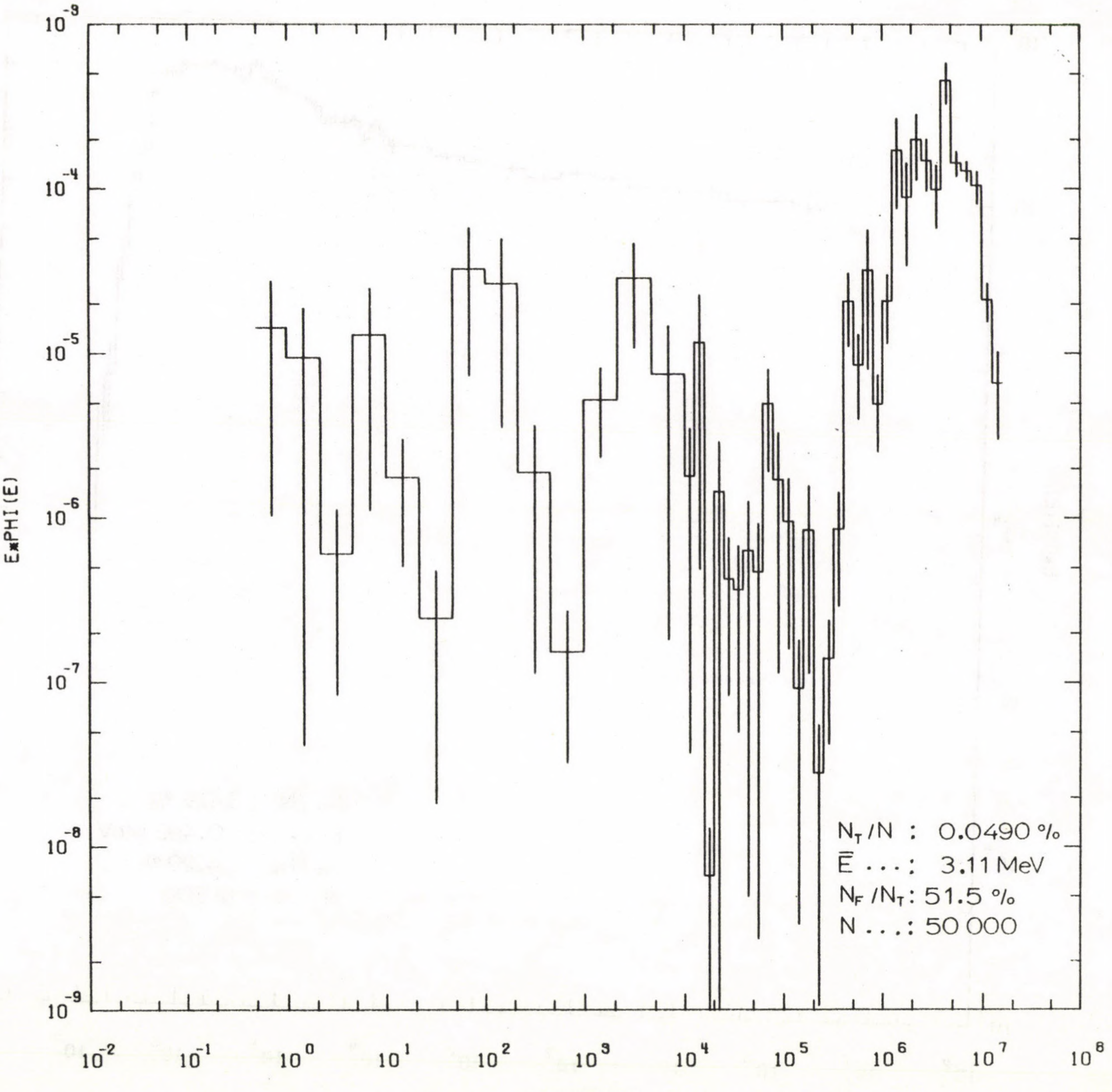

10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
10^-8
E*PHI(E)
10^-2 10^-1 10^0 10^1 10^2 10^3 10^4 10^5 10^6 10^7 10^8
ENERGY(EV)
N_T / N : 1.47 %
Ē ... : 1.97 MeV
N_F / N_T : 30.8 %
N ... : 12500
TRANS. 20.0 CM PE FISSION. COS.DIST

F. 21



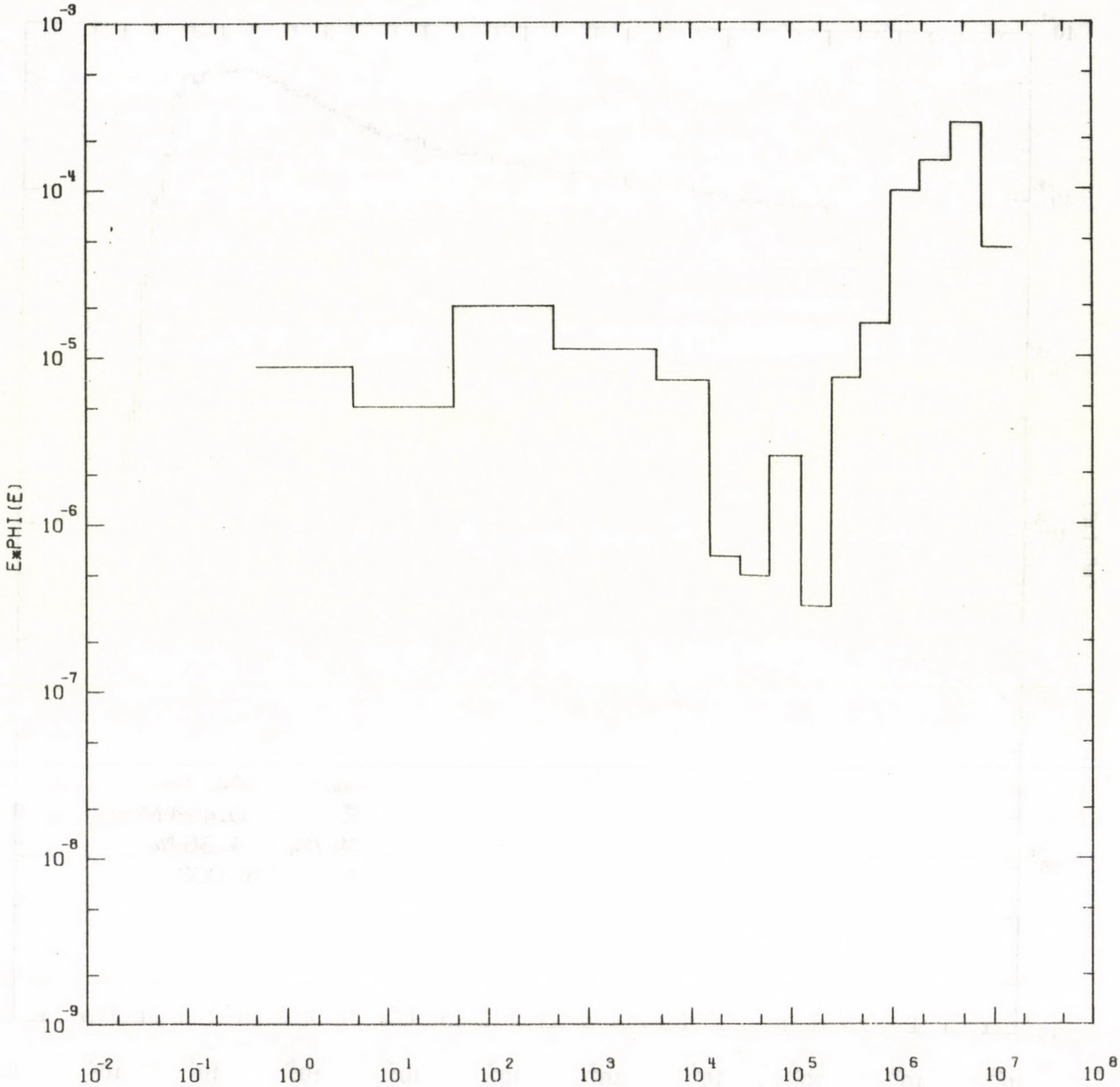

REFL. 20.0 CM PE FISSION. COS.DIST

F. 22



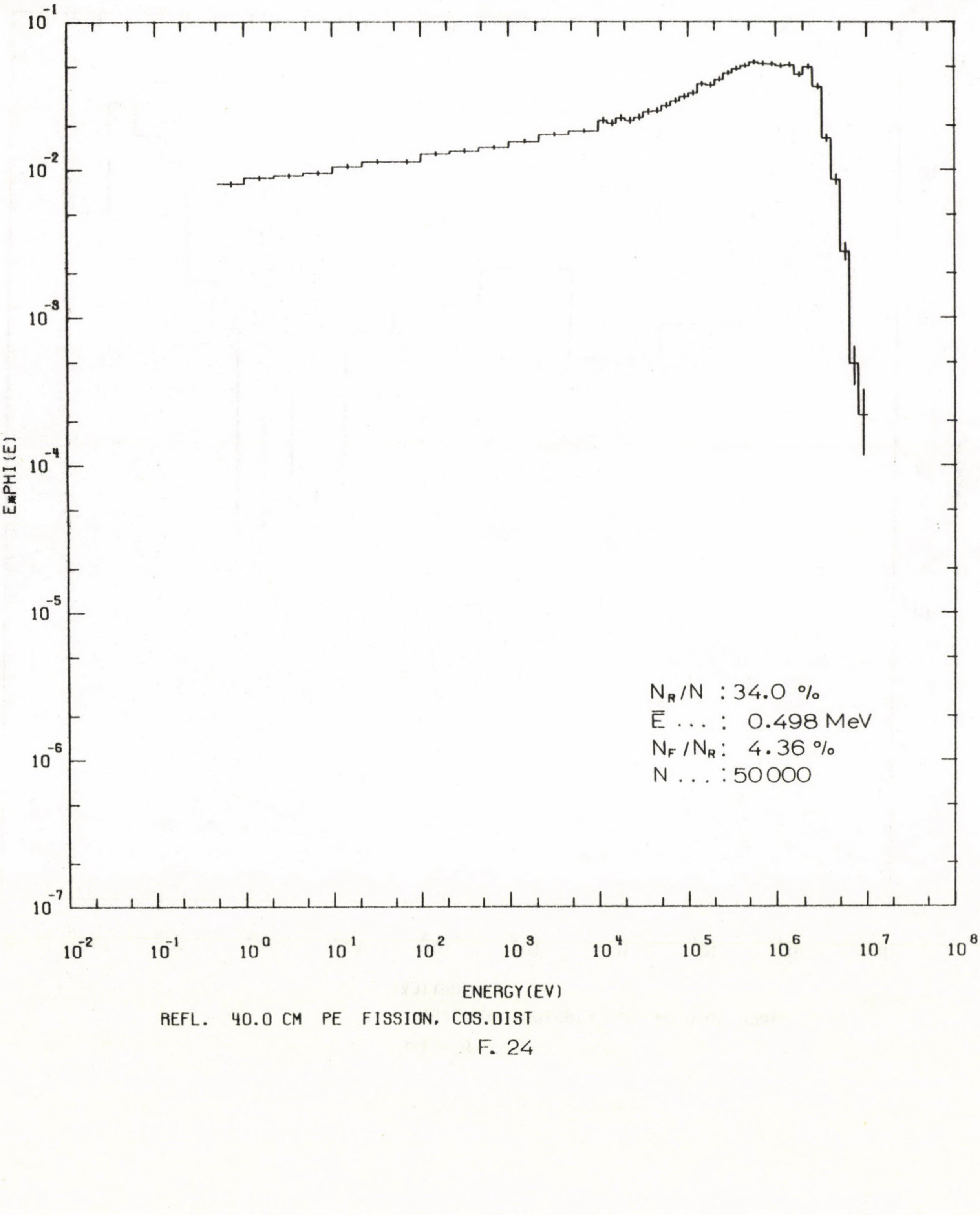

TRANS. 40.0 CM PE FISSION, COS.DIST

F. 23




TRANS. 40.0 CM PE FISSION. COS.DIST

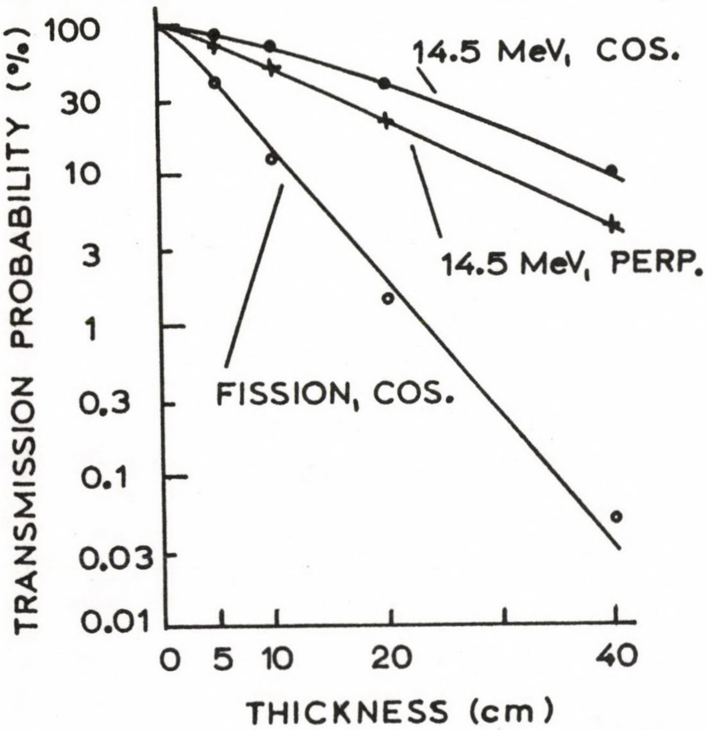
F. 23 a



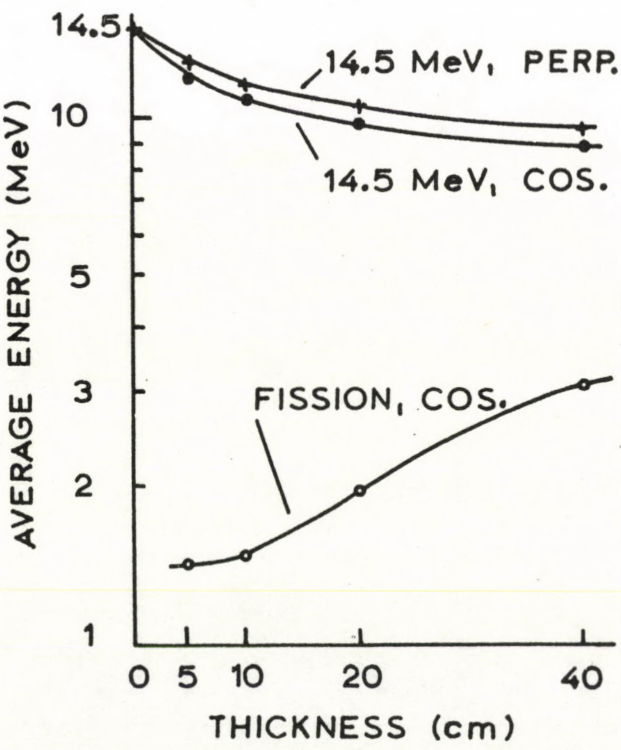

REFL. 40.0 CM PE FISSION, COS.DIST

F. 24

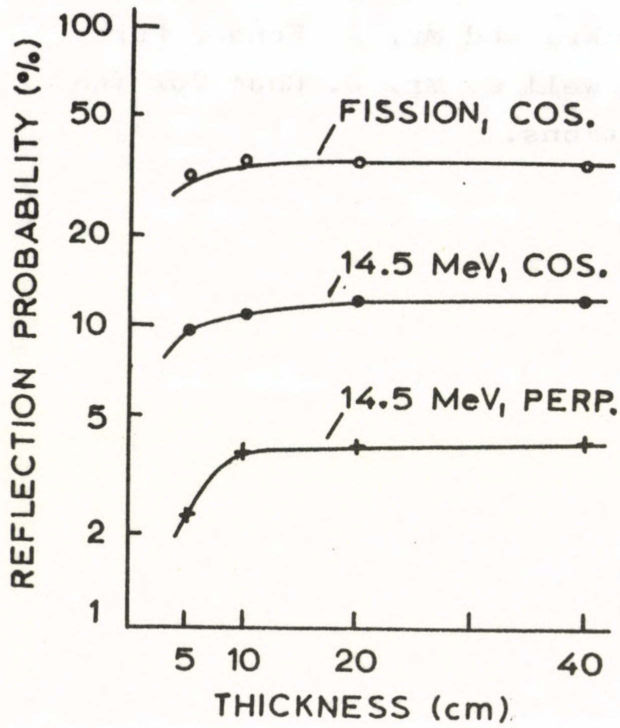
Figs 25-28
Characteristic quantities of the spectra



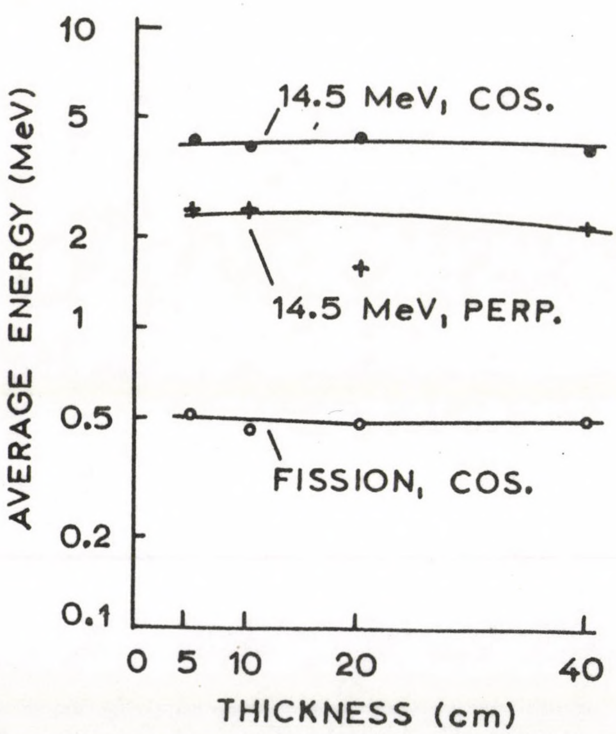

F. 25.
The probability of transmission without thermalisation vs slab thickness.




F. 26.
The average energy of the transmitted non-thermal neutrons.




F. 27.
The probability of reflection without thermalisation vs. slab thickness.




F. 28.
The average energy of the reflected non-thermal neutrons.

Acknowledgements

The authors thank Dr. S. Makra and Mr. A. Kondor for their valuable comments as well as Mr. J. Gadó for the use of the THERMOS instructions.

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Pálfalvi, J., Koblinger, L., 1974, KFKI-74-63.

62153

Kiadja a Központi Fizikai Kutató Intézet
Felelős kiadó: Szabó Ferenc tud. igazgató
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Készült a KFKI házi sokszorositó üzemében
Budapest, 1975. március hó