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R.L. WALSH*

- September 1987 -

* Permanent address : Australian Atomic Energy Commission Research Establishment
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I INTRODUCTION

For fission reactors in general and, in particular, for the proposed fusion-fission hybrid reactors [Ha82] where the D-T fusion reaction provides a source of 14 MeV neutrons to breed ^{233}U or ^{239}Pu fuel, it is important to know the relative number of high energy neutrons released in the fission reaction itself. Further, this number should be known as a function of the energy (0-14 MeV say) of the neutron which causes the fission. To calculate this high energy contribution it is first necessary to calculate the fission neutron energy spectrum. Experimental data above 10 MeV in the spectrum are generally not of high accuracy, because of low count rates.

Two recent formalisms for calculation of the fission neutron spectrum were considered for the present work. The formalism of Madland and Nix [Ma82] of Los Alamos National Laboratory calculates the fission neutron spectrum (FNS) for a range of fissioning nuclei and a range of excitation energies with just a single choice of the nuclear level density parameter and with no need to employ any other adjustable parameters. It gives good agreement with experimental data. A theoretical drawback is that it considers only average values for fragment properties (e.g. temperature maxima, excitation energy and kinetic energy) and uses a restricted range of fragments to represent the full mass and charge distributions of fission.

A second formalism considered was the 'Complex Cascade Evaporation Model' of Märten and Seeliger [Mär84a,Mär84b] of Technische Universität Dresden. This formalism incorporates the full range of fragment mass division. It also calculates the distribution of fragment excitation energy for each mass split - a calculation based on experimental values

of neutron emission, gamma-ray emission and fragment kinetic energy.

Of these two formalisms, the Dresden approach is more representative of the actual physical processes occurring. However, it is less straightforward than the Los Alamos approach and thus is not as readily applicable to a wide range of fission reactions. Therefore, for the present work, the Los Alamos formalism is used. An added feature is that the Los Alamos calculation is herewith extended to take account of fission fragment spin.

Once the numerical form of the spectrum is known, the relative contribution of high energy neutrons (15-20 MeV) can be determined by numerical integration (see section 3).

The Fortran IV program FISNEN (Fission Neutron Energies) was written to perform the calculations. FISNEN takes about six minutes of CPU time on the Cray computer at CEN Saclay, France, and about sixteen minutes CPU time on the IBM 4381-3 at AAEC, Lucas Heights, Australia. It occupies about 400K bytes of CPU memory.

II CALCULATION OF FISSION NEUTRON ENERGY SPECTRUM

II.1 The basic details of the Los Alamos formalism are now given. A fuller description may be found in [Ma82].

From the nuclear evaporation theory of Weisskopf [We37,B152] the centre-of-mass neutron energy spectrum $\phi(\varepsilon,\sigma_c)$ corresponding to a fixed residual nuclear temperature T is given approximately by

$$\phi(\varepsilon,\sigma_c) = k(T)\sigma_c(\varepsilon)\varepsilon \exp(-\varepsilon/T) \tag{1}$$

where k(T) is a temperature-dependent normalization constant given by

$$k(T) = \left[\int_0^\infty \sigma_c(\varepsilon)\varepsilon \exp(-\varepsilon/T)d\varepsilon\right]^{-1} \tag{2}$$

and where

ε = neutron energy in centre-of-mass system (CMS)

$\sigma_c(\varepsilon)$ = energy dependent cross section for process of compound nucleus formation. This process is the inverse of neutron emission. These two processes are related through the reciprocity theorem [Bl52].

The triangular distribution of temperature proposed by Terrell [Te59] is used

$$P(T) = \begin{cases} 2T/T_m^2, & \text{for } T \leq T_m \\ 0, & \text{for } T > T_m \end{cases} \tag{3}$$

where the maximum temperature T_m is related to E*, the initial average total fragment excitation energy, and to a, the nuclear level density parameter, approximately by

$$T_m = \left(\frac{\langle E^* \rangle}{a}\right)^{\frac{1}{2}} \tag{4}$$

$\langle E^* \rangle$ is given by

$$\langle E^* \rangle = \langle E_r \rangle + B_n + E_n - \langle E_f^{tot} \rangle \tag{5}$$

$\langle E_r \rangle$ = average energy release in fission, calculated for seven mass and charge divisions centred about the average of the A and Z distributions

B_n = neutron separation energy

E_n = kinetic energy of neutron inducing the fission

$\langle E_f^{tot} \rangle$ = average total fragment kinetic energy; from data of [Un73].

The nuclear level density parameter a is taken as

$$a = A/(11 \text{ MeV}) \tag{6}$$

where A is the mass number of the fissioning compound nucleus. This expression for a is found to give the best fit to experimental data for the fission neutron spectrum.

If the spectrum of equation (1) is integrated over the triangular temperature distribution of equation (3), the result for the CMS neutron energy spectrum is

$$\Phi(\varepsilon,\sigma_c) = \frac{2\sigma_c(\varepsilon)\varepsilon}{T_m^2} \int_0^{T_m} k(T)T \exp(-\varepsilon/T)dT \tag{7}$$

To transform from the CMS to the laboratory system one uses the general result [Fe42,Te59]

$$N(E,E_f,\sigma_c) = \frac{1}{4\sqrt{E_f}} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \frac{\Phi(\varepsilon,\sigma_c)}{\sqrt{\varepsilon}} d\varepsilon \tag{8}$$

where E = energy of neutron in laboratory system.

Inherent in equation (8) is the assumption that the neutrons are emitted isotropically from a fission fragment moving with average kinetic energy per nucleon E_f.

Thus, insertion of equation (8) into equation (7) gives for the laboratory fission neutron energy spectrum $N(E,E_f,\sigma_c)$

$$N(E,E_f,\sigma_c) = \frac{1}{2\sqrt{E_f}\ T_m^2} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \sigma_c(\varepsilon)\ \sqrt{\varepsilon}\ d\varepsilon \quad \times$$

$$\times \int_0^{T_m} k(T)T \exp(-\varepsilon/T)dT \tag{9}$$

Thus equation (9) comprises the essential form of the Los Alamos calculation [Ma82].

Values for the compound nucleus formation cross section $\sigma_c(\varepsilon)$ are calculated for the central fragments of the light and heavy mass peaks using the global optical model potential of Becchetti-Greenlees [Be69]. For the present work, values of $\sigma_c(\varepsilon)$ for central fragments in the reactions $^{235}U(n,f)$ and $^{239}Pu(n,f)$ were taken from

[Ma85]. For the reactions ^{232}Th(n,f), ^{233}U(n,f), ^{238}U(n,f) and ^{252}Cf(sf) the values of $\sigma_c(\varepsilon)$ were calculated with the computer code SCAT2 [Be81]. The range of values of ε used in the above is 1 keV $\leq \varepsilon \leq$ 40 MeV. For ε < 1 keV, $\sigma_c(\varepsilon)$ is taken to be

$$\sigma_c(\varepsilon) = \alpha + \beta/\sqrt{\varepsilon} \tag{10}$$

with the constants α, β determined from the $\sigma_c(\varepsilon)$ value and slope at 1 keV. For ε > 40 MeV the value of $\sigma_c(\varepsilon)$ at 40 MeV is used. One hundred values of $\sigma_c(\varepsilon)$ were calculated for particular energies. Other $\sigma_c(\varepsilon)$ values at energies intermediate to these were derived using cubic-spline interpolation.

For ^{232}Th(n,f) the average light and heavy fragments were taken to be ^{92}Kr and ^{141}Xe respectively, and $\langle E_f^{tot} \rangle$ was taken to be 163.5 MeV. These values were taken from the data of [Tr79] for 2 MeV neutron fission of ^{232}Th.

In the present work the fission neutron spectrum was calculated for values of E between 10 keV and 20 MeV, in 10 keV steps.

II.2 Incorporation of fragment spin

It has been found by Wilhelmy et al. [Wi72] that ^{252}Cf(sf) fission fragments possess large angular momentum ($J \sim 7 \pm 2\hbar$) aligned perpendicular to the fission axis. On this basis, it has been shown by Gavron [Ga76] that the resulting neutron emission is not isotropic in the CMS but contains about 10% anisotropy. Thus the transformation given by equation (8) is not stricly valid.

When a neutron is emitted from a moving fragment at an angle θ' in the CMS, the laboratory neutron energy E is given by

$$E = E_f + \varepsilon + 2(E_f\varepsilon)^{\frac{1}{2}}\cos\theta' \tag{11}$$

For anisotropic neutron emission which is nevertheless symmetrical about 90°, the CMS neutron energy spectrum may be taken as [Te59]

$$\Phi(\varepsilon,\sigma_c,\theta') = \frac{\Phi(\varepsilon,\sigma_c)(1+b\cos^2\theta')}{1 + b/3} \tag{12}$$

where $b = W(\theta')/W(90°)-1$ is the anisotropy

and $W(\theta')$ is the neutron angular distribution in the CMS.

Combination of equations (11) and (12) gives for the correct transformation to the laboratory system when anisotropy is present

$$N(E,E_f,\sigma_c) = \frac{1}{4\sqrt{E_f}} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \frac{\Phi(\varepsilon,\sigma_c)\left[1+b(E-\varepsilon-E_f)^2/4\varepsilon E_f\right]d\varepsilon}{\sqrt{\varepsilon}\ (1+b/3)} \tag{13}$$

rather than that of equation (8). Therefore, the modified form for the laboratory fission neutron energy spectrum becomes

$$\begin{aligned} N(E,E_f,\sigma_c) = {} & \frac{1}{2\sqrt{E_f}\ T_m^2(1+b/3)} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \sigma_c(\varepsilon)\ \sqrt{\varepsilon}\ d\varepsilon \ \ x \\ & x \int_0^{T_m} k(T)T\ \exp(-\varepsilon/T)dT \ \ + \\ & + \frac{b}{8E_f^{3/2}T_m^2(1+b/3)} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \frac{\sigma_c(\varepsilon)(E-\varepsilon-E_f)^2}{\sqrt{\varepsilon}}\ d\varepsilon\ x \\ & x \int_0^{T_m} k(T)T\ \exp(-\varepsilon/T)dT \end{aligned} \tag{14}$$

The calculation represented by equation (14) gives the results of the present work. Numerical integration was used: Gauss-Legendre for the integrations over ε and T, Gauss-Laguerre to calculate k(T) (see equation (2)). Thirty-two order quadrature was used for the Gauss-Legendre integrations and sixteen order quadrature for the Gauss-Laguerre integration.

The value of the anisotropy constant b was taken as 0.1, from [Ga76].

III RESULTS

The results of the present calculation for neutron fission of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu and for spontaneous fission of ^{252}Cf are shown in Figures 1 to 6 respectively. Tables 1 to 6 list the numerical data for these spectra (b = 0.1). The area under each fission neutron spectrum curve is normalised to 1.

Table 7 gives the ratio R of fission spectrum integrals for the above fission reactions, for incident neutron energies 0, 2 and 14 MeV,

$$R = \frac{\int_{15\ MeV}^{20\ MeV} N(E)\ dE}{\int_{0}^{20\ MeV} N(E)\ dE} \qquad (15)$$

The R values are obtained by numerical integration of the data of Tables 1 to 6. As expected, R increases with E_n. That is, the spectra become 'harder'

For 14 MeV incident neutron energy, a correction for multiple - chance fission is made, based on the data of [Ma 82] for 14 MeV neutron fission of ^{235}U. The R value calculated from first-chance fission in the present work is reduced by 50% to give the multiple chance value. This correction was assumed to be the same for each reaction listed in Table 1.

In Figures 1 to 6 the dashed line shows the fission neutron spectrum after incorporation of fragment spin (b = 0.1) ; the continuous line shows the spectrum assuming no fragment spin. The effect of fragment spin is only small - it hardens the spectrum below 0.5 MeV and above 5 MeV. The amount of this hardening is about 2% at 0.1 MeV, about 1% at 10 MeV and 2 to 3% at 20 MeV. Inclusion of fragment spin increases the average energy of the spectrum by 2 to 3 keV.

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TABLE 1

Numerical Values of Fission Neutron Spectrum for $^{232}Th(n,f)$ at E_n = 2 MeV
(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	6.231D-02
0.05	1.376D-01
0.10	1.910D-01
0.15	2.293D-01
0.20	2.592D-01
0.40	3.299D-01
0.60	3.531D-01
0.80	3.520D-01
1.00	3.405D-01
1.20	3.238D-01
1.40	3.047D-01
1.60	2.841D-01
1.80	2.625D-01
2.00	2.400D-01
2.20	2.177D-01
2.40	1.957D-01
2.60	1.747D-01
2.80	1.552D-01
3.00	1.373D-01
3.20	1.210D-01
3.40	1.064D-01
3.60	9.328D-02
3.80	8.176D-02
4.00	7.157D-02
4.20	6.261D-02
4.40	5.473D-02
4.60	4.781D-02
4.80	4.174D-02
5.00	3.641D-02
5.20	3.174D-02
5.40	2.763D-02
5.60	2.403D-02
5.80	2.088D-02
6.00	1.812D-02
6.20	1.571D-02
6.40	1.360D-02
6.60	1.176D-02
6.80	1.016D-02
7.00	8.759D-03
7.20	7.555D-03
7.40	6.502D-03
7.60	5.594D-03
7.80	4.808D-03
8.00	4.130D-03
8.20	3.545D-03
8.40	3.046D-03
8.60	2.612D-03
8.80	2.236D-03
9.00	1.918D-03
9.20	1.645D-03
9.40	1.408D-03
9.60	1.204D-03
9.80	1.033D-03
10.00	8.836D-04
10.20	7.559D-04
10.40	6.470D-04
10.60	5.535D-04
10.80	4.735D-04
11.00	4.049D-04
11.20	3.462D-04
11.40	2.959D-04
11.60	2.531D-04
11.80	2.161D-04
12.00	1.846D-04
12.20	1.578D-04
12.40	1.347D-04
12.60	1.150D-04
12.80	9.822D-05
13.00	8.387D-05
13.20	7.147D-05
13.40	6.099D-05
13.60	5.210D-05
13.80	4.437D-05
14.00	3.783D-05
14.20	3.228D-05
14.40	2.749D-05
14.60	2.342D-05
14.80	1.996D-05
15.00	1.700D-05
15.20	1.446D-05
15.40	1.232D-05
15.60	1.048D-05
15.80	8.917D-06
16.00	7.590D-06
16.20	6.455D-06
16.40	5.485D-06
16.60	4.664D-06
16.80	3.964D-06
17.00	3.370D-06
17.20	2.862D-06
17.40	2.433D-06
17.60	2.066D-06
17.80	1.754D-06
18.00	1.488D-06
18.20	1.262D-06
18.40	1.071D-06
18.60	9.106D-07
18.80	7.731D-07
19.00	6.562D-07
19.20	5.563D-07
19.40	4.707D-07
19.60	3.987D-07
19.80	3.381D-07
20.00	2.868D-07

MEAN ENERGY = 1.98311

TABLE 2

Numerical Values of Fission Neutron Spectrum for $^{233}U(n,f)$ at E_n = 0 MeV

(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	5.910D-02
0.05	1.304D-01
0.10	1.810D-01
0.15	2.173D-01
0.20	2.455D-01
0.40	3.127D-01
0.60	3.361D-01
0.80	3.367D-01
1.00	3.274D-01
1.20	3.130D-01
1.40	2.963D-01
1.60	2.780D-01
1.80	2.584D-01
2.00	2.378D-01
2.20	2.171D-01
2.40	1.965D-01
2.60	1.768D-01
2.80	1.583D-01
3.00	1.412D-01
3.20	1.255D-01
3.40	1.113D-01
3.60	9.854D-02
3.80	8.717D-02
4.00	7.703D-02
4.20	6.802D-02
4.40	6.002D-02
4.60	5.292D-02
4.80	4.664D-02
5.00	4.106D-02
5.20	3.613D-02
5.40	3.174D-02
5.60	2.787D-02
5.80	2.444D-02
6.00	2.141D-02
6.20	1.874D-02
6.40	1.637D-02
6.60	1.430D-02
6.80	1.247D-02
7.00	1.086D-02
7.20	9.464D-03
7.40	8.230D-03
7.60	7.155D-03
7.80	6.216D-03
8.00	5.397D-03
8.20	4.682D-03
8.40	4.067D-03
8.60	3.527D-03
8.80	3.054D-03
9.00	2.648D-03
9.20	2.297D-03
9.40	1.988D-03
9.60	1.721D-03
9.80	1.492D-03
10.00	1.291D-03
10.20	1.118D-03
10.40	9.678D-04
10.60	8.373D-04
10.80	7.244D-04
11.00	6.268D-04
11.20	5.420D-04
11.40	4.686D-04
11.60	4.054D-04
11.80	3.501D-04
12.00	3.024D-04
12.20	2.615D-04
12.40	2.258D-04
12.60	1.948D-04
12.80	1.683D-04
13.00	1.454D-04
13.20	1.253D-04
13.40	1.081D-04
13.60	9.337D-05
13.80	8.040D-05
14.00	6.931D-05
14.20	5.982D-05
14.40	5.151D-05
14.60	4.437D-05
14.80	3.826D-05
15.00	3.294D-05
15.20	2.834D-05
15.40	2.441D-05
15.60	2.102D-05
15.80	1.808D-05
16.00	1.557D-05
16.20	1.339D-05
16.40	1.151D-05
16.60	9.898D-06
16.80	8.509D-06
17.00	7.317D-06
17.20	6.286D-06
17.40	5.406D-06
17.60	4.642D-06
17.80	3.988D-06
18.00	3.422D-06
18.20	2.936D-06
18.40	2.521D-06
18.60	2.169D-06
18.80	1.863D-06
19.00	1.600D-06
19.20	1.373D-06
19.40	1.175D-06
19.60	1.007D-06
19.80	8.643D-07
20.00	7.419D-07

MEAN ENERGY = 2.08325

TABLE 3

Numerical Values of Fission Neutron Spectrum for $^{235}U(n,f)$ at E_n = 0 MeV

(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	5.951D-02
0.05	1.314D-01
0.10	1.826D-01
0.15	2.194D-01
0.20	2.481D-01
0.40	3.175D-01
0.60	3.430D-01
0.80	3.440D-01
1.00	3.342D-01
1.20	3.194D-01
1.40	3.018D-01
1.60	2.827D-01
1.80	2.621D-01
2.00	2.405D-01
2.20	2.189D-01
2.40	1.974D-01
2.60	1.769D-01
2.80	1.577D-01
3.00	1.401D-01
3.20	1.240D-01
3.40	1.094D-01
3.60	9.644D-02
3.80	8.489D-02
4.00	7.462D-02
4.20	6.554D-02
4.40	5.752D-02
4.60	5.045D-02
4.80	4.422D-02
5.00	3.870D-02
5.20	3.386D-02
5.40	2.958D-02
5.60	2.582D-02
5.80	2.252D-02
6.00	1.960D-02
6.20	1.705D-02
6.40	1.481D-02
6.60	1.285D-02
6.80	1.115D-02
7.00	9.647D-03
7.20	8.350D-03
7.40	7.218D-03
7.60	6.234D-03
7.80	5.379D-03
8.00	4.642D-03
8.20	4.002D-03
8.40	3.452D-03
8.60	2.975D-03
8.80	2.559D-03
9.00	2.204D-03
9.20	1.899D-03
9.40	1.633D-03
9.60	1.404D-03
9.80	1.209D-03
10.00	1.040D-03
10.20	8.933D-04
10.40	7.684D-04
10.60	6.603D-04
10.80	5.674D-04
11.00	4.874D-04
11.20	4.186D-04
11.40	3.594D-04
11.60	3.087D-04
11.80	2.648D-04
12.00	2.272D-04
12.20	1.950D-04
12.40	1.671D-04
12.60	1.432D-04
12.80	1.229D-04
13.00	1.053D-04
13.20	9.014D-05
13.40	7.725D-05
13.60	6.621D-05
13.80	5.662D-05
14.00	4.846D-05
14.20	4.152D-05
14.40	3.548D-05
14.60	3.035D-05
14.80	2.598D-05
15.00	2.220D-05
15.20	1.897D-05
15.40	1.622D-05
15.60	1.386D-05
15.80	1.183D-05
16.00	1.011D-05
16.20	8.638D-06
16.40	7.369D-06
16.60	6.290D-06
16.80	5.367D-06
17.00	4.579D-06
17.20	3.908D-06
17.40	3.335D-06
17.60	2.843D-06
17.80	2.423D-06
18.00	2.063D-06
18.20	1.757D-06
18.40	1.498D-06
18.60	1.280D-06
18.80	1.091D-06
19.00	9.296D-07
19.20	7.906D-07
19.40	6.716D-07
19.60	5.716D-07
19.80	4.874D-07
20.00	4.151D-07

MEAN ENERGY = 2.03401

TABLE 4

Numerical Values of Fission Neutron Spectrum for $^{238}U(n,f)$ at E_n = 2 MeV

(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	5.984D-02
0.05	1.321D-01
0.10	1.835D-01
0.15	2.204D-01
0.20	2.489D-01
0.40	3.175D-01
0.60	3.420D-01
0.80	3.425D-01
1.00	3.326D-01
1.20	3.182D-01
1.40	3.010D-01
1.60	2.818D-01
1.80	2.613D-01
2.00	2.398D-01
2.20	2.181D-01
2.40	1.967D-01
2.60	1.764D-01
2.80	1.574D-01
3.00	1.399D-01
3.20	1.240D-01
3.40	1.096D-01
3.60	9.674D-02
3.80	8.532D-02
4.00	7.516D-02
4.20	6.614D-02
4.40	5.815D-02
4.60	5.109D-02
4.80	4.485D-02
5.00	3.933D-02
5.20	3.447D-02
5.40	3.017D-02
5.60	2.637D-02
5.80	2.303D-02
6.00	2.009D-02
6.20	1.750D-02
6.40	1.524D-02
6.60	1.324D-02
6.80	1.150D-02
7.00	9.977D-03
7.20	8.649D-03
7.40	7.494D-03
7.60	6.484D-03
7.80	5.609D-03
8.00	4.850D-03
8.20	4.194D-03
8.40	3.623D-03
8.60	3.129D-03
8.80	2.699D-03
9.00	2.332D-03
9.20	2.013D-03
9.40	1.735D-03
9.60	1.497D-03
9.80	1.292D-03
10.00	1.113D-03
10.20	9.591D-04
10.40	8.275D-04
10.60	7.129D-04
10.80	6.139D-04
11.00	5.290D-04
11.20	4.554D-04
11.40	3.922D-04
11.60	3.375D-04
11.80	2.904D-04
12.00	2.498D-04
12.20	2.150D-04
12.40	1.848D-04
12.60	1.588D-04
12.80	1.366D-04
13.00	1.173D-04
13.20	1.007D-04
13.40	8.662D-05
13.60	7.433D-05
13.80	6.378D-05
14.00	5.478D-05
14.20	4.701D-05
14.40	4.030D-05
14.60	3.458D-05
14.80	2.967D-05
15.00	2.541D-05
15.20	2.179D-05
15.40	1.868D-05
15.60	1.599D-05
15.80	1.370D-05
16.00	1.175D-05
16.20	1.006D-05
16.40	8.607D-06
16.60	7.366D-06
16.80	6.303D-06
17.00	5.391D-06
17.20	4.618D-06
17.40	3.951D-06
17.60	3.377D-06
17.80	2.884D-06
18.00	2.463D-06
18.20	2.106D-06
18.40	1.805D-06
18.60	1.544D-06
18.80	1.320D-06
19.00	1.126D-06
19.20	9.597D-07
19.40	8.188D-07
19.60	7.002D-07
19.80	5.989D-07
20.00	5.115D-07

MEAN ENERGY = 2.04501

TABLE 5

Numerical Values of Fission Neutron Spectrum for $^{239}Pu(n,f)$ at E_n = 0 MeV

(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	5.519D-02
0.05	1.217D-01
0.10	1.690D-01
0.15	2.030D-01
0.20	2.294D-01
0.40	2.939D-01
0.60	3.189D-01
0.80	3.222D-01
1.00	3.155D-01
1.20	3.040D-01
1.40	2.899D-01
1.60	2.737D-01
1.80	2.558D-01
2.00	2.368D-01
2.20	2.175D-01
2.40	1.981D-01
2.60	1.796D-01
2.80	1.619D-01
3.00	1.455D-01
3.20	1.304D-01
3.40	1.165D-01
3.60	1.040D-01
3.80	9.275D-02
4.00	8.258D-02
4.20	7.344D-02
4.40	6.527D-02
4.60	5.796D-02
4.80	5.141D-02
5.00	4.556D-02
5.20	4.035D-02
5.40	3.568D-02
5.60	3.153D-02
5.80	2.784D-02
6.00	2.454D-02
6.20	2.162D-02
6.40	1.903D-02
6.60	1.672D-02
6.80	1.469D-02
7.00	1.289D-02
7.20	1.130D-02
7.40	9.909D-03
7.60	8.674D-03
7.80	7.593D-03
8.00	6.644D-03
8.20	5.812D-03
8.40	5.082D-03
8.60	4.443D-03
8.80	3.879D-03
9.00	3.391D-03
9.20	2.961D-03
9.40	2.584D-03
9.60	2.256D-03
9.80	1.970D-03
10.00	1.718D-03
10.20	1.498D-03
10.40	1.308D-03
10.60	1.140D-03
10.80	9.934D-04
11.00	8.660D-04
11.20	7.542D-04
11.40	6.571D-04
11.60	5.723D-04
11.80	4.980D-04
12.00	4.335D-04
12.20	3.774D-04
12.40	3.281D-04
12.60	2.853D-04
12.80	2.484D-04
13.00	2.157D-04
13.20	1.873D-04
13.40	1.630D-04
13.60	1.415D-04
13.80	1.229D-04
14.00	1.067D-04
14.20	9.272D-05
14.40	8.038D-05
14.60	6.979D-05
14.80	6.061D-05
15.00	5.251D-05
15.20	4.556D-05
15.40	3.954D-05
15.60	3.424D-05
15.80	2.969D-05
16.00	2.575D-05
16.20	2.232D-05
16.40	1.933D-05
16.60	1.675D-05
16.80	1.450D-05
17.00	1.255D-05
17.20	1.088D-05
17.40	9.425D-06
17.60	8.156D-06
17.80	7.054D-06
18.00	6.095D-06
18.20	5.273D-06
18.40	4.572D-06
18.60	3.963D-06
18.80	3.430D-06
19.00	2.965D-06
19.20	2.558D-06
19.40	2.208D-06
19.60	1.910D-06
19.80	1.655D-06
20.00	1.432D-06

MEAN ENERGY = 2.18354

TABLE 6

Numerical Values of Fission Neutron Spectrum for ^{252}Cf Spontaneous Fission

(Anisotropy Constant b = 0.1)

ENERGY (MEV)	N(E)
0.01	5.309D-02
0.05	1.169D-01
0.10	1.620D-01
0.15	1.942D-01
0.20	2.195D-01
0.40	2.811D-01
0.60	3.057D-01
0.80	3.096D-01
1.00	3.044D-01
1.20	2.945D-01
1.40	2.821D-01
1.60	2.673D-01
1.80	2.509D-01
2.00	2.334D-01
2.20	2.152D-01
2.40	1.974D-01
2.60	1.799D-01
2.80	1.634D-01
3.00	1.479D-01
3.20	1.335D-01
3.40	1.202D-01
3.60	1.082D-01
3.80	9.715D-02
4.00	8.711D-02
4.20	7.806D-02
4.40	6.988D-02
4.60	6.250D-02
4.80	5.582D-02
5.00	4.983D-02
5.20	4.442D-02
5.40	3.957D-02
5.60	3.521D-02
5.80	3.129D-02
6.00	2.780D-02
6.20	2.465D-02
6.40	2.186D-02
6.60	1.936D-02
6.80	1.715D-02
7.00	1.516D-02
7.20	1.341D-02
7.40	1.185D-02
7.60	1.046D-02
7.80	9.237D-03
8.00	8.155D-03
8.20	7.195D-03
8.40	6.346D-03
8.60	5.593D-03
8.80	4.929D-03
9.00	4.346D-03
9.20	3.827D-03
9.40	3.368D-03
9.60	2.968D-03
9.80	2.611D-03
10.00	2.295D-03
10.20	2.020D-03
10.40	1.776D-03
10.60	1.560D-03
10.80	1.371D-03
11.00	1.205D-03
11.20	1.057D-03
11.40	9.284D-04
11.60	8.154D-04
11.80	7.147D-04
12.00	6.271D-04
12.20	5.501D-04
12.40	4.820D-04
12.60	4.226D-04
12.80	3.705D-04
13.00	3.243D-04
13.20	2.841D-04
13.40	2.491D-04
13.60	2.179D-04
13.80	1.908D-04
14.00	1.672D-04
14.20	1.461D-04
14.40	1.278D-04
14.60	1.120D-04
14.80	9.787D-05
15.00	8.559D-05
15.20	7.493D-05
15.40	6.548D-05
15.60	5.720D-05
15.80	5.003D-05
16.00	4.374D-05
16.20	3.821D-05
16.40	3.342D-05
16.60	2.921D-05
16.80	2.548D-05
17.00	2.227D-05
17.20	1.946D-05
17.40	1.699D-05
17.60	1.483D-05
17.80	1.294D-05
18.00	1.129D-05
18.20	9.868D-06
18.40	8.628D-06
18.60	7.534D-06
18.80	6.576D-06
19.00	5.730D-06
19.20	4.992D-06
19.40	4.354D-06
19.60	3.803D-06
19.80	3.321D-06
20.00	2.898D-06

MEAN ENERGY = 2.28151

TABLE 7

RATIO R OF FISSION SPECTRUM INTEGRALS FOR SEVERAL FISSION REACTIONS

$$R = \int_{15\ \mathrm{MeV}}^{20\ \mathrm{MeV}} N(E)dE \Big/ \int_{0}^{20\ \mathrm{MeV}} N(E)dE \quad \times 10^{-5}$$

	E_n = 0 MeV	E_n = 2 MeV	E_n = 14 MeV†
^{232}Th(n,f)	-	2.080	7.54
^{233}U(n,f)	4.311	6.106	13.50
^{235}U(n,f)	2.779	4.132	10.79
^{238}U(n,f)	-	3.234	9.13
^{239}Pu(n,f)	7.184	9.593	17.29
^{252}Cf(sf)	12.35	-	-

† Corrected for multiple-chance fission (see text).

FIGURE CAPTIONS

Figure 1 Fission Neutron Spectrum for $^{232}Th(n,f)$ at $E_n = 2$ MeV. Dashed line incorporates fragment spin ($b = 0.1$). Continuous line neglects fragment spin.

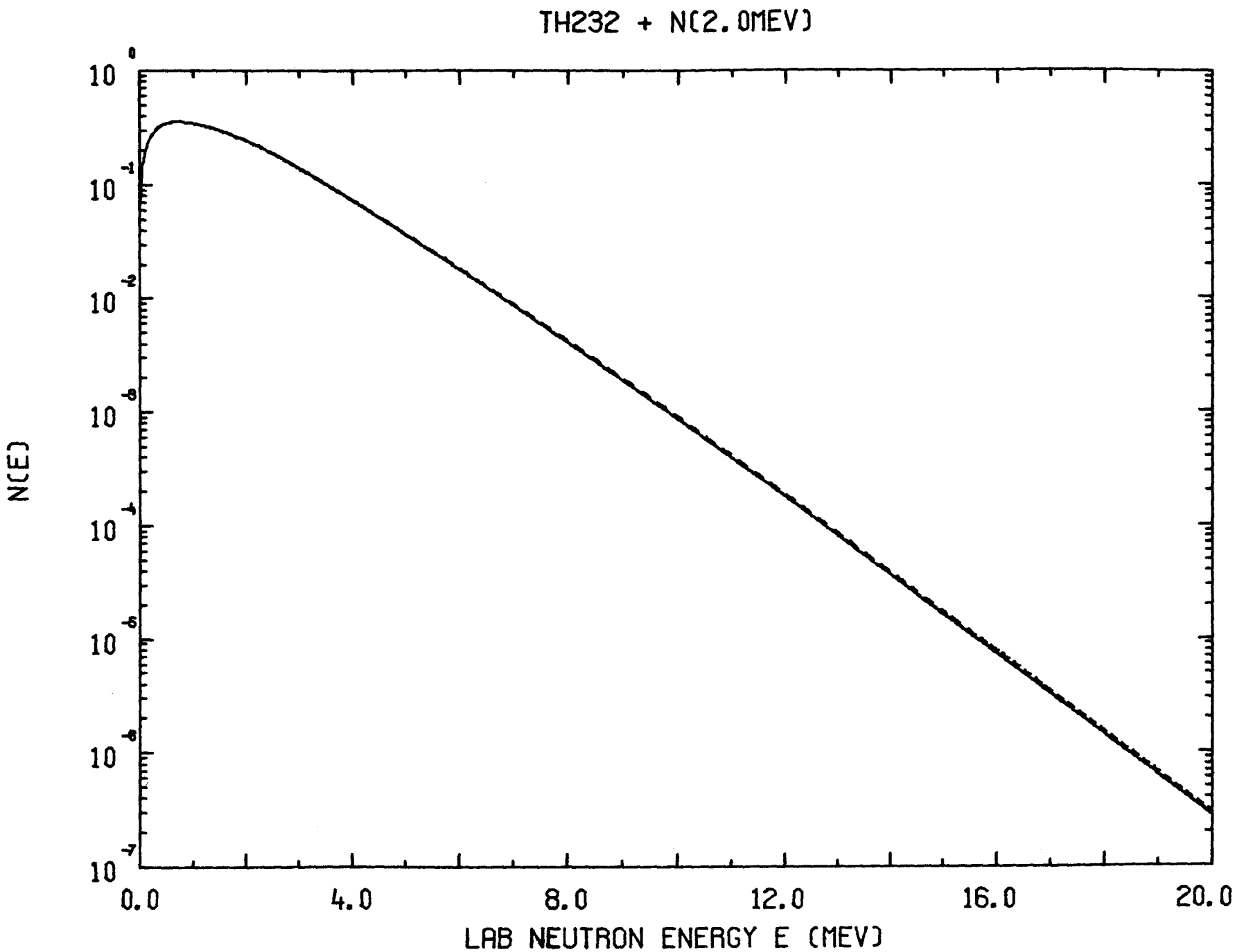
Figure 2 Fission Neutron Spectrum for $^{233}U(n,f)$ at $E_n = 0$ MeV. Dashed line as for Figure 1.

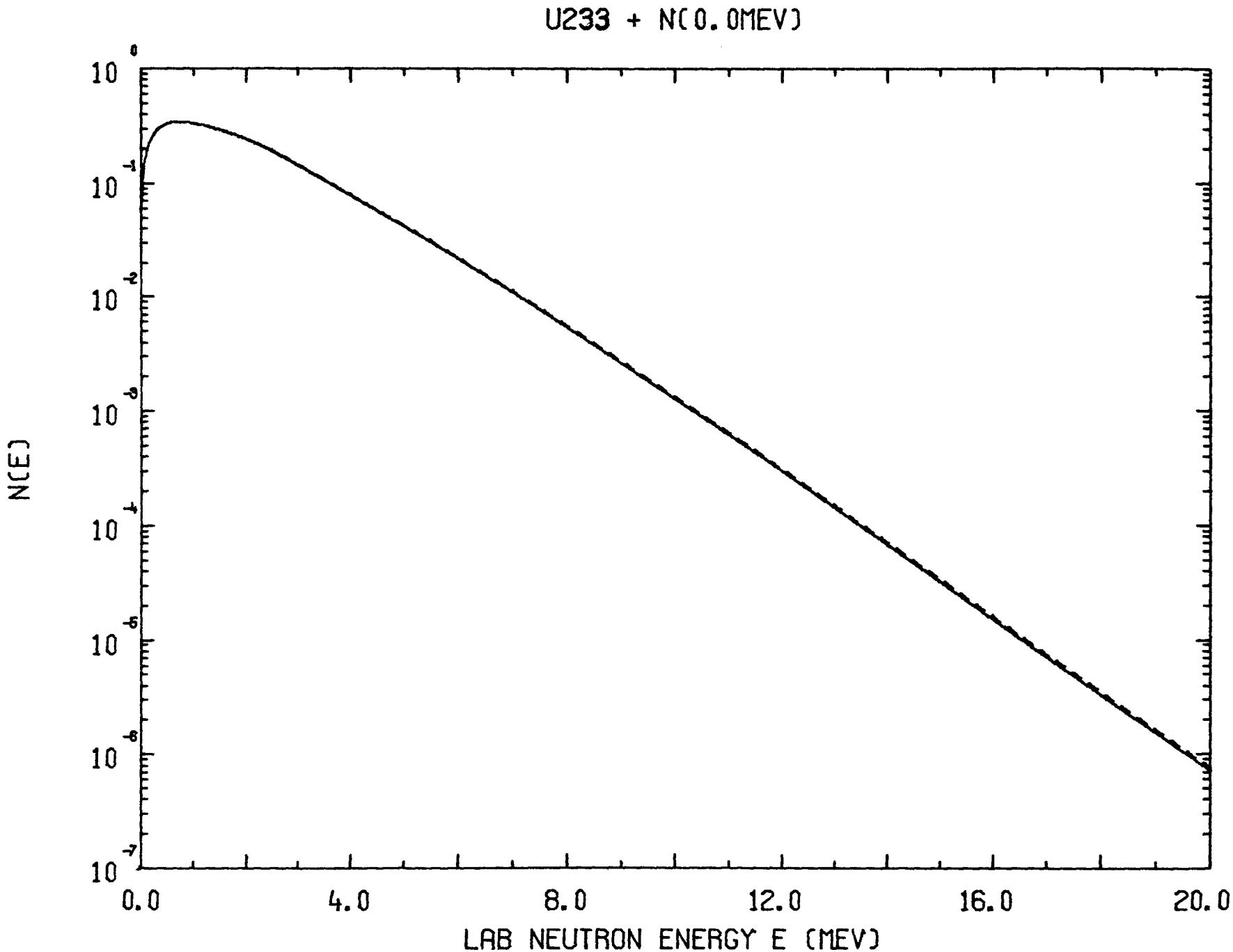
Figure 3 Fission Neutron Spectrum for $^{235}U(n,f)$ at $E_n = 0$ MeV.

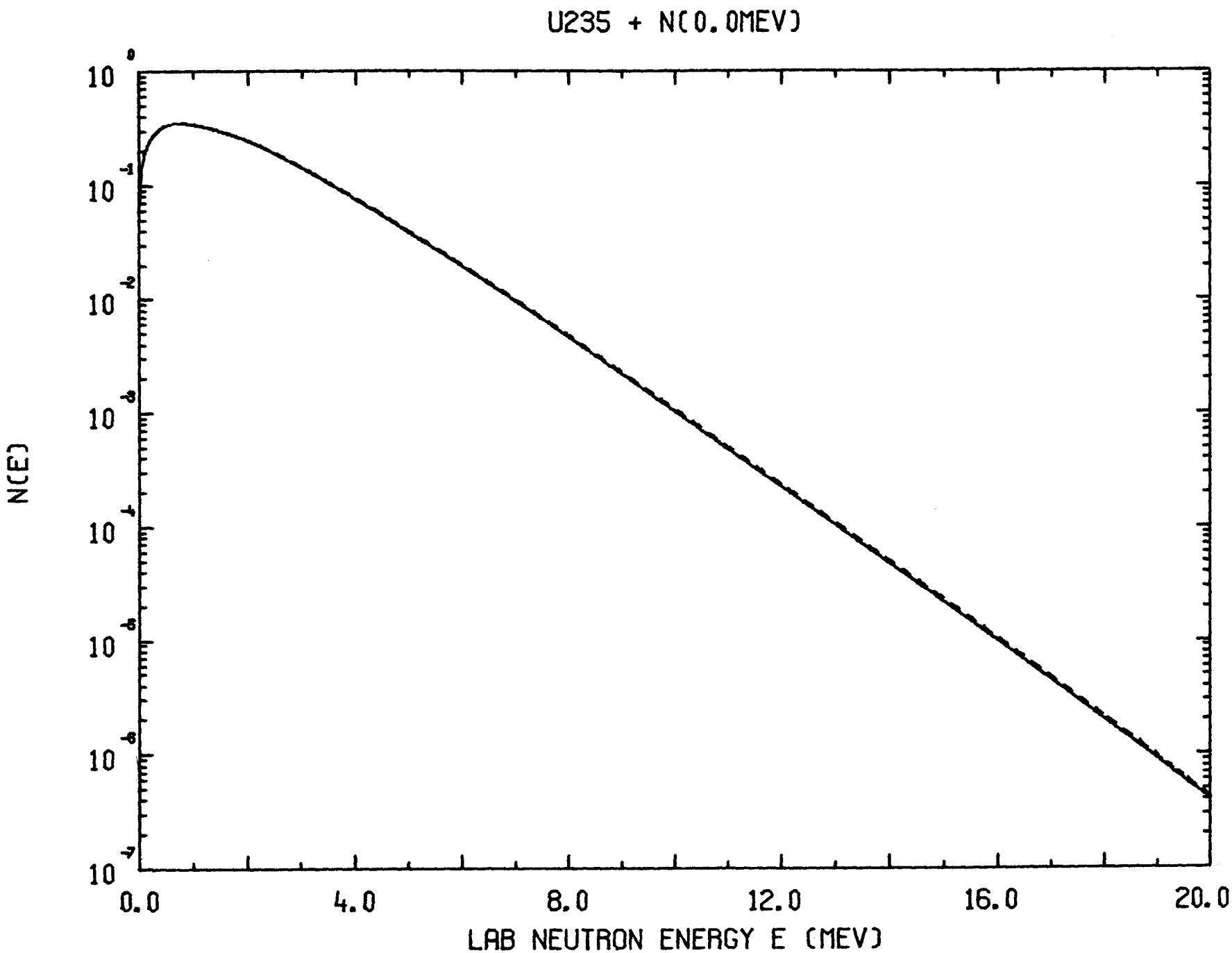
Figure 4 Fission Neutron Spectrum for $^{238}U(n,f)$ at $E_n = 2$ MeV.

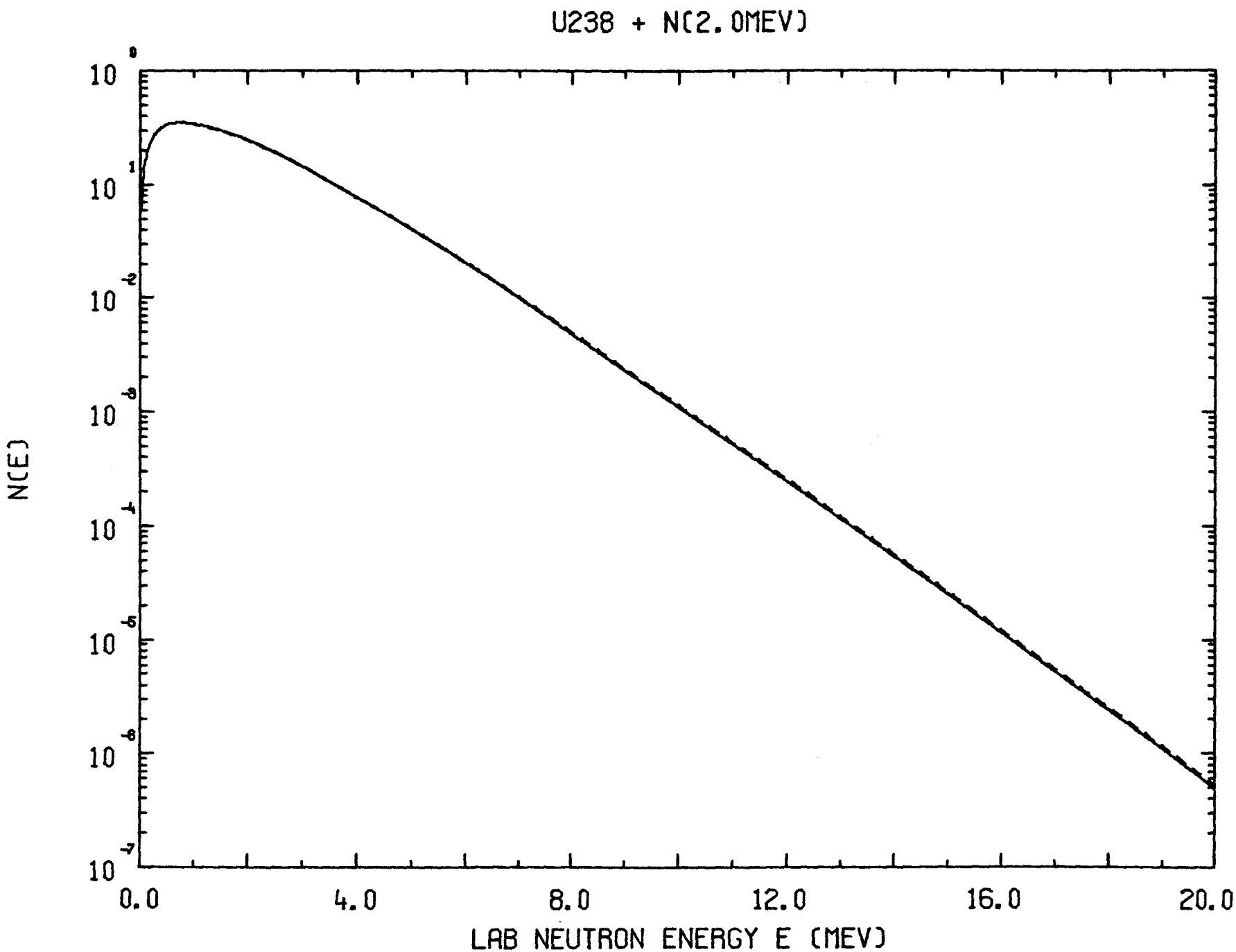
Figure 5 Fission Neutron Spectrum for $^{239}Pu(n,f)$ at $E_n = 0$ MeV

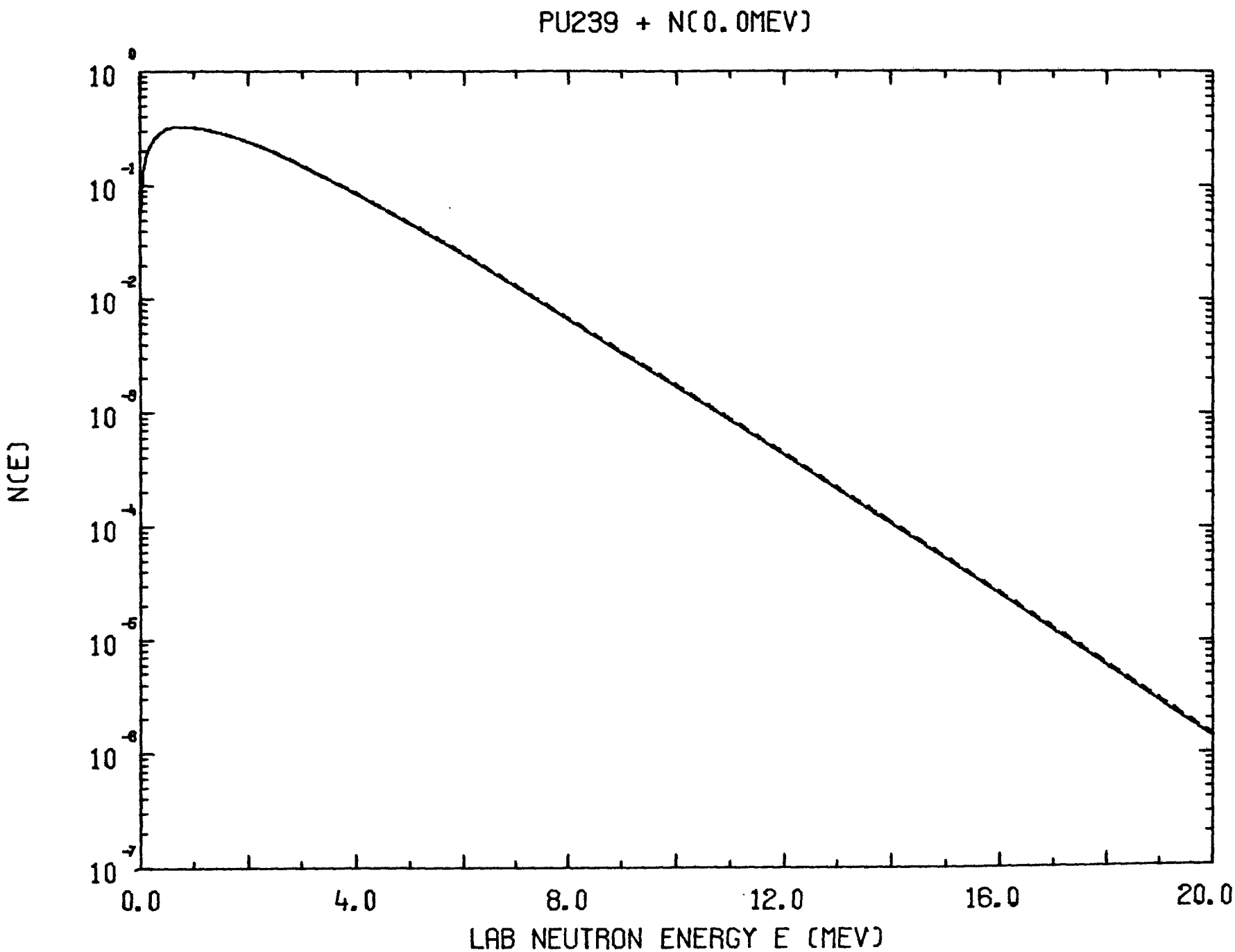
Figure 6 Fission Neutron Spectrum for ^{252}Cf Spontaneous Fission.

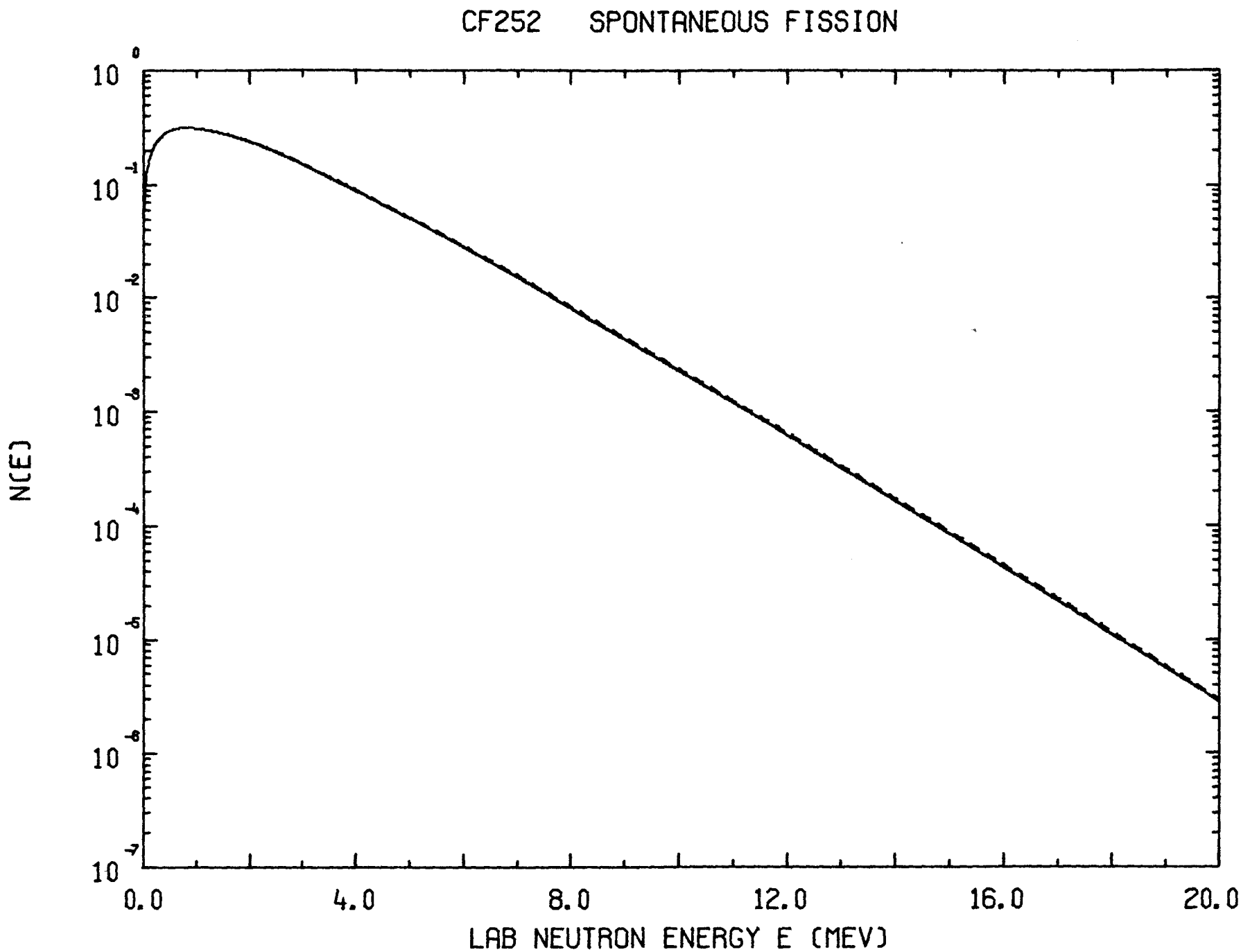

TH232 + N(2.0MEV)
N(E)
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
0.0
4.0
8.0
12.0
16.0
20.0
LAB NEUTRON ENERGY E (MEV)


U233 + N(0.0MEV)
N(E)
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
0.0
4.0
8.0
12.0
16.0
20.0
LAB NEUTRON ENERGY E (MEV)


U235 + N(0.0MEV)
N(E)
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
0.0
4.0
8.0
12.0
16.0
20.0
LAB NEUTRON ENERGY E (MEV)


U238 + N(2.0MEV)
N(E)
LAB NEUTRON ENERGY E (MEV)
0.0
4.0
8.0
12.0
16.0
20.0


PU239 + N(0.0MEV)
N(E)
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
0.0
4.0
8.0
12.0
16.0
20.0
LAB NEUTRON ENERGY E (MEV)


CF252 SPONTANEOUS FISSION
N(E)
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^-6
10^-7
0.0
4.0
8.0
12.0
16.0
20.0
LAB NEUTRON ENERGY E (MEV)