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MADNIX A CODE TO CALCULATE PROMPT FISSION NEUTRON SPECTRA AND AVERAGE PROMPT NEUTRON MULTIPLICITIES

A.C. Merchant

NOTA TÉCNICA IEAv-03/86 (Março 1986)

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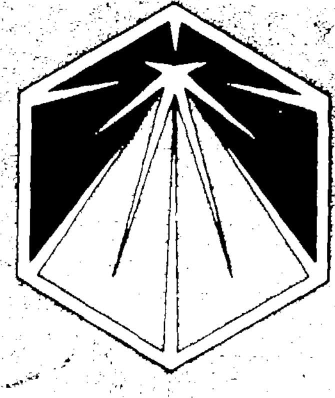

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ABSTRACT

A code has been written and tested on the CDC Cyber-170 to calculate the prompt fission neutron spectrum, N(E), as a function of both the fissioning nucleus and its excitation energy. In this note a brief description of the underlying physical principles involved and a detailed explanation of the required input data (together with a sample output for the fission of ^{235}U induced by 14 MeV neutrons) are presented. Weisskopf's standard nuclear evaporation theory provides the basis for the calculation. Two important refinements are that the distribution of fission-fragment residual nuclear temperature and the cooling of the fragments as neutrons are emitted are approximately taken into account, and also the energy dependence of the cross section for the inverse process of compound nucleus formation is included. This approach is then used to calculate the average number of prompt neutrons emitted per fission, $\bar{\nu}_p$. At high excitation energies, where fission is still possible after neutron emission, the consequences of the competition between first, second and third chance fission on N(E) and $\bar{\nu}_p$ are calculated. Excellent agreement with all the examples given in the original work of Madland and Nix is obtained.

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FIGURE CAPTIONS

(FIG.-1) The neutron energy spectra of Le Couteur, Maxwell and Madland/Nix in the fission fragment centre-of-mass system for the fission of ^{235}U induced by 14 MeV neutrons. The Le Couteur and Maxwellian temperatures are chosen such that the three distributions have equal first moments. Constant CN formation cross-sections and 100% first chance fission are assumed.

(FIG.-2) Ratios of the Le Couteur and Maxwellian spectra to that of Madland/Nix, corresponding to the curves in Figure 1.

(FIG.-3) The prompt fission neutron spectra of Watt, Maxwell and Madland/Nix in the laboratory system for the fission of ^{235}U induced by 14 MeV neutrons. The Watt and Maxwellian temperatures are chosen such that the three distributions have equal first moments. Constant CN formation cross-sections and 100% first chance fission are assumed.

(FIG.-4) Ratios of the Watt and Maxwellian spectra to that of Madland/Nix corresponding to the curves in Figure 3.

(FIG.-5) The neutron energy spectra of Madland/Nix, (i) with constant CN formation cross-sections and (ii) with energy dependent CN formation cross-sections calculated with the optical potential of Becchetti and Greenlees for the most probable light and heavy fission fragments (and the average of these two latter spectra) in the fission fragment centre-of-mass system for the fission of ^{235}U induced by 14 MeV neutrons. 100% first chance fission is assumed.

(FIG.-6) Ratios of the average light fragment, average heavy fragment and mean spectra to the spectrum calculated with constant CN formation cross-sections, corresponding to the curves in Figure 5.

(FIG.-7) The prompt fission neutron spectra of Madland/Nix, (i) with constant CN formation cross-sections and (ii) with energy dependent CN formation cross-sections calculated with the optical potential of Becchetti and Greenlees for the most probable light and heavy fission fragments (and the average of these two latter spectra) in the laboratory system for the fission of ^{235}U induced by 14 MeV neutrons. 100% first chance fission is assumed.

(FIG.-8) Ratios of the average light fragment, average heavy fragment and mean spectra to the spectrum calculated with constant CN formation cross-sections, corresponding to the curves in Figure 7.

(FIG.-9) Average prompt neutron multiplicity as a function of the incident neutron energy for the neutron induced fission of ^{235}U, predicted by the formalism of Madland/Nix using energy dependent CN formation cross-sections calculated with the optical potential of Becchetti and Greenlees and assuming 100% first chance fission.

(FIG.-10) The multiple chance fission components of the prompt fission neutron spectrum in the laboratory system for the fission of ^{235}U induced by 14 MeV neutrons.

(FIG.-11) Prompt fission neutron spectra in the laboratory system for the fission of ^{235}U induced by 14 MeV neutrons.

(FIG.-12) Ratios of the total multiple chance and first chance fission spectra calculated using energy dependent CN formation cross-sections to the first chance fission spectrum calculated using constant CN formation cross-sections, corresponding to the curves in Figure 11.

1. INTRODUCTION

Because of its importance in reactor design the prompt fission neutron spectrum, N(E), has been studied more or less continuously since the discovery of nuclear fission. In spite of this forty year effort, until recently most calculations which were to be used for practical purposes employed a Maxwellian or Watt spectrum /1,2/, and adjusted the available free parameters to obtain a best fit to a given set of data. Although this procedure can give a good description of the spectrum of interest, it does not provide a basis for predictions of the prompt fission neutron spectrum of that same nucleus at different excitation energies, nor of the spectra of other nuclei. In addition, it is objectionable on theoretical grounds since it neglects two important physical effects.

1) The distribution of fission-fragment residual nuclear temperature resulting from the initial distribution of fission-fragment excitation energy and the subsequent cooling of the fragments as neutrons are emitted.

2) The energy dependence of the cross-section for the inverse process of compound nucleus formation.

Furthermore, the Maxwellian spectrum even neglects the centre of mass motion of the fission fragments from which the neutrons are emitted.

To improve on this state of affairs, Madland and Nix/3/ presented a calculation which took into account the two neglected effects mentioned above. The distribution of fission-fragment residual nuclear temperature was taken to be triangular in shape, extending linearly from zero to a maximum value of T_m (to be discussed in the next section). Also the energy dependence of the compound nucleus formation cross-section, σ_c, for representative average fission fragments was accounted for by use of an optical model. They were then able to calculate N(E) for any fissioning nucleus at arbitrary excitation energies. Their approach also allows the calculation of $\bar{\nu}_p$ (the average number of prompt neutrons per fission) as a function of the excitation energy of any fissioning nucleus. If one is not interested in this predictive power, it is still possible to fit a given data set to high accuracy by parameter adjustment within their formalism (in preference to that of Watt or Maxwell as mentioned above).

2. THEORETICAL BASIS AND DETAILS OF THE CALCULATION

The nuclear evaporation theory of Weisskopf /4/ is used to calculate the centre of mass neutron energy spectrum, $\phi(\varepsilon,\sigma_c)$ for a given fission fragment. (Here the arguments ε and σ_c refer to the centre of mass neutron energy and the cross-section for the inverse process of compound nucleus formation respectively.) This spectrum can then be transformed into the laboratory system. According to the theory, $\phi(\varepsilon,\sigma_c)$ corresponding to a fixed residual nuclear temperature, T, is given approximately by

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

where the normalization constant k(T) given by

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

ensures that the energy integral of the spectrum from zero to infinity is unity. If $\sigma_c(\varepsilon)$ is assumed constant, then the product $k(T)\sigma_c(\varepsilon)$ in Eq. (1) reduces to $1/T^2$. This temperature T is not that of the evaporating compound nucleus at excitation energy E*, but the temperature of the residual nucleus at excitation energy E^*-B_n (i.e. diminished by the neutron separation energy B_n).

In principle, one should calculate separate spectra, integrated over the appropriate residual temperature distributions, for each possible pair of fission fragments. Then one should fold them all together with weights depending on the probability of each particular fragmentation, before finally comparing the results with an experimentally measured spectrum. However, this would involve a prohibitively long and expensive calculation, and the approximations actually employed will now be outlined.

2.1 Distribution of Fission Fragment Residual Nuclear Temperature

The initial distribution of the total excitation energy of the fission fragments is roughly Gaussian in shape with an average value given by

$$\langle E^*\rangle = \langle E_r\rangle + B_n + E_n - \langle E_f^{tot}\rangle \tag{3}$$

where $\langle E_r \rangle$ is the average energy release (which depends on the ground state mass differences between the fissioning compound nucleus and the two fission fragments), B_n and E_n are the separation and kinetic energies of the neutron inducing fission (both zero if fission occurs spontaneously) and $\langle E_f^{tot} \rangle$ is the sum of the average kinetic energies of the fission fragments. The values of B_n and $\langle E_f^{tot} \rangle$ are either taken directly from experiment or else taken from interpolations/extrapolations of experimental data /5,6,7/. The average energy release, $\langle E_r \rangle$, is determined approximately as follows. One obtains the most probable complementary pair of fission fragments, (A_1,Z_1) and (A_2,Z_2), from experiment and calculates E_r for them. Then a similar calculation is performed for the six neighbouring fragments having masses and charges (A_1-3,Z_1-1), (A_1-2,Z_1-1), (A_1-1,Z_1), (A_1+1,Z_1), (A_1+2,Z_1+1) and (A_1+3,Z_1+1) and their complements. An approximation for $\langle E_r \rangle$ is now obtained by averaging these seven values, with a weighting factor of 2 for the central pair. This recipe eliminates spurious odd-particle fluctuations and should be accurate to about 1 MeV.

The residual excitation energy E* can be related to the temperature T by the Fermi gas model, so that

$$E^* = aT^2 \tag{4}$$

and the level density parameter, a, may be approximated by A/11 (MeV) or left free and adjusted so as to optimize the fit to the experimental data of interest. Terrell /2/ showed that the temperature distribution obtained in this way from observed fission-fragment excitation energy distributions is roughly triangular in shape with a broad high-temperature cut-off. However, this diffuse cut-off may be replaced by a sharp cut-off, such that the temperature distribution, P(T), becomes

$$P(T) = \begin{cases} 2T/T_m^2 & T \le T_m \\ 0 & T > T_m \end{cases} \tag{5}$$

where the maximum temperature T_m is related to $\langle E^* \rangle$ by the relation

$$T_m = \left[\langle E^* \rangle / a \right]^{1/2} \tag{6}$$

Madland and Nix follow this latter procedure and further assume that the same temperature distribution P(T) applies to both light and heavy fragments.

2.2 Centre-of-mass Neutron Energy Spectrum

The neutron energy spectrum in the centre-of-mass system of a fission fragment is obtained by integrating Eq. (1) over the triangular temperature distribution of Eq. (5)

$$\Phi(\varepsilon,\sigma_c) = \int_0^{\infty} \phi(\varepsilon,\sigma_c)P(T)dT = \frac{2\sigma_c(\varepsilon)\varepsilon}{T_m^2} \int_0^{T_m} k(T)Te^{-\varepsilon/T}dT \ . \tag{7}$$

To evaluate this expression numerically, the compound nucleus formation cross-sections are generated by an optical model code (SCAT2) using the Becchetti-Greenlees potential /8/ for 79 energies between 1 keV and 40 MeV. Values of $\sigma_c(\varepsilon)$ at arbitrary energies are then found

(a) by making a cubic spline interpolation for energies between 1 keV and 40 MeV

(b) by extrapolation using the ' 1/v law ' for energies below 1 keV

(c) by setting σ_c equal to its value at 40 MeV for energies in excess of 40 MeV.

The temperature integral in Eq. (7) is then evaluated using a 96-order Gauss-Legendre quadrature at the required temperatures and the integral for k(T) (in Eq. (2)) is found using a 15-order Gauss-Laguerre quadrature. The use of a 32-order Gauss-Laguerre quadrature was not found to improve the accuracy significantly. Finally, the centre-of-mass neutron energy spectrum, $\phi(\varepsilon)$, is given by the average of the spectra calculated for neutron emission from the light and heavy average fission fragments

$$\Phi(\varepsilon) = \frac{1}{2}\left[\Phi(\varepsilon,\sigma_c^{\ell}) + \Phi(\varepsilon,\sigma_c^{h})\right] \tag{8}$$

In the special case of constant σ_c, Eq. (7) yields /9/

$$\Phi_{con}(\varepsilon) = \frac{2\varepsilon}{T_m^{\ 2}} E_1(\varepsilon/T_m) \tag{9}$$

with the exponential integral $E_1(x)$ given by

$$E_1(x) = \int_x^\infty \frac{e^{-u}}{u}\,du\ , \tag{10}$$

We also note two other commonly used parametrizations of $\Phi_{con}(\varepsilon)$. The Le Couteur spectrum /10/

$$\Phi_{lec}(\varepsilon) = \frac{\varepsilon^{5/11}\, e^{-\varepsilon/(\frac{11T_m}{12})}}{\Gamma(\frac{16}{11})(\frac{11T_m}{12})^{16/11}}\ , \tag{11}$$

and the centre-of-mass Maxwellian spectrum

$$\Phi_{max}(\varepsilon) = \frac{2\sqrt{\varepsilon}\, e^{-\varepsilon/(\frac{8T_m}{9})}}{\sqrt{\pi}\ (\frac{8T_m}{9})^{3/2}}\ . \tag{12}$$

The three approximate spectra derived from Eqs. (9), (11) and (12) may be calculated by the code MADNIX, if required, for comparison with the more accurate spectrum derived from Eq. (7) (in all cases making use of Eq. (8)).

2.3 Transformation to the Laboratory System

The centre-of-mass neutron spectrum of a fission fragment is transformed into the laboratory system under the assumption that neutrons are emitted isotropically from the fission fragment which moves with average kinetic energy per nucleon E_f. The general result is

$$N(E,E_f) = \frac{1}{4\sqrt{E_f}} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \frac{\phi(\varepsilon,\sigma_c)}{\sqrt{\varepsilon}}\,d\varepsilon\ , \tag{13}$$

with E the laboratory neutron energy. Substitution of Eq. (7) for $\Phi(\varepsilon,\sigma_c)$ leads to the result

$$N(E,E_f,\sigma_c) = \frac{1}{2\sqrt{E_f}T_m^2} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} d\varepsilon \left\{\sigma_c(\varepsilon)\sqrt{\varepsilon}\left[\int_0^{T_m} k(T)T\,e^{-\varepsilon/T}\,dT\right]\right\}. \tag{14}$$

The laboratory prompt fission neutron energy spectrum is given by the average of the spectra calculated for neutron emission from the light and heavy average fission fragments,

$$N(E) = \frac{1}{2}\left[N(E,E_f^{\ell},\sigma_c^{\ell}) + N(E,E_f^{h},\sigma_c^{h})\right] . \tag{15}$$

This entails a further integral over ε which is carried out using a 96-order Gauss-Legendre quadrature. If the cross-sections, σ_c, are assumed constant, Eq. (14) leads to the expression

$$N_{con}(E,E_f) = \frac{1}{3\sqrt{E_f T_m}}\left[u_2^{3/2}E_1(u_2) - u_1^{3/2}E_1(u_1) + \gamma(\tfrac{3}{2},u_2) - \gamma(\tfrac{3}{2},u_1)\right] \tag{16}$$

where

$$u_1 = \frac{(\sqrt{E} - \sqrt{E_f})}{T_m} , \tag{17}$$

$$u_2 = \frac{(\sqrt{E} + \sqrt{E_f})}{T_m} , \tag{18}$$

the incomplete gamma function $\gamma(a,x)$ is given by

$$\gamma(a,x) = \int_0^x u^{a-1}e^{-u}du , \tag{19}$$

and the exponential integral $E_1(x)$ was defined in Eq. (10). Two commonly used approximate expressions for $N(E,E_f)$ are the Watt spectrum

$$N_w(E) = \frac{e^{-E_f/T_w}}{\sqrt{\pi E_f T_w}} \sinh\left[\frac{2\sqrt{EE_f}}{T_w}\right] e^{-E/T_w} , \tag{20}$$

where the effective Watt temperature T_w is given by $T_w = 8T_m/9$ and the laboratory Maxwellian spectrum

$$N_{max}(E) = \frac{2\sqrt{E}\, e^{-E/T_m}}{\sqrt{\pi}\, T_m^{3/2}} \tag{21}$$

where the effective Maxwellian temperature $T_m = \frac{1}{3}(E_f^\ell + E_f^h) + 8T_m/9$ and

$$E_f^\ell = \frac{A_h}{A_\ell} \frac{\langle E_f^{tot} \rangle}{A} \tag{22}$$

$$E_f^h = \frac{A_\ell}{A_h} \frac{\langle E_f^{tot} \rangle}{A} \tag{23}$$

with A_ℓ, A_h and A the average mass numbers of the light fragment, heavy fragment and the compound nucleus undergoing fission respectively. The three approximate spectra derived from Eqs. (16), (20) and (21) may be calculated by the code MADNIX, if required, for comparison with the more accurate spectrum derived from Eq. (14) (in all cases making use of Eq. (15)).

2.4 Average Prompt Neutron Multiplicities

Conservation of energy dictates that the sum of the average fission fragment excitation energies $\langle E^* \rangle$ should equal the product of the average prompt neutron multiplicity $\bar{\nu}_p$ and the average energy removed per emitted neutron $\langle \eta \rangle$ plus the total average prompt gamma ray energy $\langle E_\gamma^{tot} \rangle$

$$\langle E^* \rangle = \bar{\nu}_p \langle \eta \rangle + \langle E_\gamma^{tot} \rangle \tag{24}$$

According to Terrell /11/, $\langle \eta \rangle$ is well represented by the sum of the average fission fragment neutron separation energy $\langle S_n \rangle$ and the average centre-of-mass energy of the emitted neutron $\langle \varepsilon \rangle$

$$\langle \eta \rangle = \langle S_n \rangle + \langle \varepsilon \rangle \tag{25}$$

Combining Eq.s (24) and (25) we obtain for $\bar{\nu}_p$

$$\bar{\nu}_p = \frac{\langle E^* \rangle - \langle E_\gamma^{tot} \rangle}{\langle S_n \rangle + \langle \varepsilon \rangle} \tag{26}$$

All the quantities appearing in this Equation have already been defined. An expression for $<E^*>$ is given by Eq. (3), $<\varepsilon>$ can be found by calculating the first moment of the distribution given in Eqs. (7) and (8) and $<E_\gamma^{tot}>$ can either be taken directly from experiment (when available) or from a least squares fit /7/ to the data from experiments on other actinide nuclei which takes the form

$$<E_\gamma^{tot}> \approx 0.028A + 0.09 \text{ MeV} \tag{27}$$

To average over pairing effects, $<S_n>$ is taken as one half of the average two-neutron separation energy, $<S_{2n}>$, for the most probable fission fragments and their nearest neighbours as described in section 2.1. As a check on the consistency of the calculation the prompt fission neutron spectrum N(E) and the average prompt neutron multiplicity $\bar{\nu}_p$ should be calculated and compared to experiment simultaneously.

2.5 Multiple Chance Fission

At high incident neutron energy (typically around 6 MeV) the compound nucleus has sufficient excitation energy to emit a neutron before fissioning. At some even higher energy (typically around 12 MeV) it becomes possible for two neutrons to be emitted prior to fission. To calculate a spectrum N(E) and the mean number of prompt neutrons $\bar{\nu}_p$ for comparison with experimental data corresponding to such energetic reactions, the neutrons emitted before fission must be combined with those actually emitted by the fission fragments in our calculation, because they will appear in coincidence with the fission event. The details of how to do this are given in section VI of reference 3. The only departure from that calculation in the code MADNIX is that the probabilities for the occurrence of first, second and third chance fission are read in directly, on the assumption that they will have been calculated by the code STAPRE as part of the evaluation of the actinide nucleus under consideration. The mean temperatures and excitation energies of the successive compound nuclei resulting from neutron emissions are then computed and the corresponding fission-spectra calculated as described in the preceding sub-sections. Finally, all contributions, weighted by the appropriate probabilities for first, second and third chance fission, are summed and the resulting spectrum may be compared with experiment.

3. DESCRIPTION OF THE INPUT DECK

All formats are free unless otherwise stated and the fortran naming convention for real and integral variables is assumed.

Card 1 Title (format A10)

Card 2 IZ, IA, IZL, IAL, IZH, IAH (all integers)

IZ,IA - the charge and mass numbers of the first compound nucleus

IZL,IAL - the charge and mass numbers of the most probable light fission fragments

IZH,IAH - the charge and mass numbers of the most probable heavy fission fragments

Card 3 MASDEF, MNEUT, EGTOT, ALEV

MASDEF - (=0) read mass defects of 7 nearest neighbours to most probable light and heavy fragments from card 3b below
(=1) obtain these values from the CERN table in the code

MNEUT - (=0) read the average neutron separation energy from card 3a below
(=1) obtain this value from the CERN table in the code

EGTOT - average total prompt gamma ray energy in MeV (if EGTOT < 0.01 the formula EGTOT = 0.028*A + 0.09 will be used automatically)

ALEV - level density parameter (a = A/ALEV so start with ALEV = 11)

Card 3a $<S_n>$ in MeV is now read if MNEUT = 0

Card 3b DEFL(7), DEFH(7) in MeV, the mass defects of the 7 nearest neighbours to the most probable light and heavy fission fragments as described in section 2.1. Only include this card if MASDEF = 0.

Card 4 EN, AVTOT, ISPONT, IMULT

EN - the incident energy of the neutron initiating fission (in MeV).

AVTOT - the average total kinetic energy of the most probable fission fragments (in MeV).

ISPONT - (=0) neutron induced fission
(=1) spontaneous fission (will set $E_n=B_n=0$)

IMULT - (=0) no neutrons emitted prior to fission
(=1) 1 neutron may be emitted prior to fission
(=2) 2 neutrons may be emitted prior to fission

N.B. If multiple chance fission is to be included, you must set IENCM=IENLAB=1 on card 6 and you must supply energy dependent CN formation cross-sections on cards 8a and 8b.

Card 5 EMAX, ESTEP, EBINL, EBINH (all in MeV)

EMAX - maximum energy to which N(E) will be calculated

ESTEP - in the energy region from 1 MeV to EMAX MeV you may choose the energy interval between successive calculations of N(E) by means of the variable ESTEP.

EBINL - lower limit on experimental energy . (use 0 if unknown)

EBINH - upper limit on experimental energy (use EMAX if unknown).

N.B. The code contains various parameter statements which determine the dimensions of various matrices. It is necessary to set the value of MATDIM = 28 + (EMAX - 1)/ESTEP in these parameter statements. e.g. if EMAX=15 MeV and ESTEP=1 MeV then MATDIM=42.

Card 6 ICM, ILAB, IENCM, IENLAB (all integers)

These are switches to decide what to calculate. (=0) do not calculate, (=1) do calculate.

ICM - centre-of-mass neutron energy spectrum with constant CN formation cross-sections

ILAB - laboratory neutron energy spectrum with constant CN formation cross-sections

IENCM - centre-of-mass neutron energy spectrum with energy dependent CN formation cross-sections

IENLAB - laboratory neutron energy spectrum with energy dependent CN formation cross-sections

Card 7 IFIG

If you have set ICM=ILAB=IENCM=IENLAB=1 you may set IFIG=1 to produce a multifile (tape 1) for use with PPFT/CPDSCUT to create graphs similar to figures 3,4,7,8,15,16,17,18 of Ref. 3. If you do not want this option set IFIG=0.

Card 8 (EDATAH(I), SDATAH(I), I=1,79)

EDATAH - the energies at which the CN formation cross-sections of the most probable heavy fission-fragment have been calculated by SCAT2. In the tests I have used energies as follows

0.001 to 0.01 MeV in steps of 0.0005 MeV
0.01 to 0.1 MeV in steps of 0.005 MeV
0.1 to 1.0 MeV in steps of 0.05 MeV
1.0 to 10.0 MeV in steps of 0.5 MeV
10.0 to 40.0 MeV in steps of 5.0 MeV

SDATAH - the CN formation cross-sections for the most probable heavy fission fragment produced by SCAT2 corresponding to the energies EDATAH. Use the optical potential of Becchetti-Greenlees unless otherwise stated.

Card 8a (EDATAL(I), SDATAL(I), I=1,79)

Similar quantities to those on card 8 for the most probable light fission fragment.

N.B. If multiple chance fission is not to be considered the input deck ends here

Card 9 (EDAT1(I), SDAT1(I), I=1,79)

Similar quantities to those on card 8 for the first compound nucleus (neutron + target) in the multiple fission calculation. The optical potential of Madland should be chosen in the run of SCAT2 to produce the 'best' values of these cross-sections.

Card 10 IZ1, IA1, IZH1, IAH1, IZL1, IAL1, AVTOT1, S2N1, EGTOT2

IZ1, IA1 - charge and mass numbers of the second compound nucleus in the multiple chance fission calculation (n.b. this is just the target nucleus)

IZH1, IAH1 - the charge and mass numbers of the most probable heavy fragments resulting from fission of the second CN.

IZL1, IAL1 - the charge and mass numbers of the most probable light fragments resulting from fission of the second CN.

AVTOT1 - the total average kinetic energy of the fission fragments resulting from fission of the second CN (in MeV).

S2N1 - the average neutron separation energy for the second CN in MeV (see section 2.4 for a discussion)

EGTOT1 - the total prompt gamma ray energy associated with the second CN (in MeV). If EGTOT1 < 0.01 MeV the value 0.028*A + 0.09 MeV will be used automatically.

Card 11 (EDATH1(I), SDATH1(I), I=1,79)

Similar quantities to those appearing on card 8 for the most probable heavy fission-fragment resulting from fission of the second CN.

Card 11a (EDATL1(I), SDATL1(I), I=1,79)

Similar quantities to those appearing on card 8 for the most probable light fission-fragment resulting from fission of the second CN.

Card 12 (EDAT2(I), SDAT2(I), I=1,79)

Similar quantities to those appearing on card 8 for the second CN. Use the optical potential of Madland to generate them with SCAT2.

Card 13 IZ2, IA2, IZH2, IAH2, IZL2, IAL2, AVTOT2, S2N2, EGTOT2

Similar quantities to those appearing on card 10 for the third CN (i.e. target nucleus less one neutron).

Card 14 (EDATH2(I), SDATH2(I), I=1,79)

Similar quantities to those appearing on card 8 for the most probable heavy fragment resulting from fission of the third CN.

Card 14a (EDATL2(I), SDATL2(I), I=1,79)

Similar quantities to those appearing on card 8 for the most probable light fragment resulting from fission of the third CN.

Card 15 PF1, PF2, PF3

PF1, PF2, PF3 - probabilities of first, second and third chance fission (these quantities may be obtained from the code STAPRE).

NOTE If multiple chance fission is not to be considered card 8 will be the last card in the input deck. If only first and second chance fission (but not third chance fission) are to be considered cards 13, 14 and 14a should be omitted and PF3 set equal to 0 on card 15. If IMULT = 2 on card 4, a multifile (tape2) will be produced for use with PPFT/CPDSCUT to create graphs similar to those in figures 40, 41 and 42 of reference 3.

4. EXAMPLE OF THE FISSION OF ^{235}U INDUCED BY 14 MeV NEUTRONS

An example of the output (listing and graphs) from a run of MADNIX for the fission of ^{235}U induced by 14 MeV neutrons is now presented. This run required a CPU time of about 800 seconds and a maximum field length of 70000 (i.e. it can run in class I1).

```
U235+N    14.00000 MEV
CM NEUTRON SPECTRA, USING CONSTANT SIGMA
 MADLAND/NIX    LECOUTEUR      MAXWELL    ENERGY(MEV)
  7.9107E-03   3.8252E-02   2.9021E-02       .00100
  1.4158E-02   5.2374E-02   4.1006E-02       .00200
  1.9779E-02   6.2921E-02   5.0179E-02       .00300
  2.4994E-02   7.1650E-02   5.7891E-02       .00400
  2.9907E-02   7.9232E-02   6.4667E-02       .00500
  3.4580E-02   8.6005E-02   7.0778E-02       .00600
  3.9054E-02   9.2170E-02   7.6382E-02       .00700
  4.3357E-02   9.7854E-02   8.1585E-02       .00800
  4.7512E-02   1.0315E-01   8.6458E-02       .00900
  5.1535E-02   1.0812E-01   9.1056E-02       .01000
  8.6604E-02   1.4691E-01   1.2765E-01       .02000
  1.1557E-01   1.7516E-01   1.5499E-01       .03000
  1.4064E-01   1.9795E-01   1.7741E-01       .04000
  1.6286E-01   2.1723E-01   1.9663E-01       .05000
  1.8284E-01   2.3401E-01   2.1353E-01       .06000
  2.0099E-01   2.4889E-01   2.2863E-01       .07000
  2.1759E-01   2.6223E-01   2.4230E-01       .08000
  2.3287E-01   2.7433E-01   2.5476E-01       .09000
  2.4699E-01   2.8536E-01   2.6621E-01       .10000
  3.4504E-01   3.5935E-01   3.4505E-01       .20000
  3.9685E-01   3.9705E-01   3.8732E-01       .30000
  4.2353E-01   4.1583E-01   4.0990E-01       .40000
  4.3470E-01   4.2291E-01   4.2002E-01       .50000
  4.3572E-01   4.2221E-01   4.2169E-01       .60000
  4.2993E-01   4.1614E-01   4.1745E-01       .70000
  4.1951E-01   4.0634E-01   4.0901E-01       .80000
  4.0600E-01   3.9393E-01   3.9760E-01       .90000
  3.9044E-01   3.7976E-01   3.8412E-01      1.00000
  2.2315E-01   2.2345E-01   2.2717E-01      2.00000
  1.1328E-01   1.1537E-01   1.1635E-01      3.00000
  5.5238E-02   5.6456E-01   5.6185E-01      4.00000
  2.6426E-02   2.6829E-02   2.6269E-02      5.00000
  1.2508E-02   1.2516E-02   1.2034E-02      6.00000
  5.8811E-03   5.7641E-03   5.4358E-03      7.00000
  2.7530E-03   2.6299E-03   2.4302E-03      8.00000
  1.2847E-03   1.1913E-03   1.0779E-03      9.00000
  5.9817E-04   5.3664E-04   4.7516E-04     10.00000
  2.7803E-04   2.4063E-04   2.0840E-04     11.00000
  1.2906E-04   1.0749E-04   9.1028E-05     12.00000
  5.9848E-05   4.7865E-05   3.9622E-05     13.00000
  2.7729E-05   2.1257E-05   1.7195E-05     14.00000
  1.2839E-05   9.4181E-06   7.4431E-06     15.00000
MEAN  AND  MEAN  SQUARE  ENERGIES  OF  MADLAND/NIX  C.M.
DISTRIBUTION USING CONSTANT X-SECTIONS
 AVERAGE E= 1.72054  AVERAGE E**2= 4.99545
```

```
 LAB NEUTRON SPECTRA, USING CONSTANT SIGMA
  MADLAND/NIX      WATT         MAXWELL    ENERGY(MEV)
   1.5553E-02   1.4703E-02   1.6564E-02      .00100
   2.1982E-02   2.0783E-02   2.3412E-02      .00200
   2.6907E-02   2.5442E-02   2.8656E-02      .00300
   3.1052E-02   2.9364E-02   3.3069E-02      .00400
   3.4697E-02   3.2814E-02   3.6950E-02      .00500
   3.7987E-02   3.5929E-02   4.0453E-02      .00600
   4.1008E-02   3.8790E-02   4.3668E-02      .00700
   4.3814E-02   4.1448E-02   4.6655E-02      .00800
   4.6445E-02   4.3941E-02   4.9455E-02      .00900
   4.8929E-02   4.6296E-02   5.2099E-02      .01000
   6.8802E-02   6.5161E-02   7.3239E-02      .02000
   8.3782E-02   7.9425E-02   8.9163E-02      .03000
   9.6189E-02   9.1274E-02   1.0234E-01      .04000
   1.0692E-01   1.0156E-01   1.1374E-01      .05000
   1.1645E-01   1.1072E-01   1.2385E-01      .06000
   1.2506E-01   1.1902E-01   1.3297E-01      .07000
   1.3292E-01   1.2663E-01   1.4130E-01      .08000
   1.4017E-01   1.3366E-01   1.4898E-01      .09000
   1.4689E-01   1.4021E-01   1.5610E-01      .10000
   1.9589E-01   1.8890E-01   2.0790E-01      .20000
   2.2616E-01   2.2026E-01   2.3980E-01      .30000
   2.4631E-01   2.4201E-01   2.6078E-01      .40000
   2.6029E-01   2.5733E-01   2.7459E-01      .50000
   2.7039E-01   2.6796E-01   2.8329E-01      .60000
   2.7710E-01   2.7499E-01   2.8817E-01      .70000
   2.8090E-01   2.7917E-01   2.9013E-01      .80000
   2.8231E-01   2.8107E-01   2.8982E-01      .90000
   2.8188E-01   2.8111E-01   2.8771E-01     1.00000
   2.2992E-01   2.3023E-01   2.2335E-01     2.00000
   1.5624E-01   1.5837E-01   1.5016E-01     3.00000
   9.8233E-02   1.0034E-01   9.5181E-02     4.00000
   5.9230E-02   6.0427E-02   5.8416E-02     5.00000
   3.4782E-02   3.5123E-02   3.5127E-02     6.00000
   2.0053E-02   1.9877E-02   2.0827E-02     7.00000
   1.1403E-02   1.1015E-02   1.2222E-02     8.00000
   6.4138E-03   6.0002E-03   7.1162E-03     9.00000
   3.5759E-03   3.2219E-03   4.1177E-03    10.00000
   1.9789E-03   1.7090E-03   2.3706E-03    11.00000
   1.0881E-03   8.9699E-04   1.3592E-03    12.00000
   5.9502E-04   4.6643E-04   7.7658E-04    13.00000
   3.2377E-04   2.4055E-04   4.4238E-04    14.00000
   1.7540E-04   1.2315E-04   2.5136E-04    15.00000

MEAN  AND MEAN SQUARE ENERGIES OF MADLAND/NIX LAB DISTRIBUTION
USING CONSTANT X-SECTIONS
 AVERAGE E= 2.50094  AVERAGE E**2=10.15925
```

```
CM NEUTRON SPECTRA, USING SIGMA(E)
LIGHT FRAGMENT, HEAVY FRAGMENT, AVERAGE, ENERGY (MEV)
   2.9609E-02   5.4606E-02   4.2108E-02      .00100
   4.1632E-02   7.5563E-02   5.8598E-02      .00200
   5.0769E-02   9.0852E-02   7.0810E-02      .00300
   5.8418E-02   1.0318E-01   8.0797E-02      .00400
   6.5119E-02   1.1360E-01   8.9359E-02      .00500
   7.1153E-02   1.2267E-01   9.6909E-02      .00600
   7.6684E-02   1.3071E-01   1.0370E-01      .00700
   8.1820E-02   1.3795E-01   1.0988E-01      .00800
   8.6636E-02   1.4453E-01   1.1558E-01      .00900
   9.1187E-02   1.5057E-01   1.2088E-01      .01000
   1.2816E-01   1.9346E-01   1.6081E-01      .02000
   1.5761E-01   2.2107E-01   1.8934E-01      .03000
   1.8381E-01   2.4221E-01   2.1301E-01      .04000
   2.0818E-01   2.5984E-01   2.3401E-01      .05000
   2.3122E-01   2.7521E-01   2.5322E-01      .06000
   2.5310E-01   2.8891E-01   2.7101E-01      .07000
   2.7387E-01   3.0127E-01   2.8757E-01      .08000
   2.9351E-01   3.1247E-01   3.0299E-01      .09000
   3.1202E-01   3.2267E-01   3.1734E-01      .10000
   4.3985E-01   3.8788E-01   4.1387E-01      .20000
   4.9308E-01   4.1836E-01   4.5572E-01      .30000
   5.0600E-01   4.3390E-01   4.6995E-01      .40000
   4.9816E-01   4.4087E-01   4.6952E-01      .50000
   4.8014E-01   4.4156E-01   4.6085E-01      .60000
   4.5755E-01   4.3696E-01   4.4725E-01      .70000
   4.3334E-01   4.2784E-01   4.3059E-01      .80000
   4.0905E-01   4.1498E-01   4.1201E-01      .90000
   3.8546E-01   3.9922E-01   3.9234E-01     1.00000
   2.0741E-01   2.1486E-01   2.1113E-01     2.00000
   1.0258E-01   1.0506E-01   1.0382E-01     3.00000
   4.7713E-02   5.0474E-02   4.9094E-02     4.00000
   2.2042E-02   2.3425E-02   2.2734E-02     5.00000
   1.0209E-02   1.0705E-02   1.0457E-02     6.00000
   4.7128E-03   4.8994E-03   4.8061E-03     7.00000
   2.1805E-03   2.2723E-03   2.2264E-03     8.00000
   1.0053E-03   1.0532E-03   1.0292E-03     9.00000
   4.6295E-04   4.8649E-04   4.7472E-04    10.00000
   2.1285E-04   2.2407E-04   2.1846E-04    11.00000
   9.7697E-05   1.0300E-04   1.0035E-04    12.00000
   4.4793E-05   4.7286E-05   4.6040E-05    13.00000
   2.0527E-05   2.1700E-05   2.1113E-05    14.00000
   9.4067E-06   9.9609E-06   9.6838E-06    15.00000
MEAN AND MEAN SQUARE ENERGIES OF CM DISTRIBUTION USING  ENERGY
DEPENDENT X-SECTIONS
 AVERAGE  E=  1.60629   AVERAGE  E**2=  4.46205  NORMALIZATION
INTEGRAL =  1.00074
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```
LAB NEUTRON SPECTRA USING SIGMA(E)
LIGHT FRAGMENT, HEAVY FRAGMENT, AVERAGE, ENERGY (MEV)
   1.1399E-02   1.9723E-02   1.5561E-02     .00100
   1.6121E-02   2.7880E-02   2.2000E-02     .00200
   1.9745E-02   3.4130E-02   2.6937E-02     .00300
   2.2799E-02   3.9392E-02   3.1096E-02     .00400
   2.5491E-02   4.4022E-02   3.4757E-02     .00500
   2.7924E-02   4.8203E-02   3.8063E-02     .00600
   3.0162E-02   5.2041E-02   4.1102E-02     .00700
   3.2245E-02   5.5610E-02   4.3927E-02     .00800
   3.4201E-02   5.8956E-02   4.6579E-02     .00900
   3.6051E-02   6.2118E-02   4.9084E-02     .01000
   5.0987E-02   8.7449E-02   6.9218E-02     .02000
   6.2444E-02   1.0661E-01   8.4526E-02     .03000
   7.2096E-02   1.2252E-01   9.7310E-02     .04000
   8.0589E-02   1.3633E-01   1.0846E-01     .05000
   8.8254E-02   1.4862E-01   1.1844E-01     .06000
   9.5288E-02   1.5975E-01   1.2752E-01     .07000
   1.0182E-01   1.6993E-01   1.3587E-01     .08000
   1.0793E-01   1.7933E-01   1.4363E-01     .09000
   1.1369E-01   1.8808E-01   1.5088E-01     .10000
   1.5892E-01   2.5211E-01   2.0552E-01     .20000
   1.9072E-01   2.9178E-01   2.4125E-01     .30000
   2.1412E-01   3.1736E-01   2.6574E-01     .40000
   2.3128E-01   3.3237E-01   2.8183E-01     .50000
   2.4363E-01   3.3932E-01   2.9147E-01     .60000
   2.5234E-01   3.4085E-01   2.9660E-01     .70000
   2.5834E-01   3.3891E-01   2.9863E-01     .80000
   2.6232E-01   3.3454E-01   2.9843E-01     .90000
   2.6472E-01   3.2836E-01   2.9654E-01    1.00000
   2.3533E-01   2.3242E-01   2.3388E-01    2.00000
   1.6263E-01   1.4111E-01   1.5187E-01    3.00000
   1.0383E-01   7.9438E-02   9.1636E-02    4.00000
   6.4453E-02   4.3605E-02   5.4029E-02    5.00000
   3.8841E-02   2.3609E-02   3.1225E-02    6.00000
   2.2701E-02   1.2566E-02   1.7633E-02    7.00000
   1.2979E-02   6.5841E-03   9.7815E-03    8.00000
   7.3314E-03   3.4198E-03   5.3756E-03    9.00000
   4.1127E-03   1.7716E-03   2.9422E-03   10.00000
   2.2934E-03   9.1797E-04   1.6057E-03   11.00000
   1.2708E-03   4.7578E-04   8.7327E-04   12.00000
   6.9982E-04   2.4562E-04   4.7272E-04   13.00000
   3.8364E-04   1.2624E-04   2.5494E-04   14.00000

   2.0909E-04   6.4617E-05   1.3685E-04   15.00000
MEAN AND MEAN SQUARE ENERGIES OF LAB DISTRIBUTION USING ENERGY
DEPENDENT X-SECTIONS
 AVERAGE E= 2.37856  AVERAGE E**2= 9.22374
NORMALIZATION INTEGRAL =   .99892
```

MULTIPLE CHANCE FISSION CALCULATION

(These quantities may be compared with table V of reference 3)

```
LIST OF QUANTITIES USED IN CALCULATION


 FIRST   STEP:   EF1L=     1.062    EF1H=      .499    MAXIMUM
TEMPERATURE=   1.290  TEMPERATURE=    .808

 AVERAGE NUMBER OF NEUTRONS EMITTED=   4.393

 SCRIPT EPSILON1=   1.607  SQUARE OF SCRIPT EPSILON1=   3.839

 SECOND  STEP:  EF2L=   1.059  EF2H=    .505  MAX TEMPERATURE=
1.173  TEMPERATURE=    .576

 SCRIPT EPSILON2=   1.155  SQUARE OF SCRIPT EPSILON2=   1.994

 ONE NEUTRON EMITTED, AVERAGE CM ENERGY= 1.45953


NORMALIZATION INTEGRAL =  1.00066

 AVERAGE NUMBER OF NEUTRONS EMITTED=   3.437

 ONE NEUTRON EMITTED: LAB ENERGY= 2.23527  LAB ENERGY SQUARED=
8.11009


NORMALIZATION INTEGRAL =   .99877

 THIRD  STEP:   EF3L=   1.076  EF3H=    .503  MAX TEMPERATURE=
1.036  TEMPERATURE=    .000

 TWO NEUTRONS EMITTED, AVERAGE CM ENERGY= 1.28744


NORMALIZATION INTEGRAL =  1.00051

 AVERAGE NUMBER OF NEUTRONS EMITTED=   2.464

 TWO  NEUTRONS  EMITTED:  LAB  ENERGY=  2.07074  LAB  ENERGY
SQUARED= 6.91168
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```
LAB NEUTRON SPECTRA INCLUDING MULTIPLE CHANCE FISSION

FIRST CHANCE  SECOND CHANCE THIRD CHANCE   TOTAL   ENERGY(MEV)
  3.701E-03     1.132E-02    7.825E-03   2.285E-02     .00100
  5.232E-03     1.590E-02    1.094E-02   3.207E-02     .00200
  6.407E-03     1.936E-02    1.327E-02   3.904E-02     .00300
  7.396E-03     2.224E-02    1.520E-02   4.484E-02     .00400
  8.266E-03     2.475E-02    1.686E-02   4.988E-02     .00500
  9.053E-03     2.700E-02    1.834E-02   5.440E-02     .00600
  9.775E-03     2.905E-02    1.968E-02   5.850E-02     .00700
  1.045E-02     3.093E-02    2.091E-02   6.229E-02     .00800
  1.108E-02     3.270E-02    2.205E-02   6.583E-02     .00900
  1.167E-02     3.435E-02    2.312E-02   6.914E-02     .01000
  1.646E-02     4.734E-02    3.134E-02   9.514E-02     .02000
  2.010E-02     5.701E-02    3.735E-02   1.145E-01     .03000
  2.314E-02     6.506E-02    4.237E-02   1.306E-01     .04000
  2.580E-02     7.213E-02    4.678E-02   1.447E-01     .05000
  2.817E-02     7.851E-02    5.079E-02   1.575E-01     .06000
  3.033E-02     8.434E-02    5.446E-02   1.691E-01     .07000
  3.232E-02     8.973E-02    5.786E-02   1.799E-01     .08000
  3.416E-02     9.474E-02    6.102E-02   1.899E-01     .09000
  3.589E-02     9.942E-02    6.396E-02   1.993E-01     .10000
  4.888E-02     1.333E-01    8.451E-02   2.667E-01     .20000
  5.738E-02     1.531E-01    9.516E-02   3.056E-01     .30000
  6.320E-02     1.649E-01    1.005E-01   3.286E-01     .40000
  6.703E-02     1.714E-01    1.027E-01   3.412E-01     .50000
  6.932E-02     1.743E-01    1.028E-01   3.464E-01     .60000
  7.054E-02     1.746E-01    1.015E-01   3.466E-01     .70000
  7.102E-02     1.733E-01    9.929E-02   3.436E-01     .80000
  7.098E-02     1.707E-01    9.653E-02   3.382E-01     .90000
  7.053E-02     1.674E-01    9.338E-02   3.313E-01    1.00000
  5.562E-02     1.176E-01    5.830E-02   2.315E-01    2.00000
  3.612E-02     7.006E-02    3.153E-02   1.377E-01    3.00000
  2.179E-02     3.914E-02    1.608E-02   7.701E-02    4.00000
  1.285E-02     2.144E-02    8.052E-03   4.234E-02    5.00000
  7.426E-03     1.156E-02    3.982E-03   2.297E-02    6.00000
  4.194E-03     6.115E-03    1.936E-03   1.224E-02    7.00000
  2.326E-03     3.184E-03    9.290E-04   6.439E-03    8.00000
  1.278E-03     1.643E-03    4.418E-04   3.363E-03    9.00000
  6.997E-04     8.446E-04    2.092E-04   1.754E-03   10.00000
  3.819E-04     4.330E-04    9.873E-05   9.136E-04   11.00000
  2.077E-04     2.213E-04    4.642E-05   4.754E-04   12.00000
  1.124E-04     1.126E-04    2.173E-05   2.467E-04   13.00000
  6.063E-05     5.708E-05    1.014E-05   1.278E-04   14.00000
  3.255E-05     2.881E-05    4.706E-06   6.606E-05   15.00000

AVERAGE E = 2.07620  AVERAGE E**2 = 7.12147
AVERAGE NUMBER OF NEUTRONS EMITTED= 4.43342
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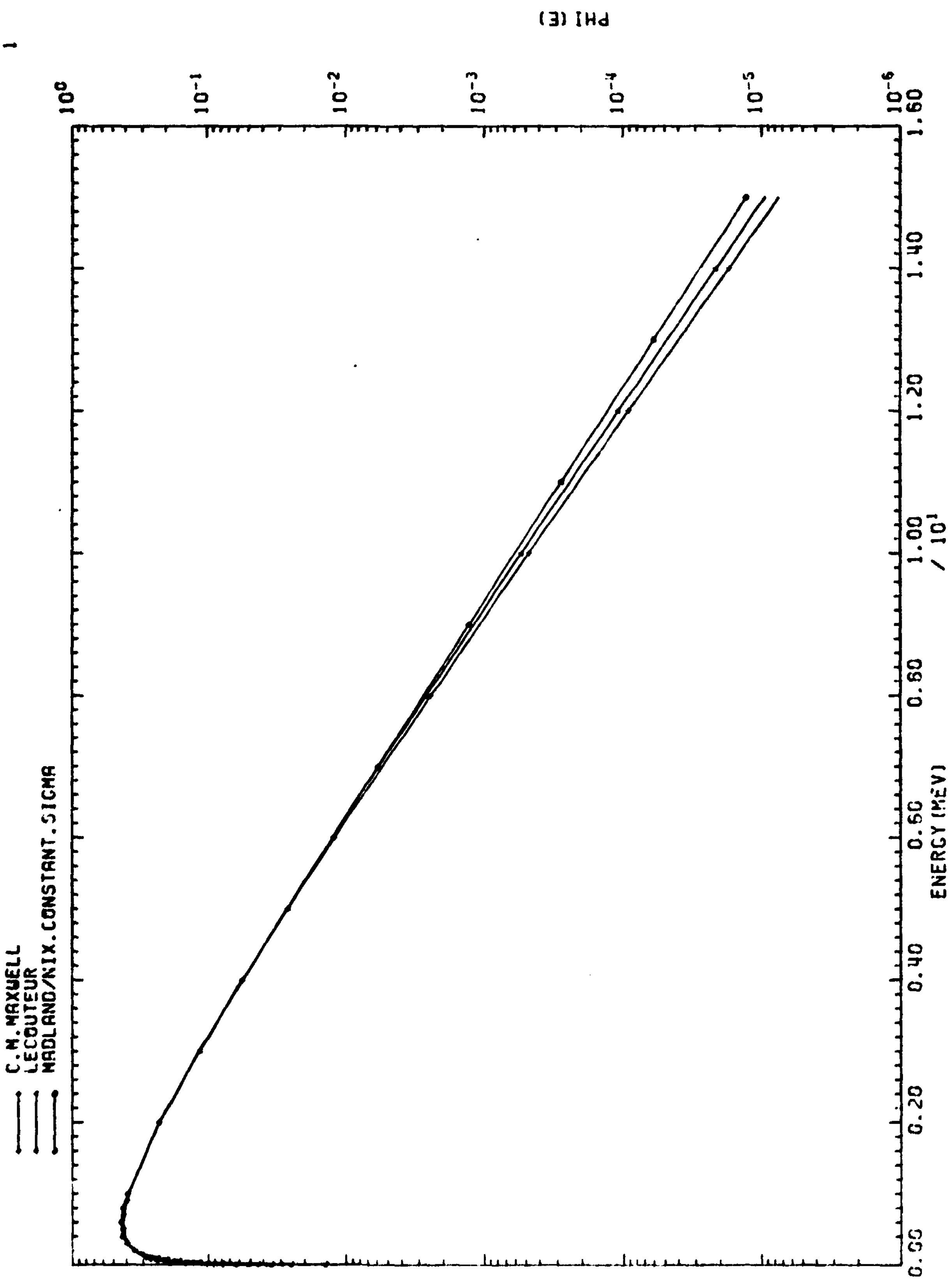

Figure 1 - Centre of mass neutron spectra (with constant CN formation cross-sections)

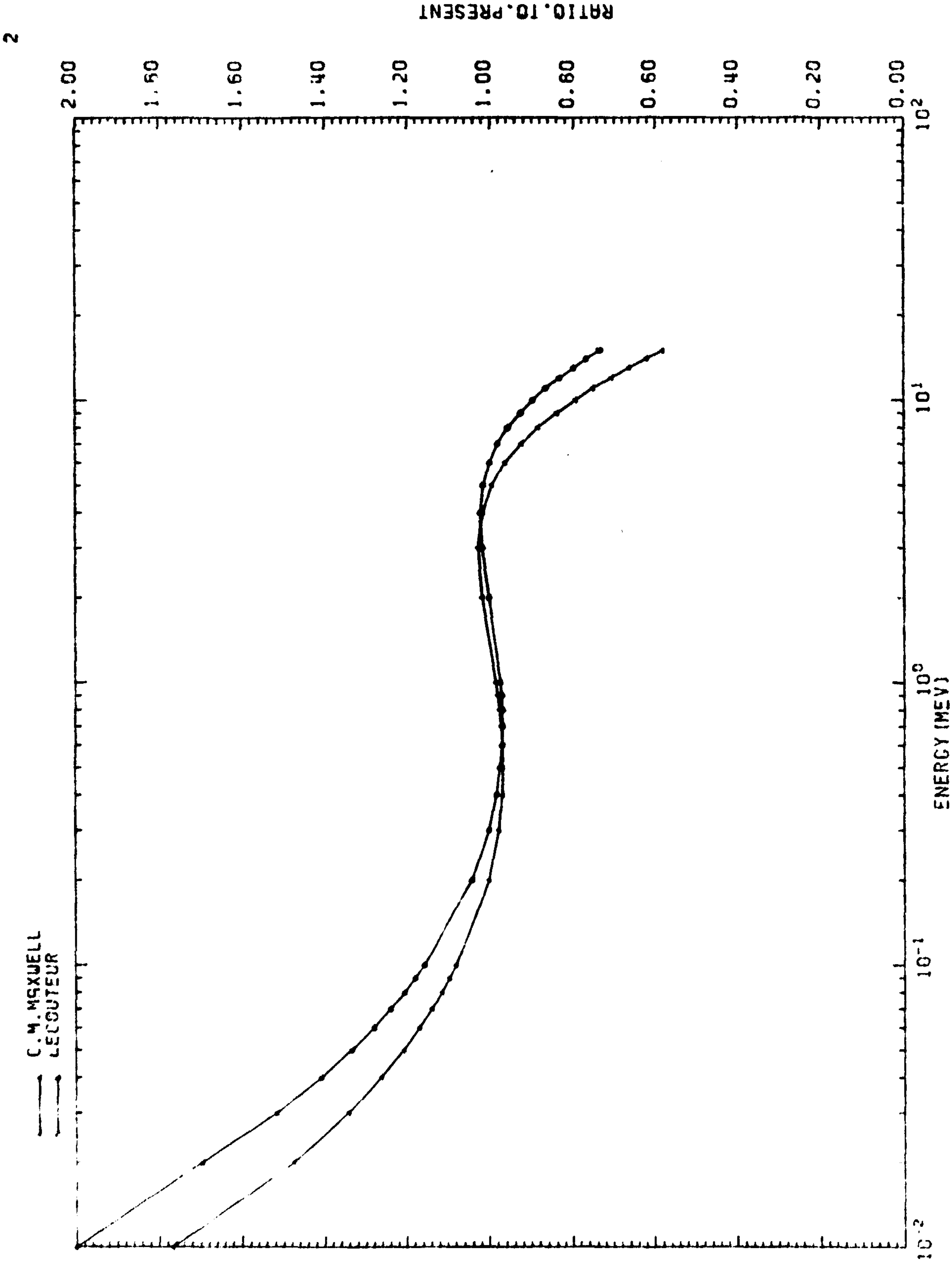

Figure 2 - Ratios of centre of mass neutron spectra to that of Madland/Nix

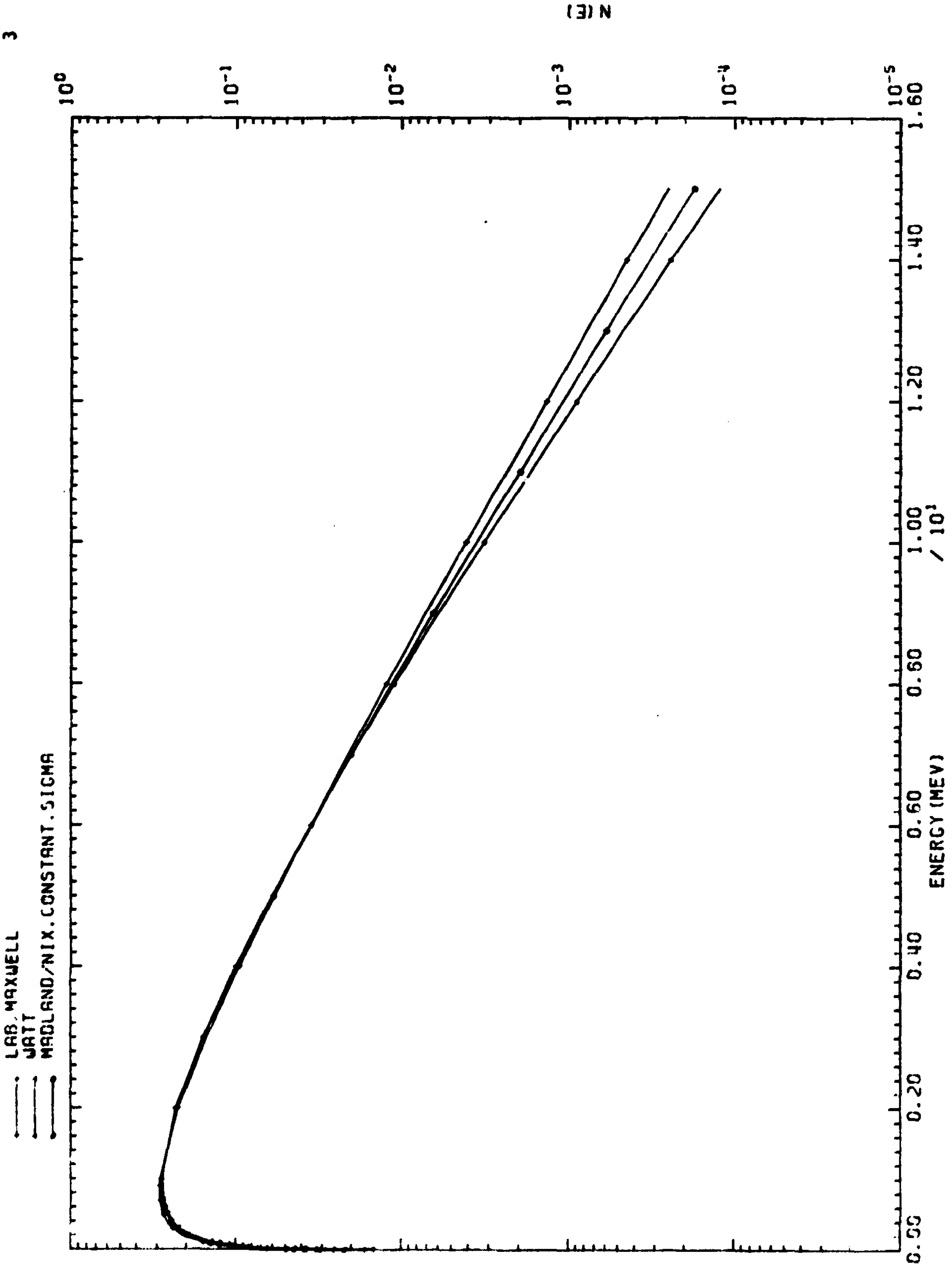

Figure 3 - Laboratory neutron spectra (with constant CN formation cross-sections)

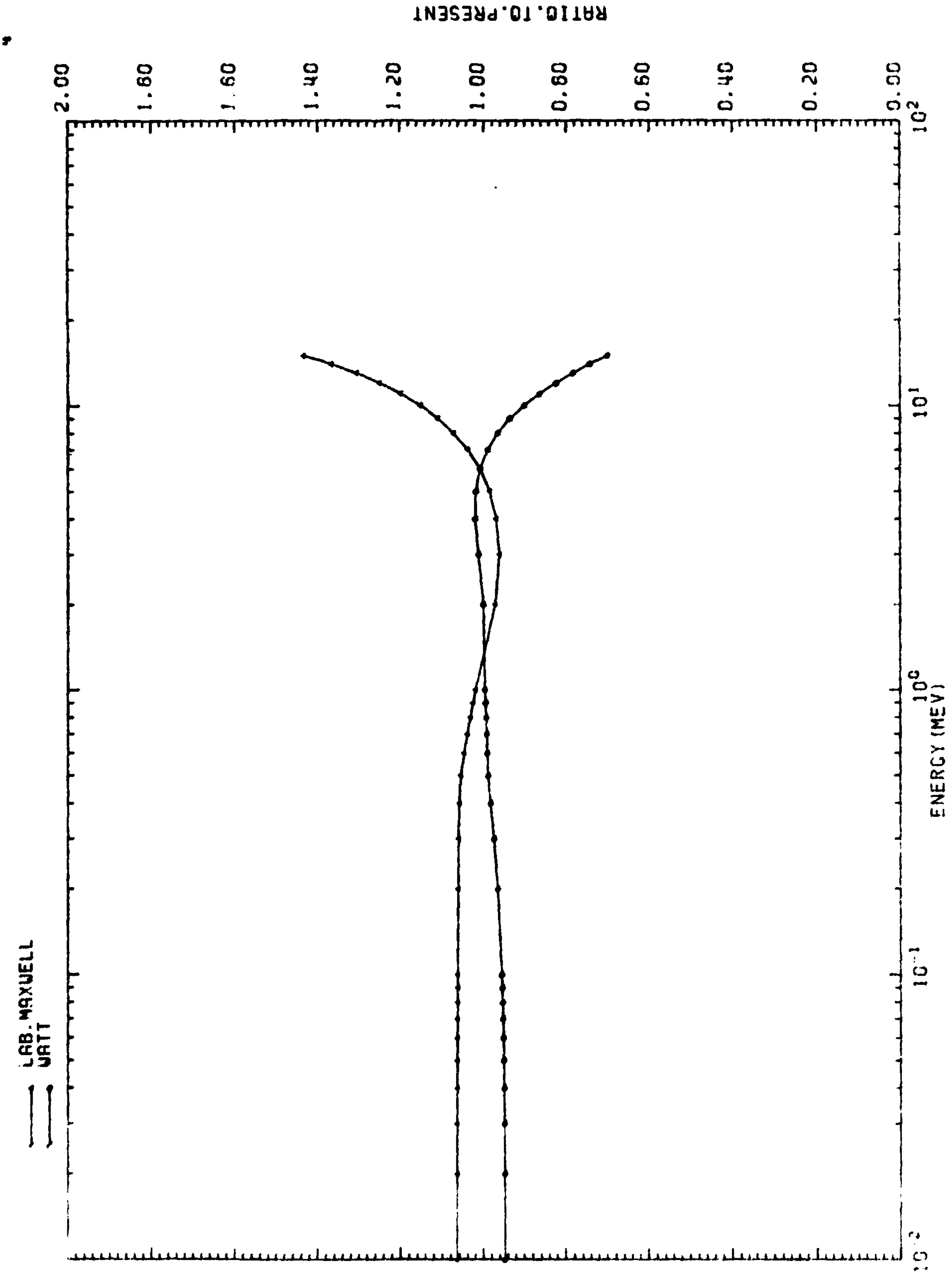

Figure 4 - Ratios of laboratory neutron spectra to that of Madland/Nix

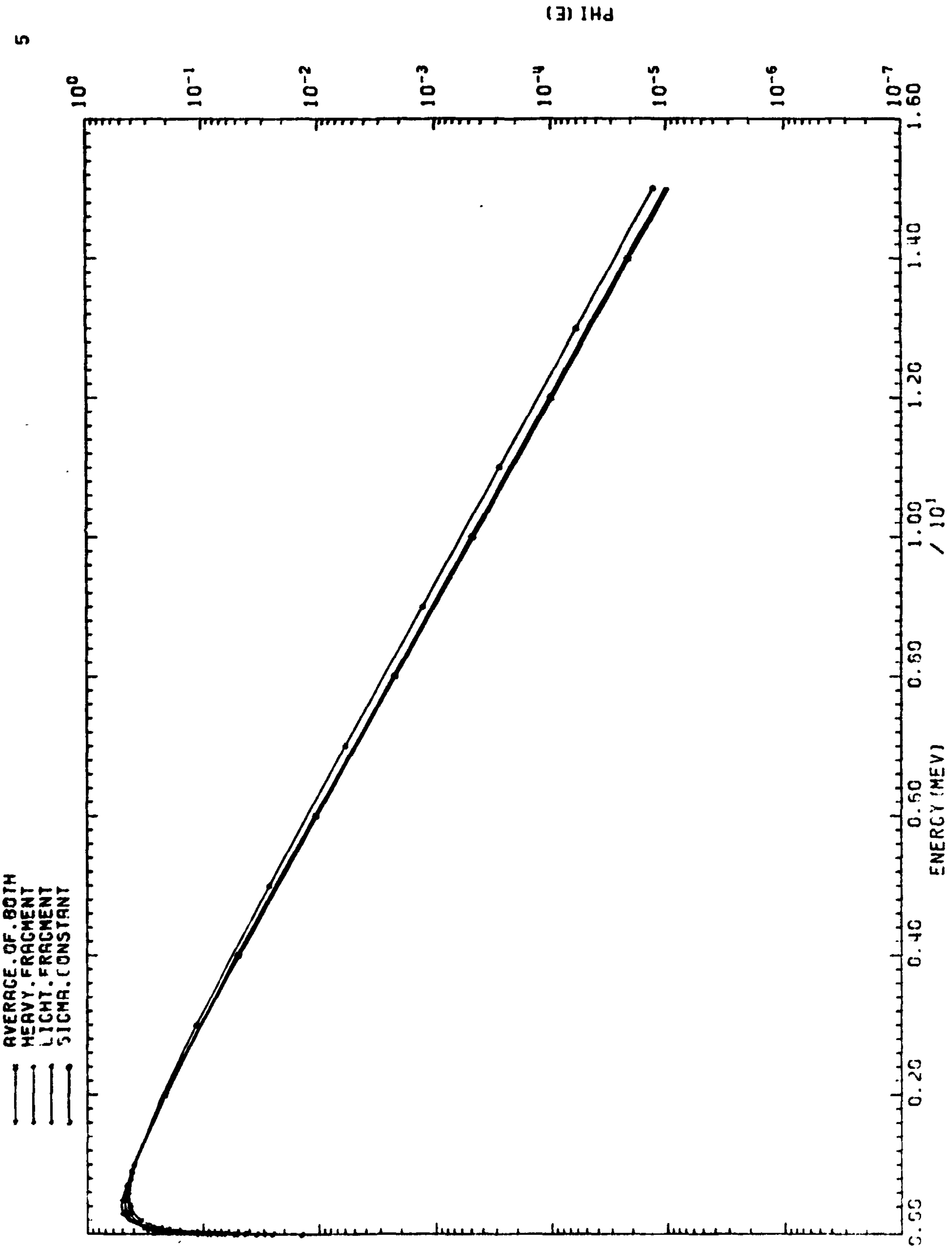

Figure 5 - Centre of mass neutron spectra

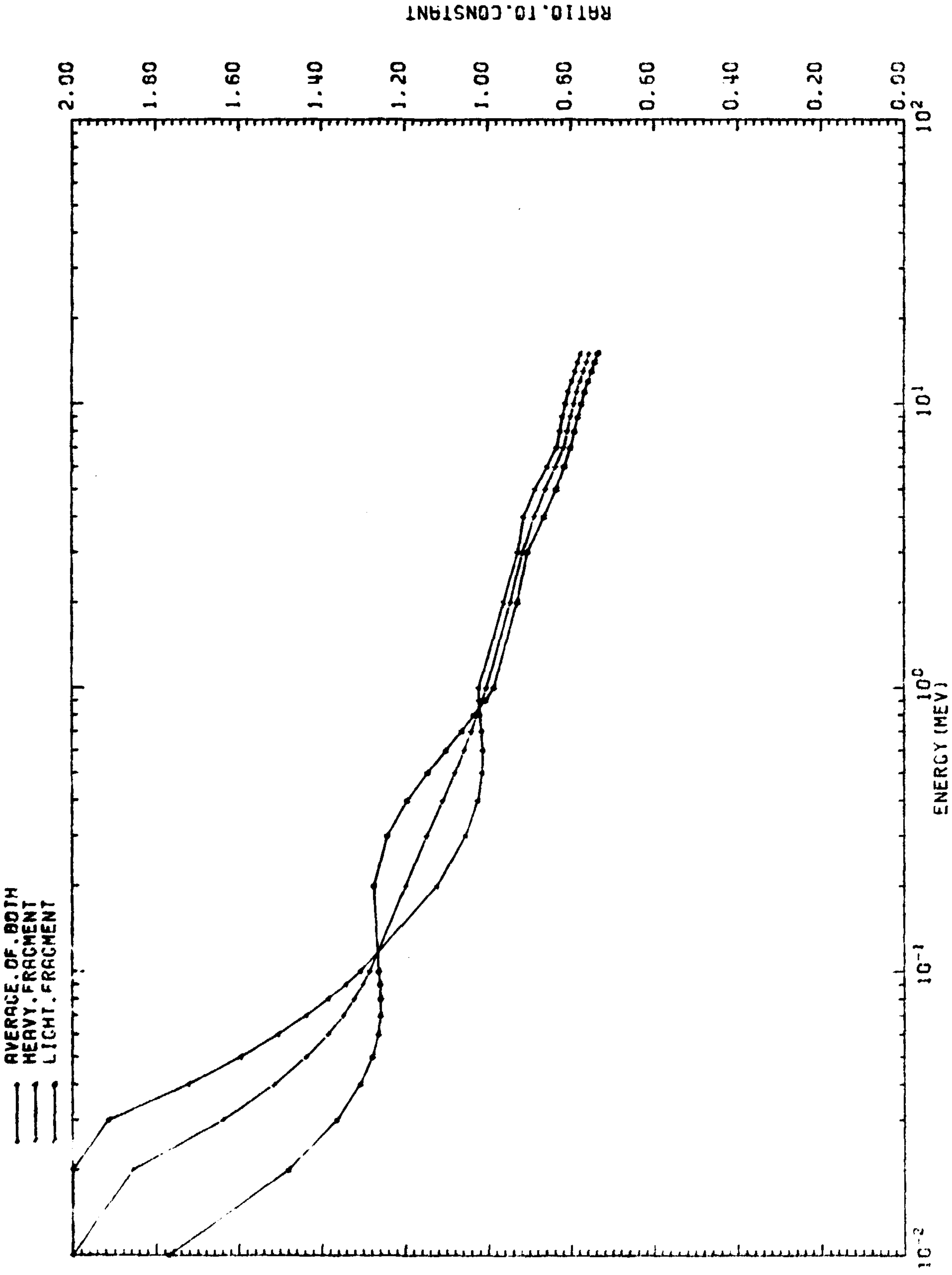

Figure 6 - Ratios of centre of mass neutron spectra to that of Madland/Nix (with constant CN formation cross-sections)

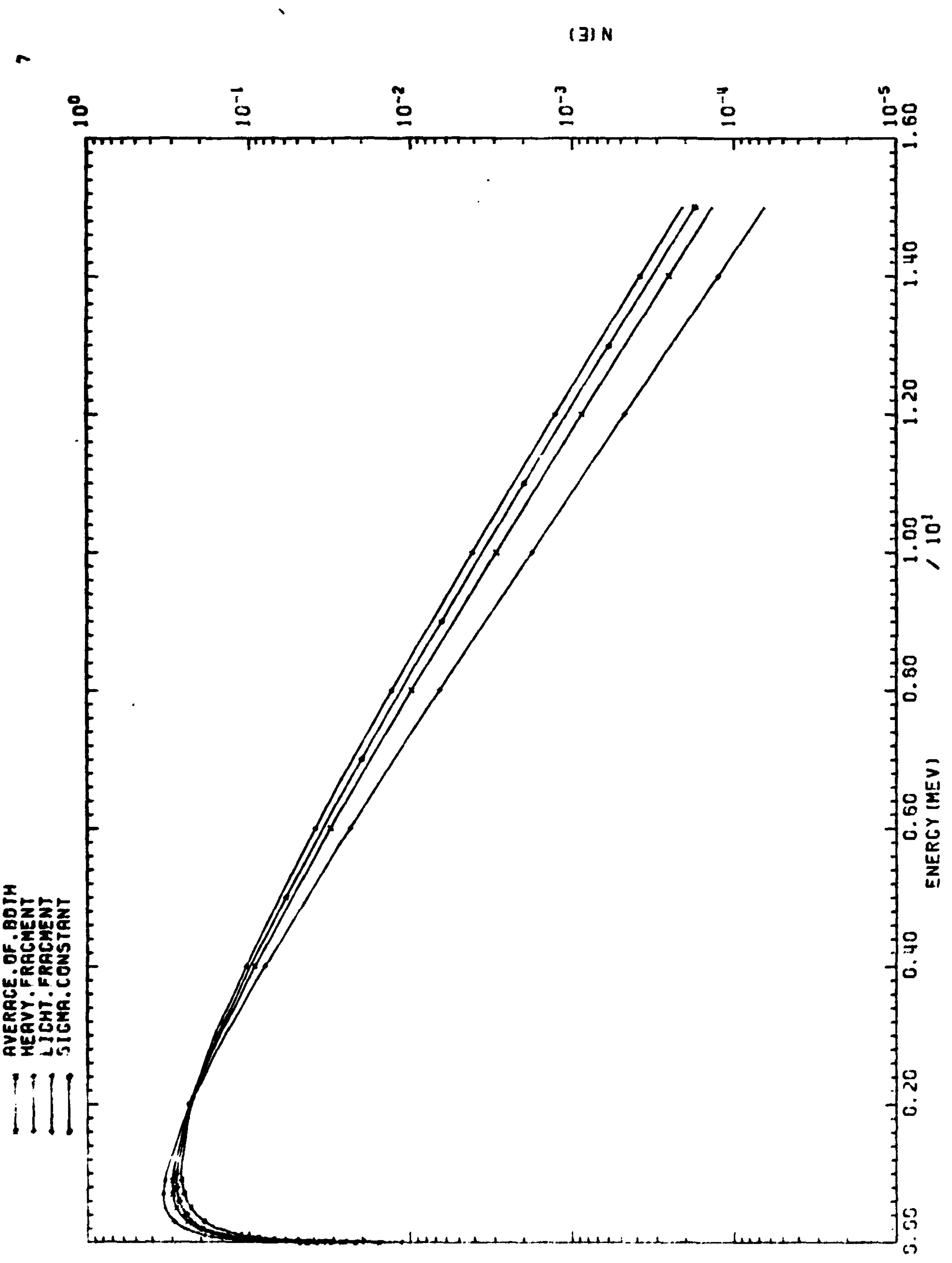

Figure 7 - Laboratory neutron spectra

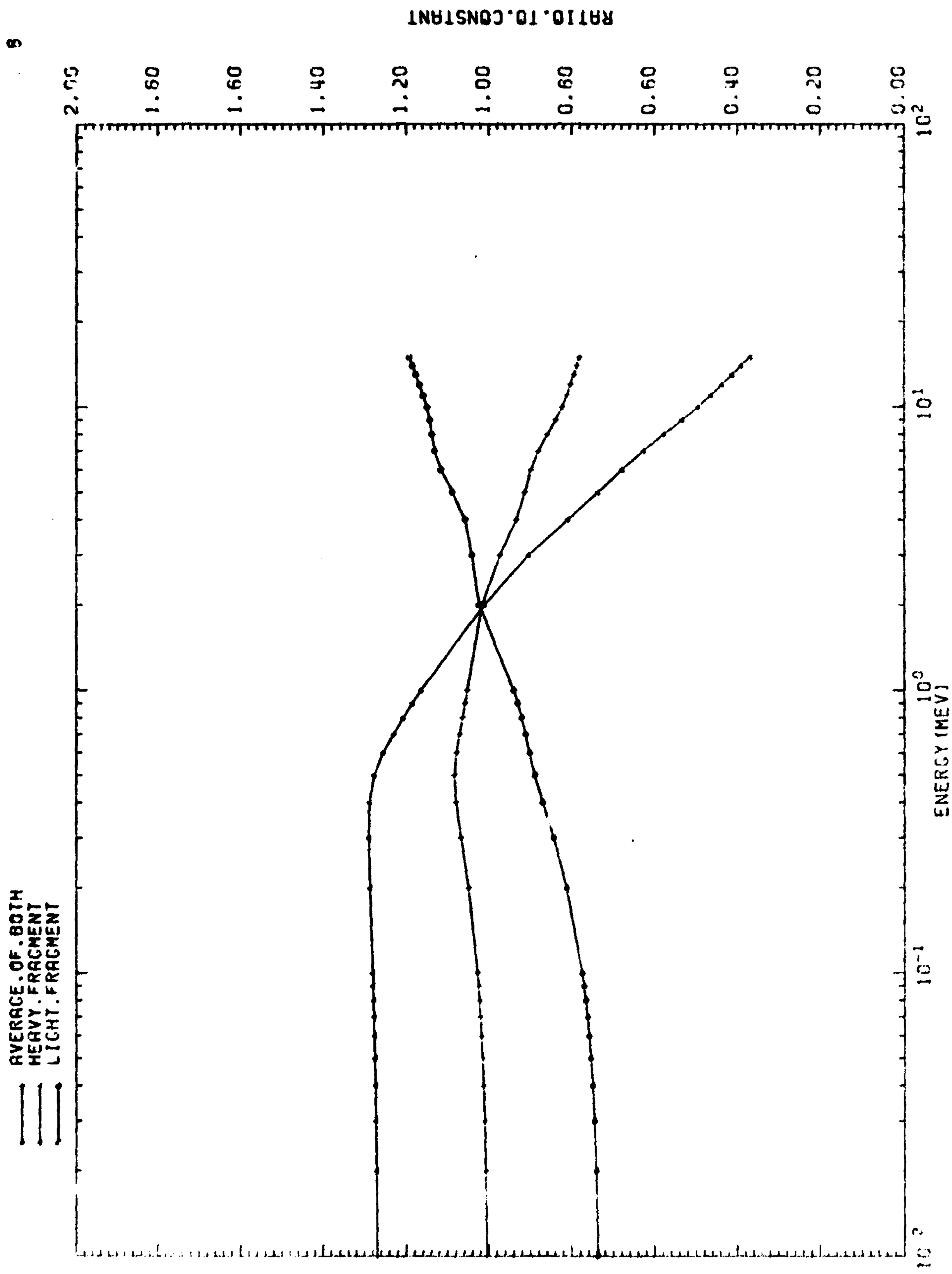

Figure 8 - Ratios of laboratory neutron spectra to that of Madland/Nix (with constant CN formation cross-sections)

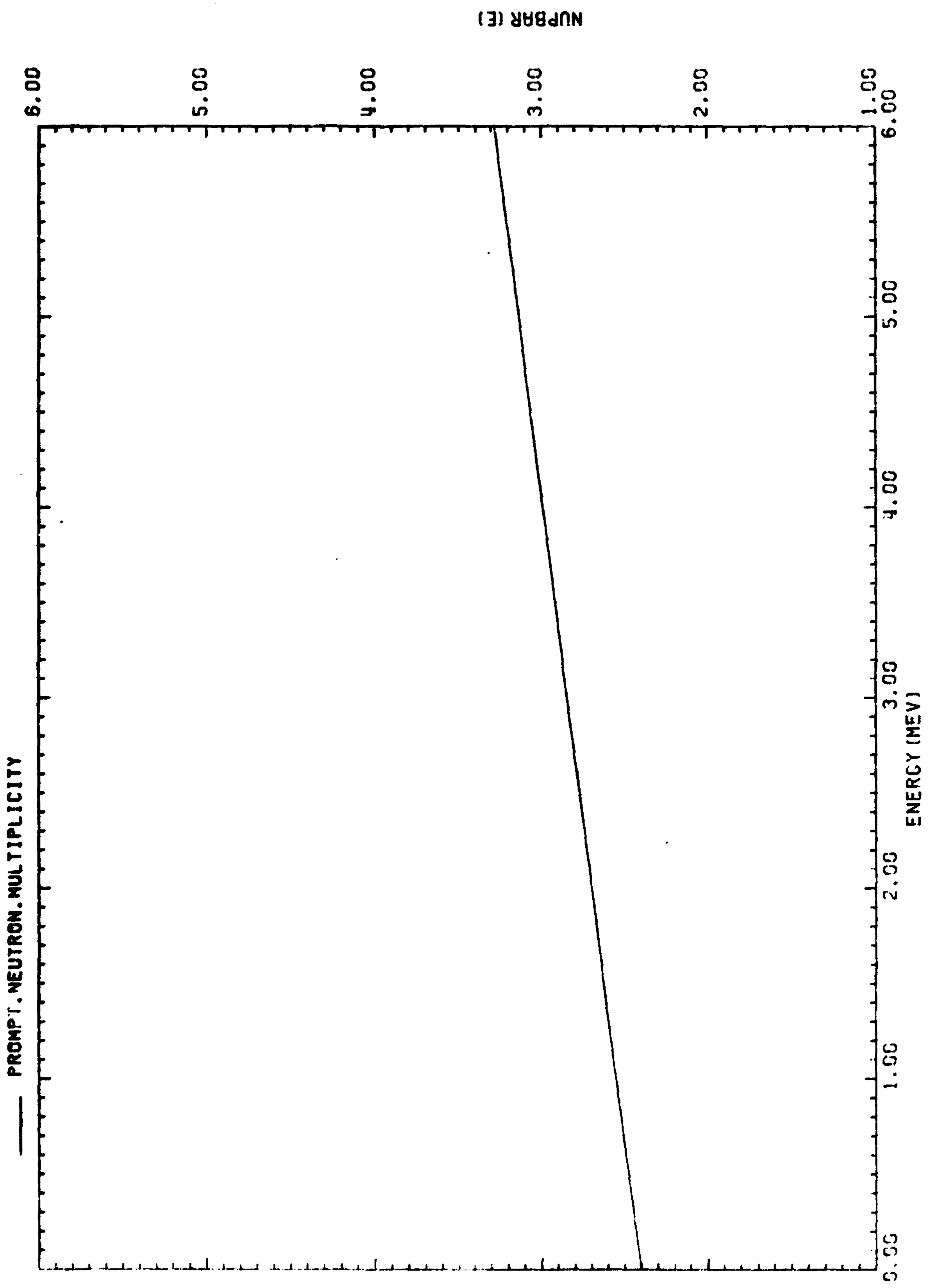

Figure 9 - Prompt neutron multiplicity, $\bar{\nu}_p$, as a function of the neutron inducing fission

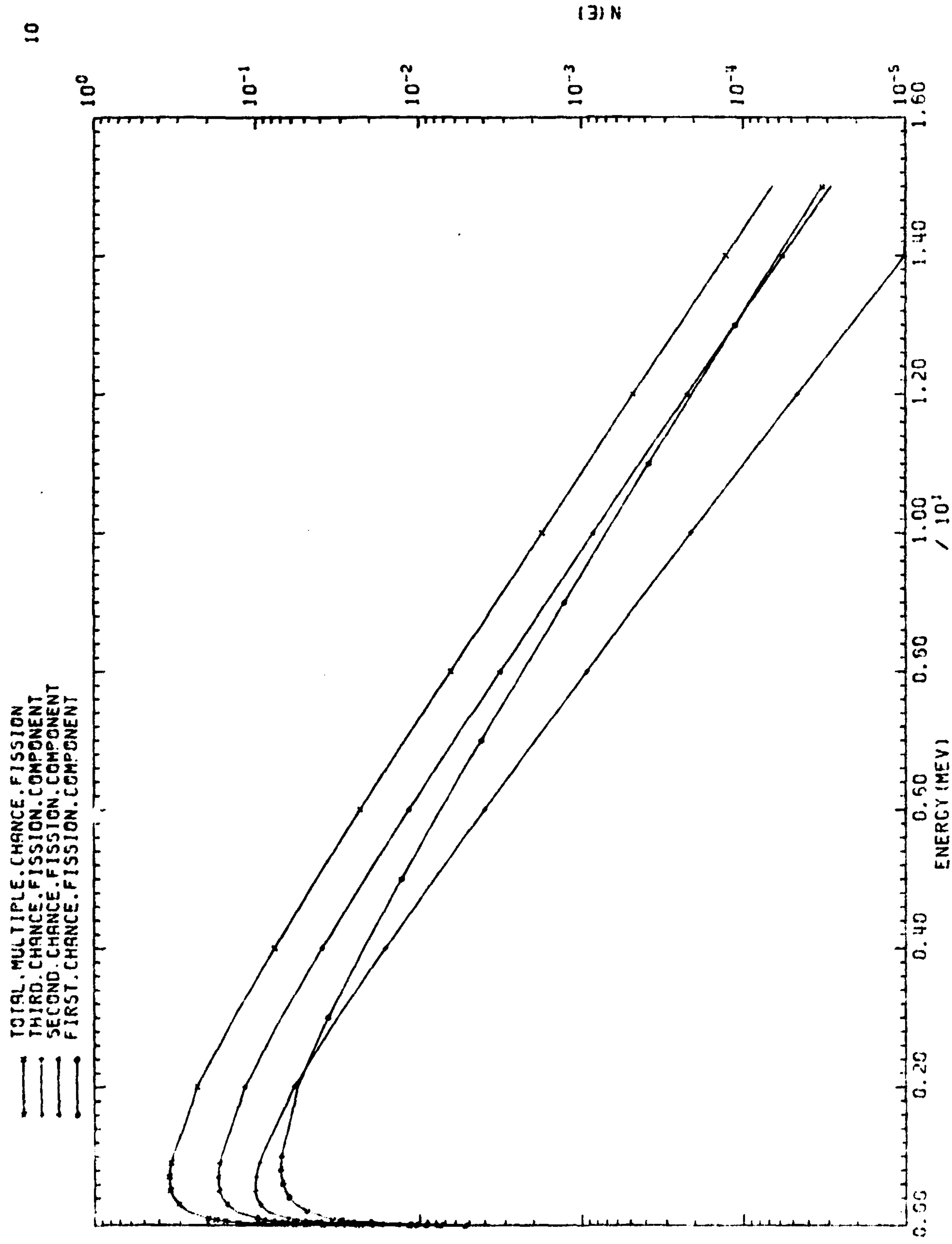

Figure 10 - Multiple chance fission components of the laboratory neutron spectrun

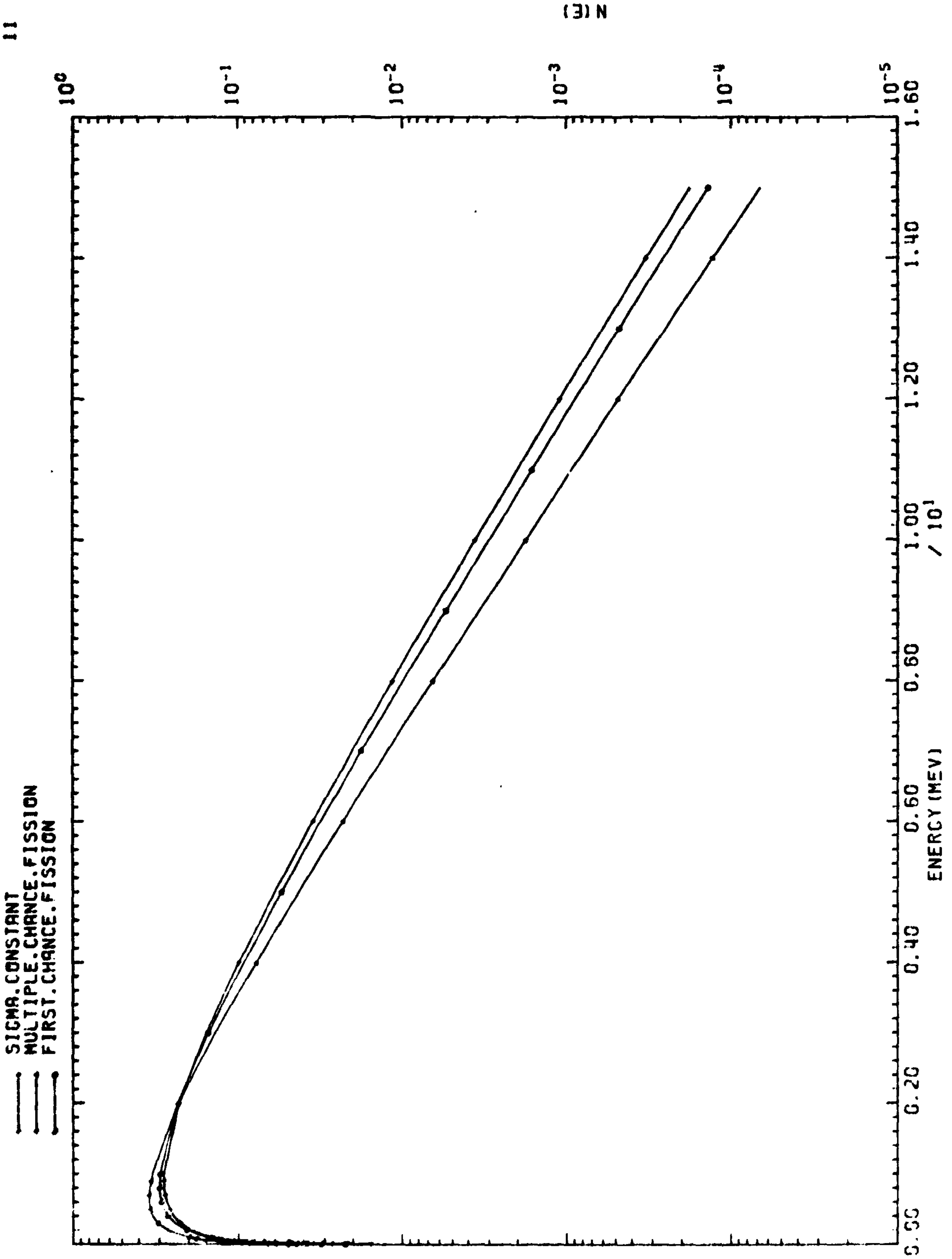

Figure 11 - Laboratory neutron spectra

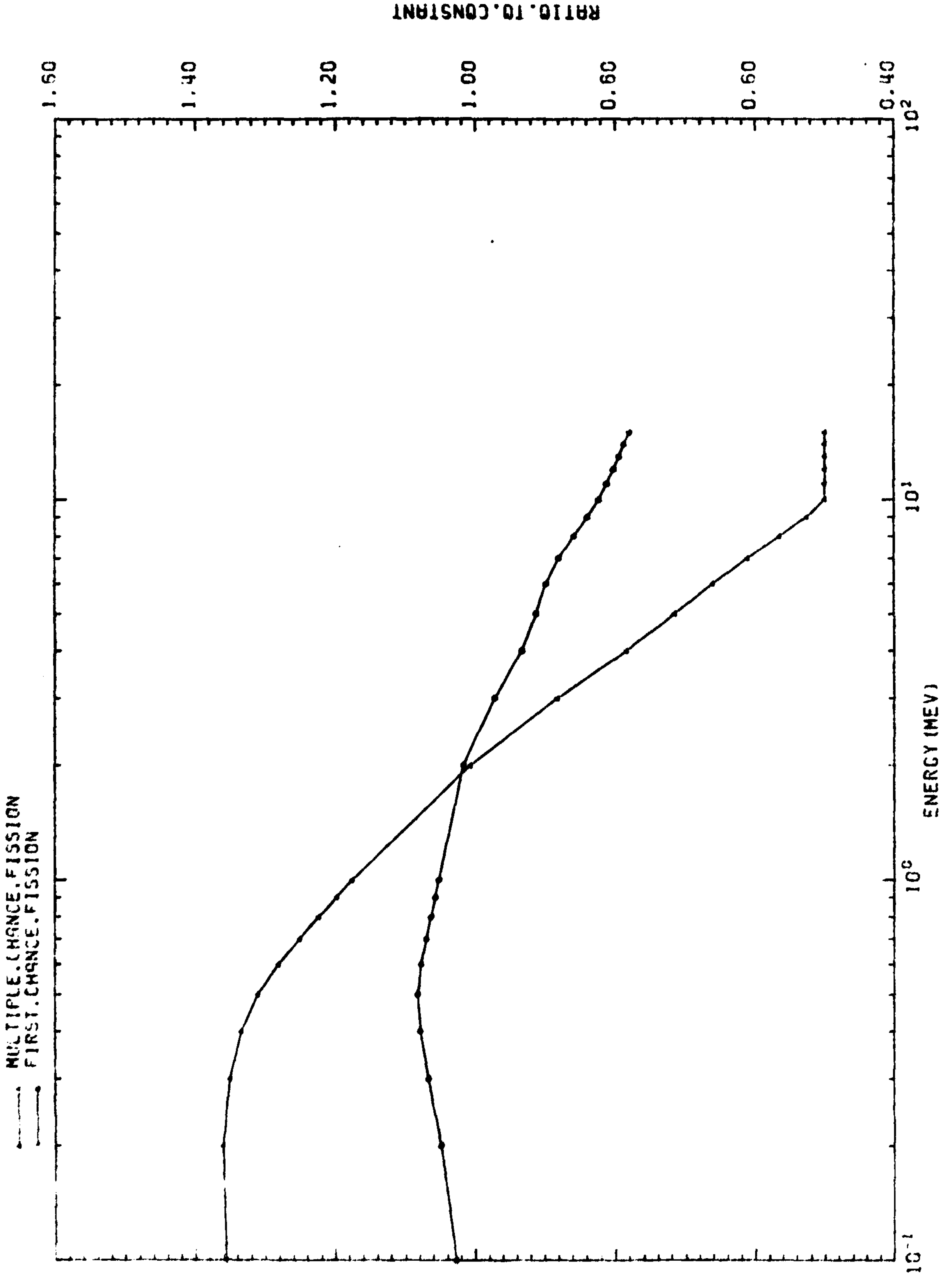

Figure 12 - Ratios of laboratory neutron spectra to that of Madland/Nix (with constant CN formation cross-sections)

6. COMMENTS

6.1 Differences between the various spectra

Four different centre of mass spectra are presented; namely the LeCouteur, Maxwellian and Madland/Nix spectra (all calculated with constant CN formation cross-sections) and the Madland/Nix spectrum employing energy-dependent CN formation cross-sections (generated from the global optical model potential of Becchetti and Greenlees). The LeCouteur and Maxwellian effective temperatures are chosen such that these two spectra have first moments equal to that of the Madland/Nix spectrum. This choice results in all three spectra peaking at very similar energies and being approximately equivalent for neutron energies between about 0.1 MeV and 5 MeV. However, the Madland/Nix calculation predicts substantially more very high and fewer very low energy neutrons (see Figures 1 and 2). When the energy-dependence of the CN formation cross-sections is included in the calculation, the neutron spectrum peaks at a somewhat lower energy, and the first and second moments of the distribution are reduced by about 7% and 11% respectively (see pages 14 and 16). This reflects the observation that more low energy and fewer high energy neutrons are now predicted (see Figure 6).

Five different laboratory neutron spectra are presented; namely the Watt, Maxwellian and Madland/Nix spectra (all calculated using constant CN formation cross-sections) and the Madland/Nix spectra calculated using energy-dependent CN formation cross-sections with and without a consideration of the multiple chance fission components. In fact, the Maxwellian spectrum is not transformed into the laboratory system, maintaining the same functional form that it had in the centre of mass. However, one attempts to compensate for this defect by a different choice of effective temperature. This actually reproduces experimentally observed trends rather well at low neutron energies, but for the wrong physical reason. The Maxwellian and Watt effective temperatures are chosen such that these spectra have the same first moments as the Madland/Nix spectrum (calculated using constant CN formation cross-sections). In addition to peaking at a lower neutron energy than the other two, the Maxwellian spectrum also predicts more neutrons at both very high and very low energies, but fewer at intermediate energies (2-5 MeV), (see Figures 3 and 4). The Watt spectrum is quite similar to the Madland/Nix spectrum for neutron energies up to about 6 MeV, but predicts far fewer high energy neutrons. When the energy dependence of the CN formation cross-sections is included in the calculation of the Madland/Nix spectrum, its peak moves to a lower energy, bringing it into closer agreement with the low energy part of the Maxwellian spectrum. Its first and second moments are reduced by about 5% and 9% respectively (see pages 15 and 17). The inclusion of the various multiple chance fission components moves the peak of the spectrum to an even lower energy (with a corresponding reduction in the first and second moments) so that considerably more low and fewer high energy neutrons are predicted. No oscillatory structure is predicted in any of the spectra calculated in this formalism.

6.2 C.P.U. time considerations

The spectra calculated using constant CN formation cross-sections are dealt with in a few seconds. When energy-dependent cross-sections are included, the time requirement increases substantially, and a majority of it is spent calculating the laboratory neutron spectra, where a three dimensional integral must be performed. The total time spent in the C.P.U. is roughly proportional to the number of energies at which N(E) is required and to one plus the number of neutrons which may be emitted prior to fission. Some typical figures, obtained using a load module compiled with OPT=2, for calculations at the 42 energies between 0.001 MeV and 15 MeV indicated in section 3 (page 10) and used throughout section 4 are;

spontaneous fission of ^{252}Cf	253 s
n(0.53 MeV) + ^{235}U (1st chance fission only)	283 s
n(7.0 MeV) + ^{235}U (1st and 2nd chance fission)	522 s
n(14.0 MeV) + ^{235}U (1st, 2nd and 3rd chance fission)	775 s

REFERENCES

[1] B.E.Watt, Phys. Rev. 87 (1952) 1037.

[2] J.Terrell, Phys. Rev. 113 (1959) 527.

[3] D.G.Madland and J.R.Nix, Nucl. Sci. and Eng. 81 (1982) 213.

[4] V.F.Weisskopf, Phys. Rev. 52 (1937) 295.

[5] B.V.Carlson, private communication. (An extrapolation of the CERN mass defect table has been included in the code)

[6] J.P.Unik, J.E.Gindler, L.E.Glendenin, K.F.Flynn, A.Gorski and R.K.Sjobolm, Proc. 3rd IEAE Symp. physics and chemistry of fission, Rochester, N.Y. (1973) vol.II p.19.

[7] D.C.Hoffmann and M.M.Hoffmann, Ann. Rev. Nucl. Sci. 24 (1974) 151.

[8] F.D.Becchetti Jr. and G.W.Greenlees, Phys. Rev. 182 (1969) 1190.

[9] S.S.Kapoor, R.Ramanna and P.N. Rama Rao, Phys. Rev. 131 (1963) 283.

[10] K.J.Le Couteur, Proc. Phys. Soc. A65 (1952) 718.

[11] J.Terrell, Phys. Rev. 108 (1957) 783.