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Calculations of a Fast Fission Blanket for DT Fusion Reactors with Two Evaluated Data Libraries*

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Abstract

A conceptual fusion-fission hybrid reactor blanket of Werner and Lee has been investigated with two evaluated data libraries, the ENDF/B-III and the Lawrence Livermore Laboratory Evaluated Nuclear Data Library (ENDL). Significant differences are found for tritium breeding, ^{239}Pu breeding, and energy production through fission of uranium in the blanket. These differences are due primarily to differences in the 14-MeV neutron-induced emission spectra as evaluated in the two libraries.

Introduction

In a preceeding paper,[1] one of us examined various neutronic aspects of one type of conceptual fusion-fission hybrid reactor blanket. The results indicated that a fast spectrum blanket has a number of attractive features, the major ones being that unenriched uranium could be efficiently used as a fuel and that tritium breeding from lithium would be sufficient to replace the tritium burned in the D-T fusion reaction. These results were all derived from a detailed neutronics analysis which was based on an evaluated neutron data set.[2]

Because of the favorable potential for this concept as a power producer, it was decided to verify the neutronics analysis using a different evaluated neutron data set.[3] If the results had turned out the same, this paper could have been a footnote in the preceeding work. But the results were quite different. Tritium breeding according to calculations with this second library would be practically inadequate to make up for that burned in fusion. The energy multiplication by fission in the blanket would be 26% lower, and there would be a decrease in the ^{239}Pu breeding. These surprising differences required further investigations and comparisons of the contents of the two evaluated data sets.

Evaluated nuclear data are based, for the most part, on experimental nuclear data when they exist and upon systematics or nuclear model codes when experimental data are inadequate or entirely absent. The two data sets used in these calculations were evaluated to a large extent from the same experimental data and as a result the evaluations are quite similar in many respects. For example, the often measured ^{238}U(n,f) cross section is evaluated almost identically in the two sets. Only rather small differences can be found in most of the evaluated total, fission, and capture cross sections for the materials in the present problem. Similarly, only small differences exist in $\bar{\nu}$ (the average number of neutrons emitted in fission) for ^{235}U and ^{238}U.

The differences which do exist appear in quantities which have not had many experimental investigations. An example is the neutron emission spectrum

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following interaction of 14-MeV neutrons with a nucleus. Except for the recent studies of Kammerdiener,[4] few measurements had been made of this quantity and evaluations of it relied on neutron evaporation models or systematics. Yet, these emission spectra appear to be especially relevant for the fusion-fission hybrid and in fact account for most of the differences in the calculated parameters.

Calculation

A conceptual hybrid reactor model of Werner[5] and Lee[1] was investigated by means of Monte Carlo calculations. To illuminate various aspects, the spherical system was divided into 19 zones as indicated in Fig. 1. Zones 2, 4, 6, 8, 10, 12, 14, and 18 are only 0.001 cm thick each and were included to investigate the neutron spectra. The materials and densities are listed in Table I. Where more than one element is present in a zone, an homogenized mixture is assumed.

Two data libraries were used. The ENDF/B-III library[2] and the Lawrence Livermore Laboratory (LLL) Evaluated Nuclear Data Library (ENDL)[3] were processed into 176 group-averaged data by the LLL processing code CMAN.[6] A flat flux weighting spectrum was used for each group.

Calculations of the hybrid systems were performed with the code TART[7] which does an analogue Monte Carlo calculation. The source of 14-MeV neutrons was assumed to be uniformly distributed within the 320 cm radius. Samples of 10,000 fusion neutrons were studied for each calculation. The code computed the number and spectrum of neutrons entering designated zones, the number of each type of reaction within a zone, and the probability that the reaction was initiated by a neutron of a particular energy.

Results

The calculated results are given in Table II. For tritium breeding, calculation with the ENDF/B-III yields tritium breeding which is on the order of 20% less than given by the calculation which used the ENDL. The former result would imply, assuming 100% recovery, a bare equality between tritium burned and that produced. On the other hand, the latter result is a comfortably adequate breeding factor. (It should be noted that in this model, unlike the so-called "standard CTR blanket" design,[8] the ^{7}Li(n,n't) reaction contributes little to tritium breeding.)

A greater disparity occurs in the number of fissions where the ENDF/B-III result is 26% lower than the calculation with ENDL. This result varies only slightly for the different zones which contain uranium. Most of the fissions in this design occur in the abundant ^{238}U isotope which has a threshold for fission of about 1.5 MeV. It is in this isotope where the largest difference occurs in the two calculations.

^{239}Pu is bred by the usual sequence of reactions ^{238}U(n,γ)^{239}U → ^{239}Np → ^{239}Pu. Thus, the number of neutron captures on ^{238}U gives the plutonium breeding factor. For this quantity, the two calculations give results differing only by 7% with the ENDL result again being the larger.

Figures 2-6 give comparisons of the calculation for the quantities of interest as a function of the neutron energy which initiates the various reactions. Representative zones have been chosen for illustration. These plotted results are proportional to the particular reaction cross section weighted by the neutron flux spectrum in the zone. Figures 7 and 8 give the spectra of neutrons entering zones 6 and 10. Except for a factor of $1/4\pi R^2$, these latter two spectra are equal to the flux spectrum at these two narrow zones, one being midway through the fissile material and the other being 10 cm into the predominantly lithium-filled zone. In all these figures, the ENDF/B-III results and the ENDL results differ in magnitude and, for the low numbered zones, in shape.

Discussion

The rather large differences in the results calculated with the ENDF/B-III and the ENDL appear to be due to differences in the neutron emission spectra induced by 14-MeV neutrons. After the first interaction of 14-MeV neutrons in unenriched uranium, the emission spectrum is that of Fig. 9. One notes that neutrons above 2 MeV are more probable according to the ENDL than according to the ENDF/B-III. A very similar result holds for the 14-MeV induced emission spectrum from the sum of the structural materials, niobium and stainless steel. (The emission spectrum of iron has been discussed at length by L. F. Hansen et al.[9]). The differences in the evaluations for lithium at 14 MeV are in the same direction but much smaller.

These spectral differences work on the ^{238}U fission probability. Recall that the ^{238}U fission cross section has a threshold at 1.5 MeV above which it rises to about 0.55 barns. Near 6.3 MeV, the threshold for (n,n'f) is reached and the total fission cross section rises to about 1 barn. Near 13.8 MeV, the cross section rises again to about 1.4 barns. Thus, the more energetic the neutron emission spectrum (i.e. that in ENDL) the more likely the secondary neutrons will induce fission. The total number of ^{238}U fissions is thus greater in the calculation with the ENDL data. Similar considerations hold for the fission of ^{235}U.

For the reactions $^{6}Li(n,t)$ and $^{238}U(n,\gamma)$, the cross sections in the MeV range decrease with increasing neutron energy. Thus, the number of these reactions is determined by the number of neutrons below the fusion source peak. Because there are more fissions per fusion neutron in the calculation with ENDL, there are more neutrons in the system per fusion. The calculation with ENDL therefore gives more tritium and ^{239}Pu breeding. We note that reactions below 1 MeV are not very important because the absorption by lithium depletes the flux at these low energies.

In conclusion we have found that the 14-MeV neutron-induced emission spectra data are crucial to the calculation of a fusion-fission hybrid reactor. Relevant data[4] for many materials have recently become available and could be included in evaluations. Finally, we note that the emission spectra find especially good tests in pulsed sphere experiments where the spheres are no larger than a few mean paths in radii. Of the materials under consideration here, spheres of iron[9,10] and isotopes of uranium and lithium[10] have been investigated at 14 MeV. The emission spectra above 2 MeV, for the lithium spheres are calculated[11] quite well with either the ENDF/B-III or the ENDL. The spheres of iron[9] or of the uranium isotopes[12] on the other hand are calculated much better with ENDL than with ENDF/B-III for emission energies above 2 MeV.

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[8]D. Steiner, "Analysis of a Benchmark Calculation of Tritium Breeding in a Fusion Reactor Blanket: The United States Contribution," ORNL-TM-4177 (1973).

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[10]C. Wong, J.D. Anderson, P.S. Brown, L.F. Hansen, J.L. Kammerdiener, C.M. Logan, and B. Pohl, "Livermore Pulsed Sphere Program: Program Summary through July 1971," UCRL-51144 (1971).

[11]R.J. Howerton, private communication.

[12]R.J. Howerton and M.H. MacGregor, "Evaluated Neutron Reaction Data for Uranium-238," UCRL-51427 (1973).

Table I - Material Composition

Zones	Density (gm/cm^3)	Material	Atom Fraction
1	10^{-7}	D + T	(a)
2-7	13.3	^{238}U	0.710
		^{235}U	0.005
		^{7}Li	0.127
		^{6}Li	0.010
		Fe	0.073
		Ni	0.017
		Nb	0.060
8-15, 17, 18	1.07	^{7}Li	0.780
		^{6}Li	0.062
		Fe	0.083
		Ni	0.020
		Nb	0.055
16	2.0	C	1.0
19	5.82	H	0.475
		B	0.003
		C	0.243
		Fe	0.280

(a) Since the ENDF/B-III library does not contain tritium as a material, pure deuterium was assumed in Zone 1.

Table II - Reactions by Zone per Fusion

Reaction	Zone	ENDF	ENDL	RATIO (ENDF/ENDL)
		Tritium Breeding		
$^{6}Li(n,t)$	3	0.044	0.053	0.83
	5	0.039	0.047	0.83
	7	0.051	0.064	0.80
	9	0.159	0.205	0.78
	11	0.132	0.172	0.77
	13	0.219	0.273	0.80
	15	0.216	0.254	0.85
	17	0.079	0.093	0.85
	Total	0.94	1.16	0.81
$^{7}Li(n,n't)$	Total	0.06	0.07	0.8
Total Tritium Breeding		1.00	1.23	0.81
		^{239}Pu Breeding		
$^{238}U(n,\gamma)$	3	0.675	0.705	0.96
	5	0.603	0.628	0.96
	7	0.847	0.944	0.90
	Total	2.13	2.28	0.93
		Fission		
^{238}U fission	3	0.278	0.372	0.75
	5	0.171	0.239	0.72
	7	0.153	0.228	0.67
	Total	0.60	0.84	0.72
^{235}U fission	3	0.039	0.044	0.88
	5	0.034	0.038	0.89
	7	0.044	0.052	0.85
	Total	0.12	0.13	0.87
Total fission		0.72	0.97	0.74

Figure Captions

1. Zoning of the fast fission blanket.
2. $^{6}Li(n,t)$ reactions versus neutron energy - zone 5.
3. $^{6}Li(n,t)$ reactions versus neutron energy - zone 11.
4. ^{238}U fissions versus neutron energy - zone 5.
5. ^{235}U fissions versus neutron energy - zone 5.
6. ^{238}U captures versus neutron energy - zone 5.
7. Spectrum of neutrons entering zone 6.
8. Spectrum of neutrons entering zone 10.
9. Spectrum of neutrons after one interaction of a 14-MeV neutron with natural uranium.

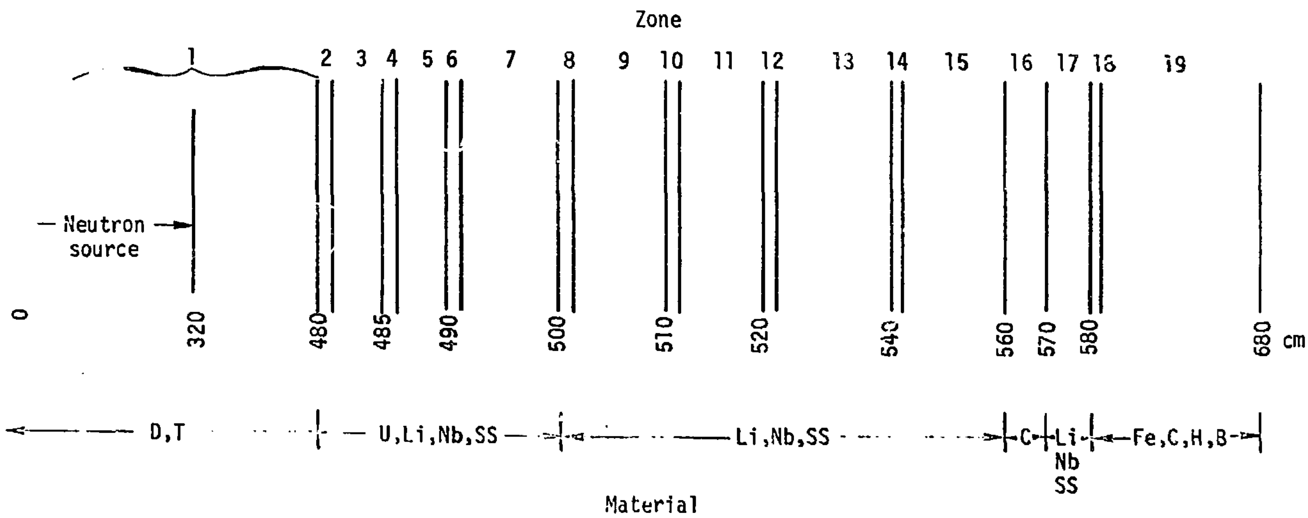

Zone
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
Neutron source
0
320
480
485
490
500
510
520
540
560
570
580
680
cm
D,T
U,Li,Nb,SS
Li,Nb,SS
C
Li Nb SS
Fe,C,H,B
Material

Fig. 1

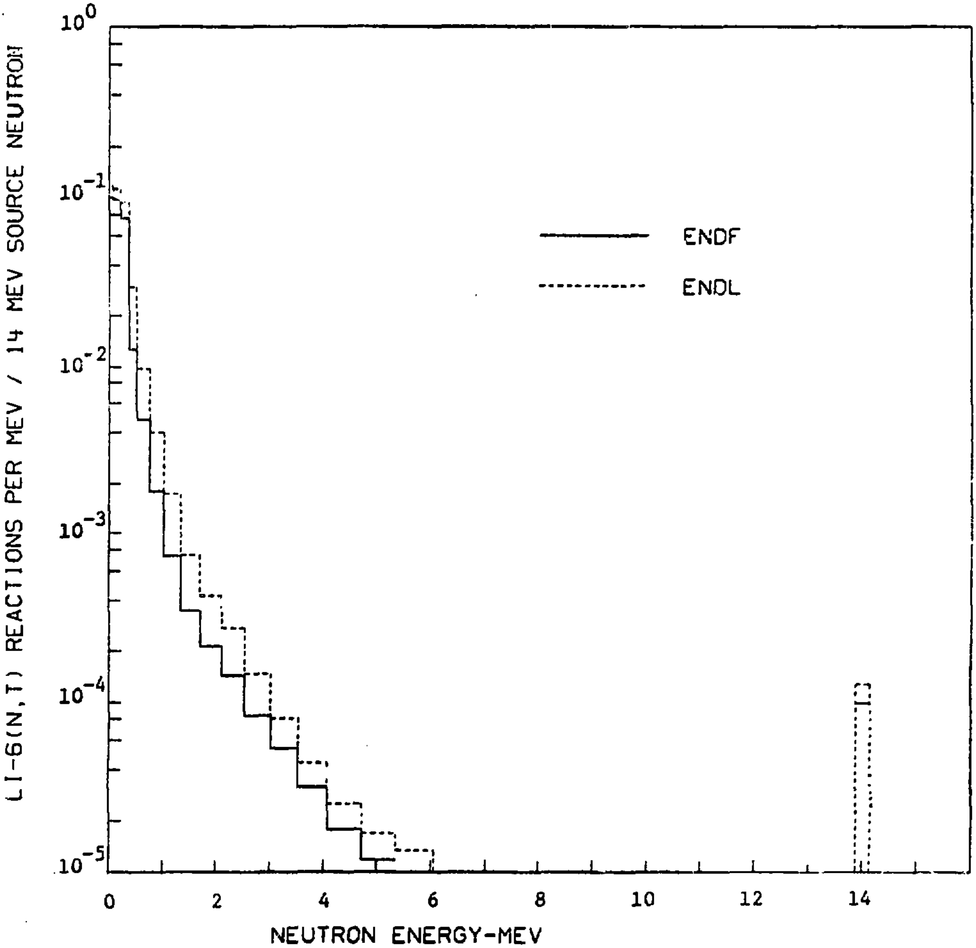
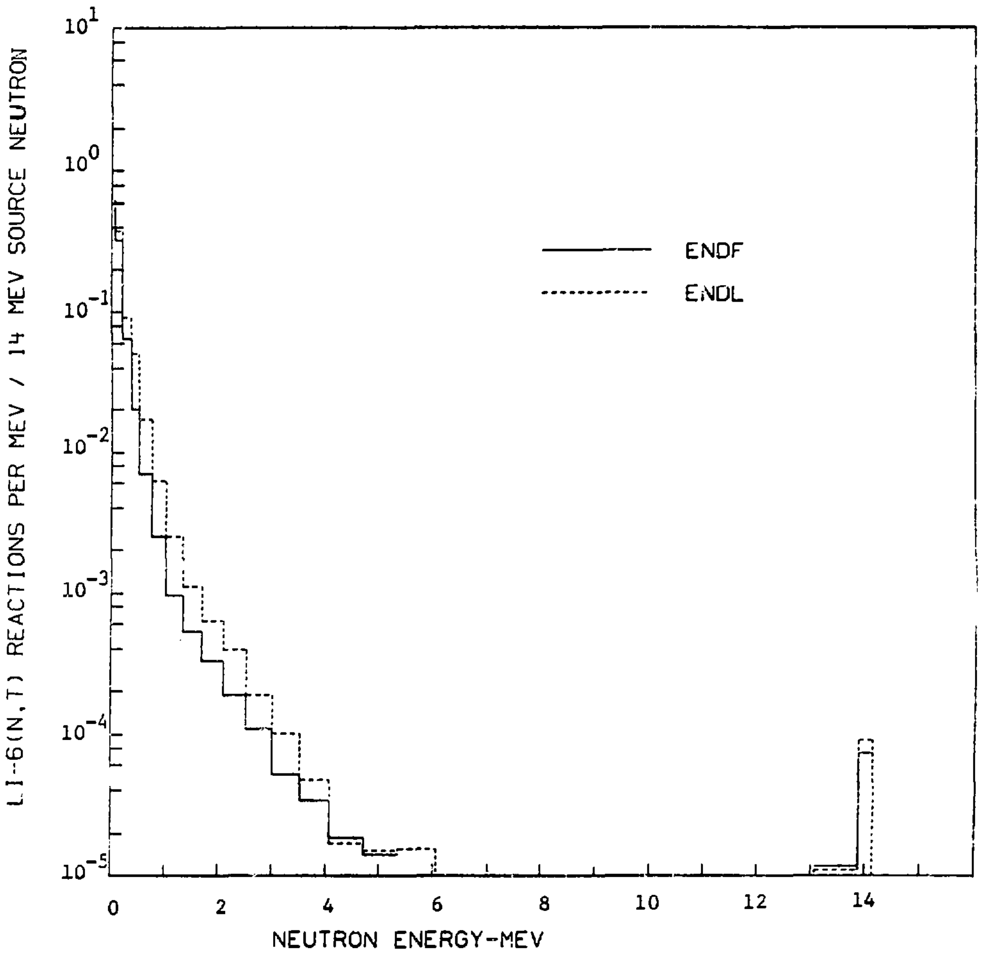
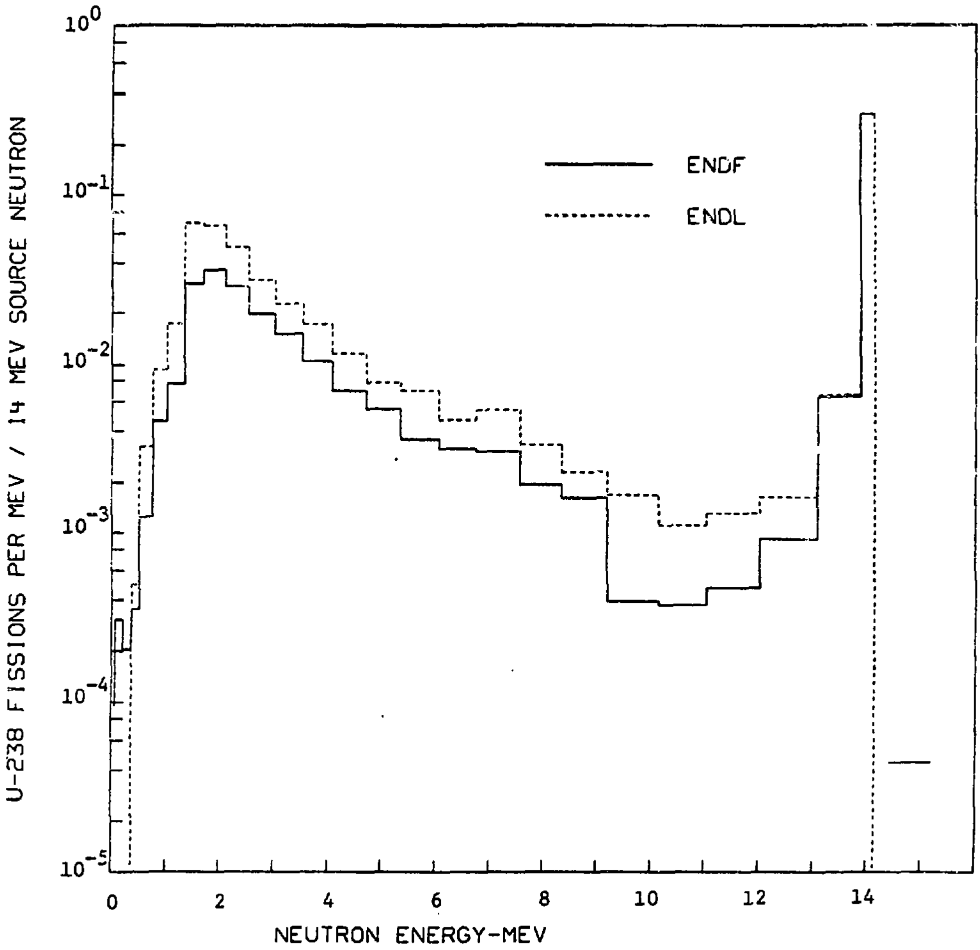

LI-6(N,T) REACTIONS PER MEV / 14 MEV SOURCE NEUTRON
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV
ENDF
ENDL

Fig. 2



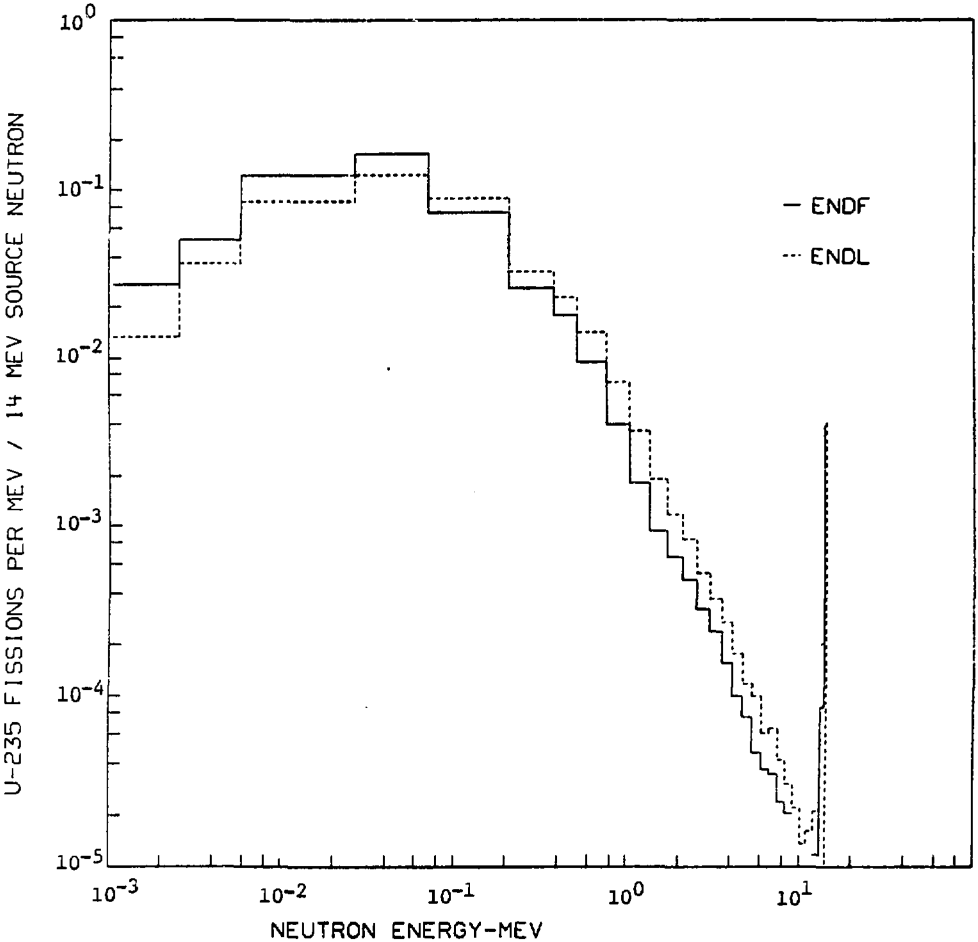
LI-6(N,T) REACTIONS PER MEV / 14 MEV SOURCE NEUTRON
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
ENDF
ENDL
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV

Fig. 3



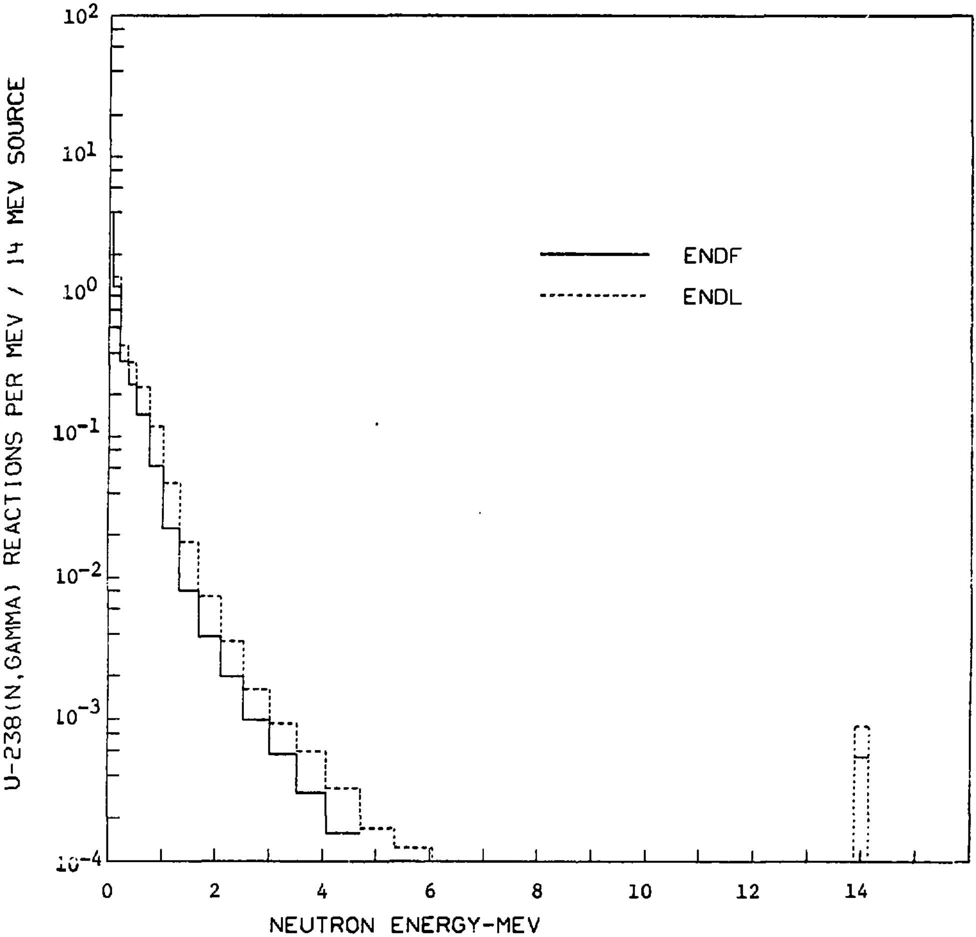
ENDF
ENDL
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
U-238 FISSIONS PER MEV / 14 MEV SOURCE NEUTRON
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV

Fig. 4



U-235 FISSIONS PER MEV / 14 MEV SOURCE NEUTRON
NEUTRON ENERGY-MEV
ENDF
ENDL
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-3}
10^{-2}
10^{-1}
10^0
10^1

Fig. 5



10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
U-238(N,GAMMA) REACTIONS PER MEV / 14 MEV SOURCE
ENDF
ENDL
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV

Fig. 6

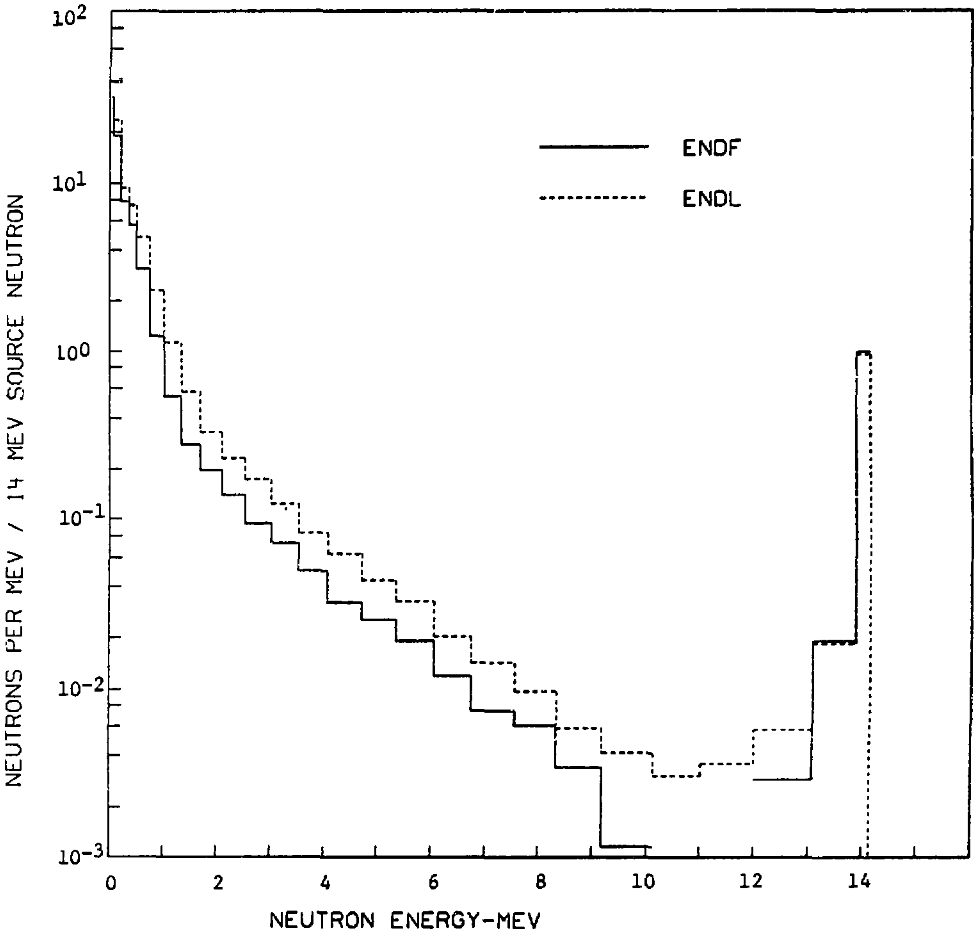
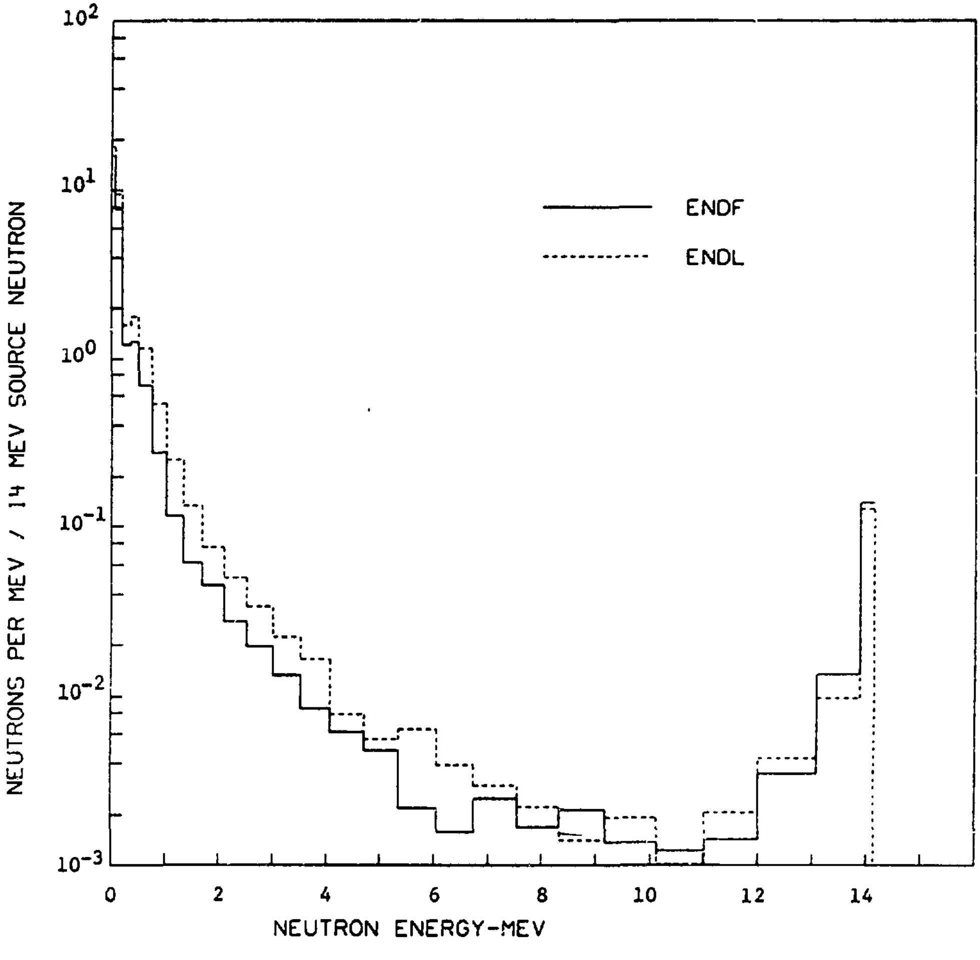

ENDF
ENDL
10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
NEUTRONS PER MEV / 14 MEV SOURCE NEUTRON
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV

Fig. 7



10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
ENDF
ENDL
NEUTRONS PER MEV / 14 MEV SOURCE NEUTRON
0
2
4
6
8
10
12
14
NEUTRON ENERGY-MEV

Fig. 8

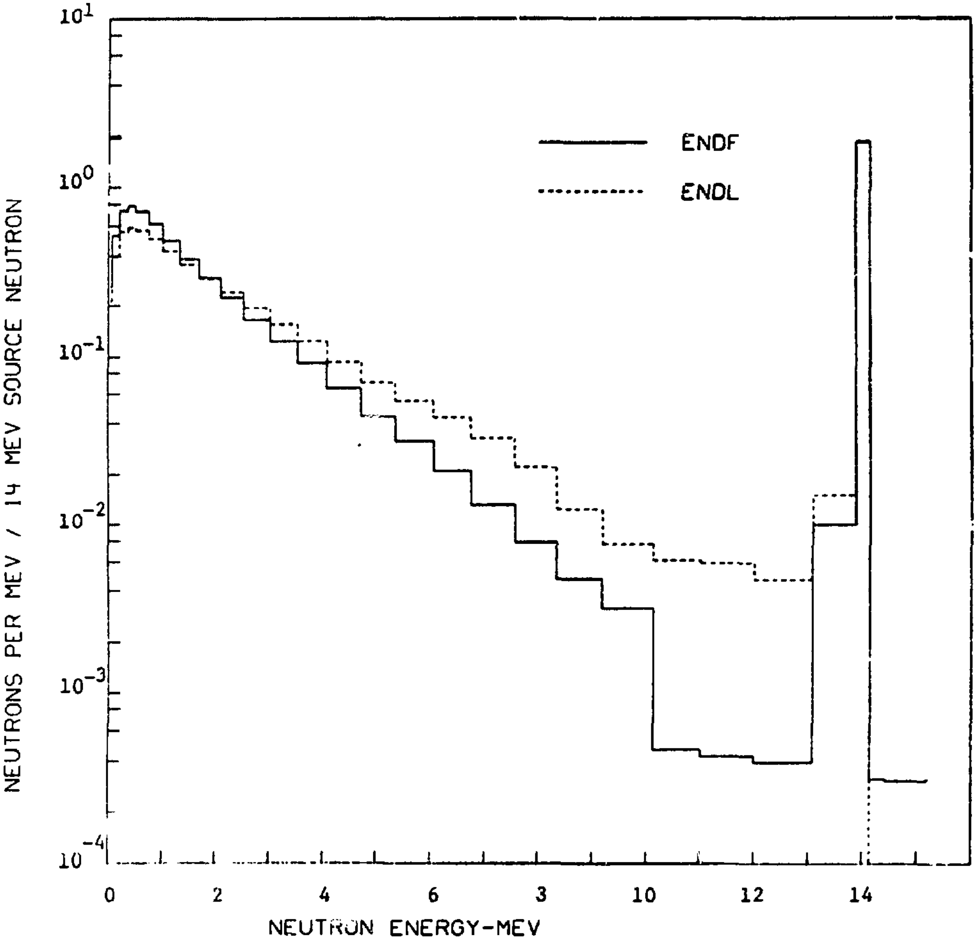

ENDF
ENDL
NEUTRONS PER MEV / 14 MEV SOURCE NEUTRON
10^1
10^0
10^-1
10^-2
10^-3
10^-4
0
2
4
6
3
10
12
14
NEUTRON ENERGY-MEV

Fig. 9