Parametric Calculations of ^{239}Pu Breeding in Hybrid Blankets With and Without Tritium Breeding

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FUSION TECHNOLOGY INSTITUTE

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Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

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I. Introduction

In this report we discuss the performance of fusion-fission hybrid blankets that produce ^{239}Pu with and without the breeding of tritium. These calculations are compared to others that are found in the literature. However, the major purpose of our calculations is to provide a self-consistent set of data to evaluate the performance of blankets when they breed tritium and when they do not. The results of these calculations are then used in an economic model to evaluate the entire system of hybrid, fission reactors, and external tritium producing reactors and to compare this system with the more conventional scheme where all of the tritium is bred in the hybrid. These results are presented elsewhere.

For each blanket configuration the following quantities are computed: (1) total breeding, (2) ^{239}Pu net breeding, (3) tritium breeding, (4) leakage, (5) blanket multiplication, (6) k_{eff}, (7) all of the relevant reaction rates. All reaction rates are computed per source neutron. Blanket configurations that give a tritium breeding ratio of one for initial ^{239}Pu concentrations of 0, 2, 3 and 4% are determined. The normalized power density in the ^{239}Pu breeding zone of the blanket is given for the case of tritium breeding and no tritium breeding. From these normalized power densities the maximum fusion source power can be determined. A few calculations have also been done to compare metal fuel with oxide and carbide fuel.

In addition to providing information for economic analyses, these simple blanket designs also give some basic guidelines regarding the performance of Pu producing hybrid blankets. Such guidelines are useful when making initial estimates for more detailed blanket designs.

II. Calculational Model

The neutronic calculations were carried out in spherical geometry using the one-dimensional discrete ordinate neutron transport code ANISN. A 25 neutron group cross section library has been used which was created from the RSIC DLC-41B/VITAMIN-C AMPX master interface library based upon ENDF/B-IV data. The group boundaries for the 25 neutron groups are given in Table 1.

Energy multiplication was calculated from the reaction rates obtained from ANISN and the reaction Q values for the particular reaction of interest. Fig. 1 shows the simple blanket configuration used for these ANISN calculations. The fuel pins were taken to be LMFBR fuel pins.

III. Results

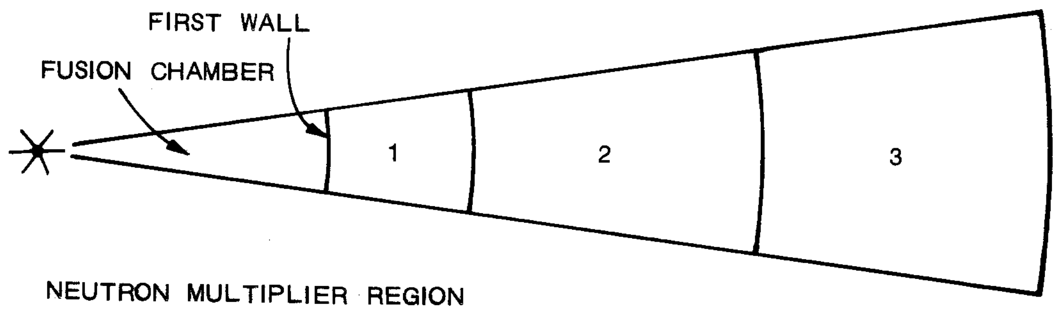
The results of the ANISN calculations described in Section II are given in Table 2. In Fig. 2, the tritium breeding ratio plus leakage per fusion neutron is plotted vs. the U fuel zone thickness for four different ^{239}Pu concentrations. The fuel zone thickness that is necessary to achieve a tritium breeding ratio equal to one is labelled on the figure. Figures 3 and 4 show the net ^{239}Pu breeding and the total breeding (^{239}Pu+T+leakage) for these blankets. There is a maximum tritium breeding ratio of about TBR=2 for a fuel zone thickness of 7 cm. After this maximum the tritium breeding ratio decreases with increasing fuel zone thickness. However, the plutonium breeding ratio, PuBR, continues to increase with increasing zone thickness and the total breeding ratio continues to increase.

This increase of the Pu breeding ratio might lead one to think that enriched blankets will be preferable to blankets that only contain natural

Table 1

Neutron 25 Energy Group Structure in eV Group Limits

Group	E(Top)	E(Low)	E(Mid Point)
1	1.4918 (+7)	1.3499 (+7)	1.4208 (+7)
2	1.3499 (+7)	1.2214 (+7)	1.2856 (+7)
3	1.2214 (+7)	1.1052 (+7)	1.1633 (+7)
4	1.1052 (+7)	1.0000 (+7)	1.0526 (+7)
5	1.0000 (+7)	9.0484 (+6)	9.5242 (+6)
6	9.0484 (+6)	8.1873 (+6)	8.6178 (+6)
7	8.1873 (+6)	7.4082 (+6)	7.7979 (+6)
8	7.4082 (+6)	6.7032 (+6)	7.0557 (+6)
9	6.7032 (+6)	6.0653 (+6)	6.3843 (+6)
10	6.0653 (+6)	5.4881 (+6)	5.7787 (+6)
11	5.4881 (+6)	4.4933 (+6)	4.9907 (+6)
12	4.4933 (+6)	3.6788 (+6)	4.0860 (+6)
13	3.6788 (+6)	3.0119 (+6)	3.3453 (+6)
14	3.0119 (+6)	2.4660 (+6)	2.7390 (+6)
15	2.4660 (+6)	1.3534 (+6)	1.9097 (+6)
16	1.3534 (+6)	7.4274 (+5)	1.0481 (+6)
17	7.4274 (+5)	4.0762 (+5)	5.7518 (+5)
18	4.0762 (+5)	1.6573 (+5)	2.8667 (+5)
19	1.6573 (+5)	3.1828 (+4)	9.8779 (+4)
20	3.1828 (+4)	3.3546 (+3)	1.7591 (+4)
21	3.3546 (+3)	3.5358 (+2)	1.8541 (+3)
22	3.5358 (+2)	3.7267 (+1)	1.9542 (+2)
23	3.7267 (+1)	3.9279 (+0)	2.0597 (+1)
24	3.9279 (+0)	4.1399 (-1)	2.1718 (+0)
25	4.1399 (-1)	2.200 (-2)	2.1800 (-1)



1 NEUTRON MULTIPLIER REGION
2 FUEL REGION
3 TRITIUM BREEDING REGION OR VACUUM DEPENDING ON TRITIUM BREEDING SYSTEM OR TRITIUMLESS SYSTEM

<u>FUELS</u>

- TYPE: METALS, OXIDES, CARBIDES
- FERTILE ISOTOPES: ^{238}U
- FISSILE ISOTOPES: ^{235}U, ^{239}Pu
- CLADDING: S.S.
- STRUCTURE: S.S., ZIRC 2

<u>COOLANT</u>

- LIQUID METAL Na

<u>NEUTRON MULTIPLIER</u>

- Pb

<u>COMPOSITION OF FUEL ZONE (VOL. %)</u>

- 63% FUEL, 24% Na, 8% CLADDING, 5% STRUCTURE

<u>COMPOSITION OF MULTIPLIER ZONE</u>

- Pb 82.2%, 9.3% Na, 8.5% ZIRC - 2

Figure 1

Table 2

Case	Pb Zone	U/Pu Zone		Li Zone	Total Breeding	^{239}Pu Breeding	T Breeding	Leakage	TBR + Leakage	M	k_{eff}
1	10	20	0%	40	2.237	1.854	0.384	0.352	0.736	7.06	0.416
2	10	60	0%	0	2.673	2.673	-	0.03	0.03	8.28	0.455
3	10	20	4%	40	3.043	2.119	0.925	0.878	1.803	29.25	0.758
4	10	60	4%	0	7.059	7.059	-	0.807	0.807	90.5	0.907
5	10	20	2%	40	2.496	1.936	0.559	0.522	1.081	14.33	0.60
6	10	60	2%		3.456	3.456	-	0.12	0.120	22.47	0.701
7	10	15	0%	45	2.073	1.486	0.587	0.443	1.030	6.28	0.39
8	10	20	3%	40	2.710	2.007	0.703	0.662	1.365	20.22	0.683
9	10	60	3%	0	4.433	4.433	-	0.277	0.277	40.35	0.809
10	10	47.5	4%	50	6.577	5.968	0.608	0.393	1.001	76.5	0.889
11	10	28.5	3%	40	3.363	2.855	0.508	0.479	0.987	27	0.742
12	10	21.5	2%	40	2.584	2.065	0.519	0.484	1.003	15	0.612
13											
14	10	60*	3%	0	2.412	2.412	-	0.088	0.088	14.5	0.608
15	10	20*	3%	40	1.853	1.349	0.504	0.433	0.937	9.5	0.499
16	10	20+	0%	40	2.005	1.647	0.358	0.309	0.667	5.4	0.349
17	10	60+	0%	0	2.370	2.370	-	0.023	0.023	6.4	0.389
18	10	60*	0%	0	2.146	2.146	-	0.025	0.025	5.4	0.343
19	10	20*	0%	40	1.812	1.446	0.366	0.303	0.669	4.5	0.302

* UO_2/PuO_2

+ UC/PuC

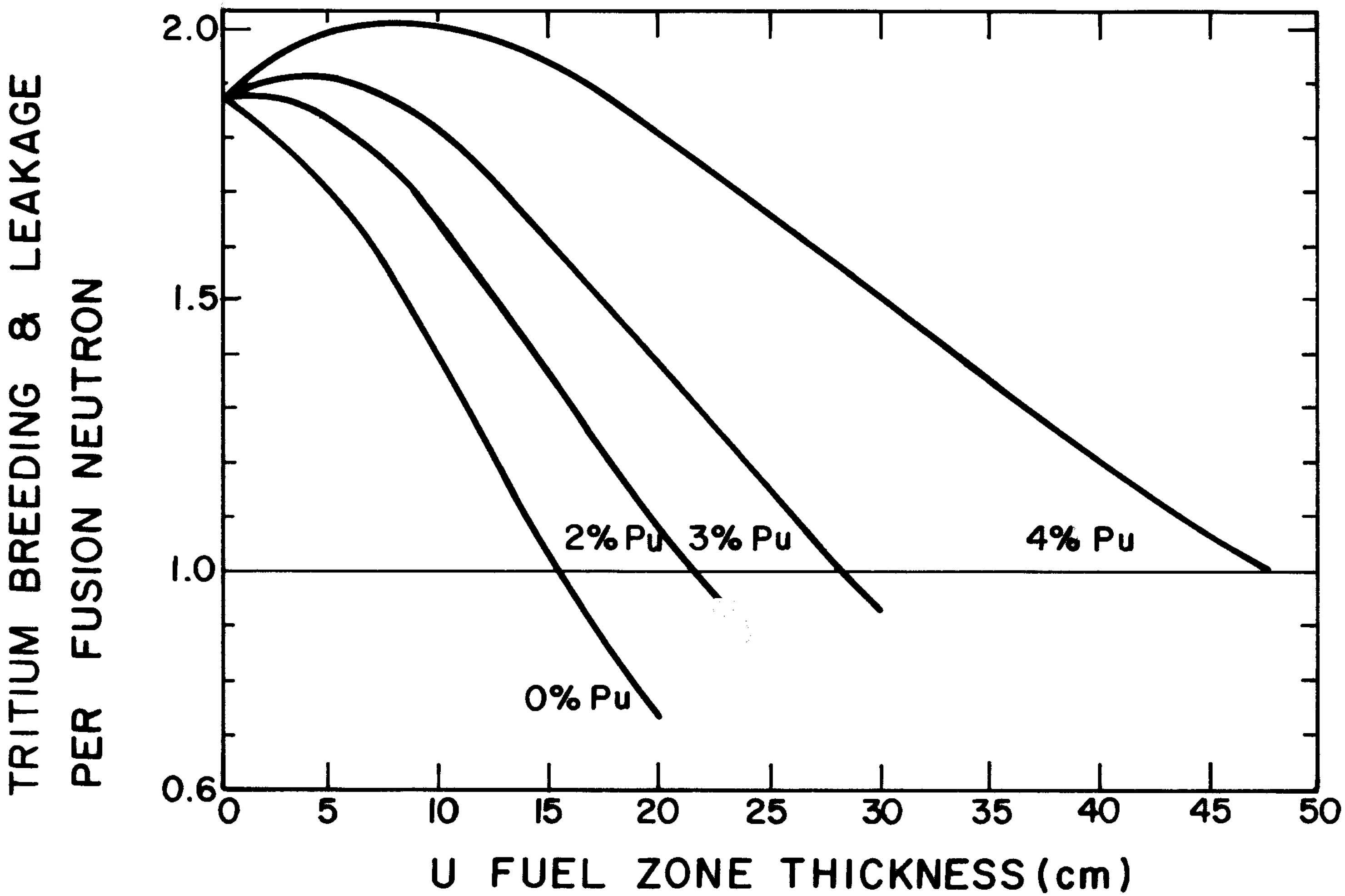

TRITIUM BREEDING & LEAKAGE
PER FUSION NEUTRON
2.0
1.5
1.0
0.6
0
5
10
15
20
25
30
35
40
45
50
U FUEL ZONE THICKNESS (cm)
0% Pu
2% Pu
3% Pu
4% Pu

Figure 2

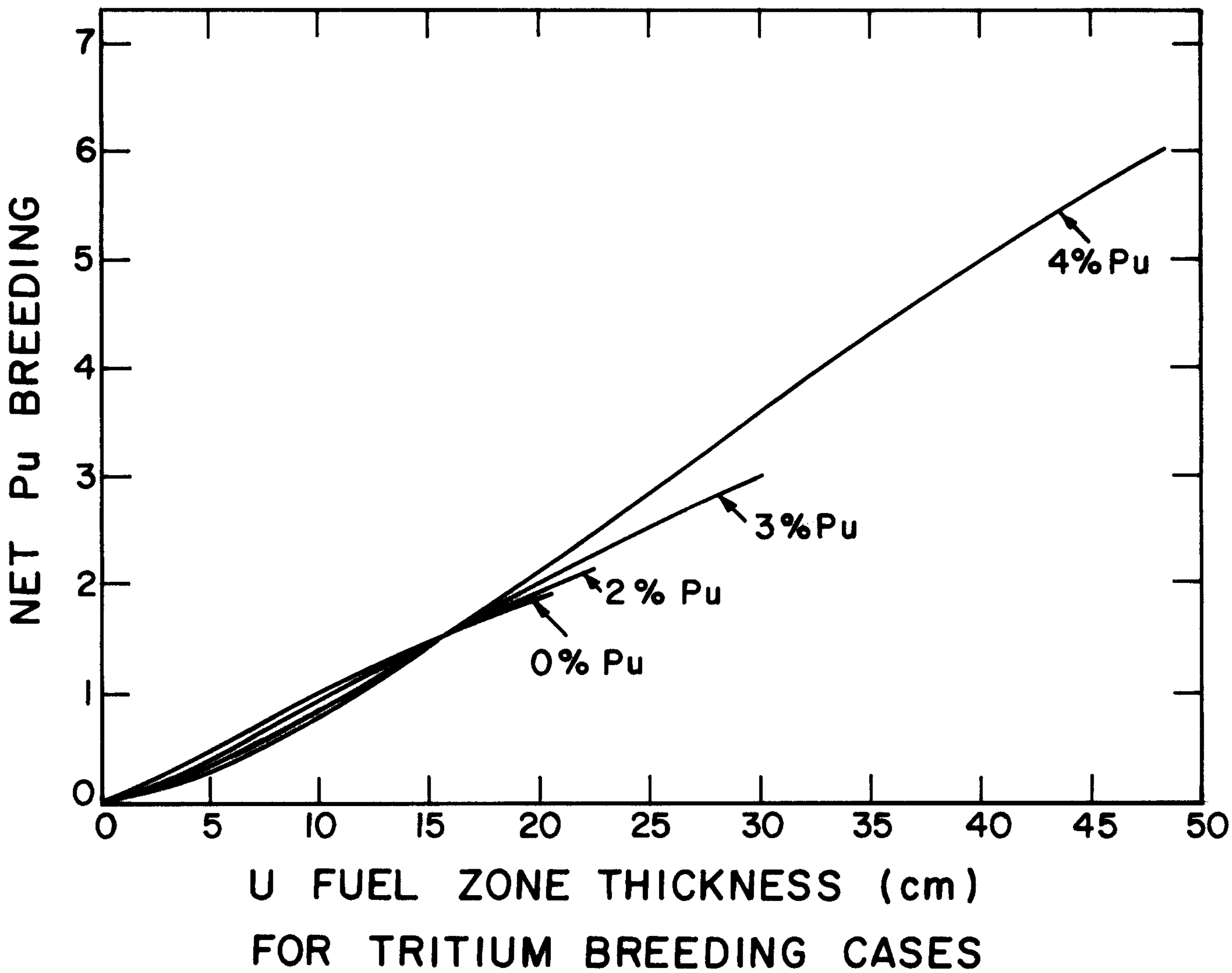

NET Pu BREEDING
0
1
2
3
4
5
6
7
0
5
10
15
20
25
30
35
40
45
50
4% Pu
3% Pu
2 % Pu
0% Pu
U FUEL ZONE THICKNESS (cm)
FOR TRITIUM BREEDING CASES

Figure 3

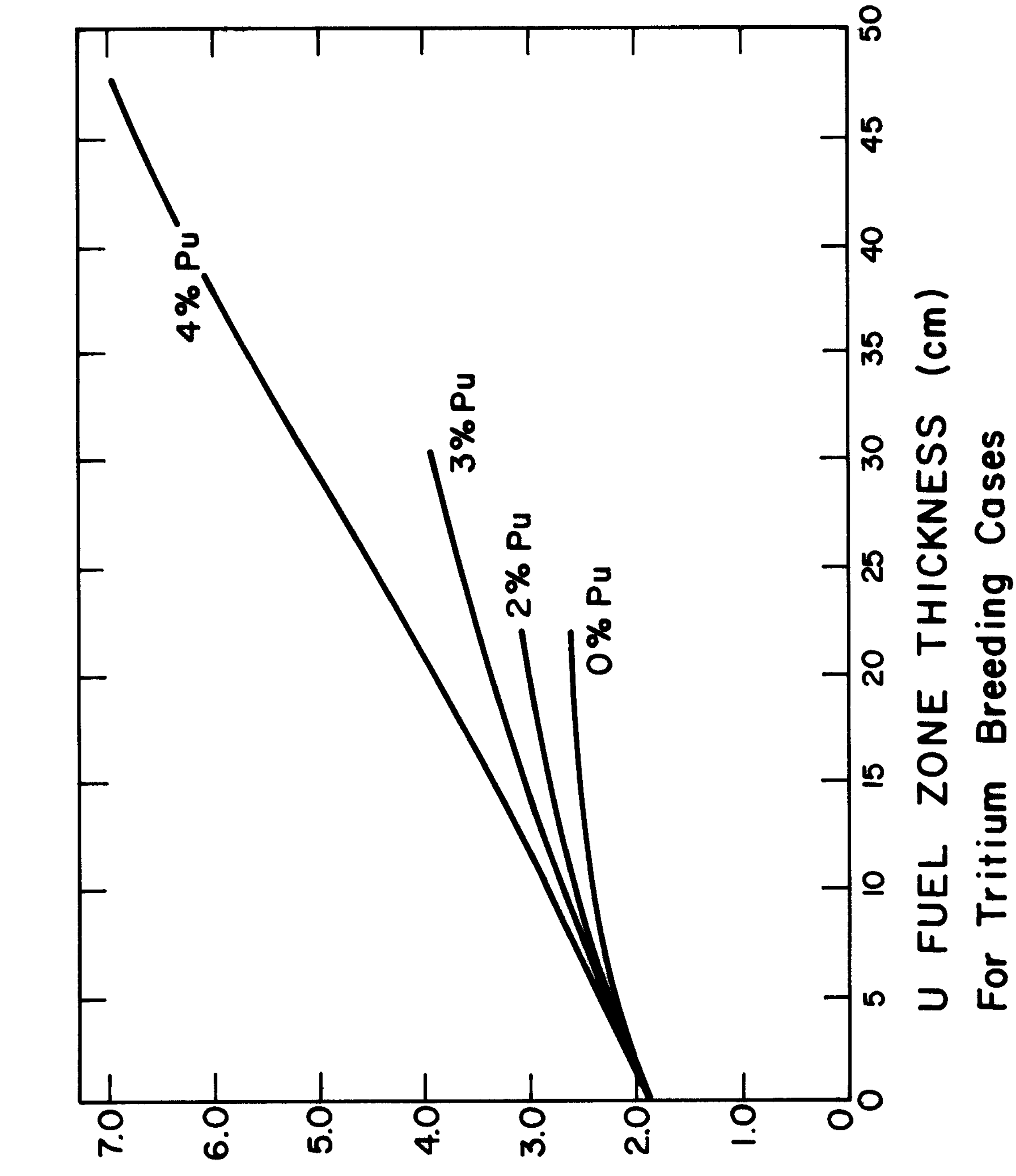

TOTAL BREEDING (Pu & Li & Leakage)
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
0
5
10
15
20
25
30
35
40
45
50
4% Pu
3% Pu
2% Pu
0% Pu
U FUEL ZONE THICKNESS (cm)
For Tritium Breeding Cases

Figure 4

uranium. However, this increase in the Pu breeding ratio comes at the expense of much higher blanket multiplication. In Fig. 5 the Pu breeding ratio is plotted vs. blanket multiplication for eight different blanket designs. Four of these have a tritium breeding ratio plus leakage equal to one and Pu concentrations of 0, 2, 3 and 4%. These correspond to the points that are labelled on Fig. 2 . The other four points are blankets with no tritium breeding so the leakage is added to the Pu breeding ratio that is plotted. The two lines that are formed by these sets of breeding ratios have constant and equal slopes. In addition, their displacement from one another is almost exactly one breeding capture. This figure clearly indicates that there is no hidden advantage in having a tritium-less blanket from a neutron balance point of view. Over a wide range of blanket multiplication the extra neutron goes to breed one more additional Pu atom. The total breeding ratio (^{239}Pu + T + leakage in one case and ^{239}Pu + leakage in the other) has the form

$$\text{Total BR} = 2.18 + 0.063\ M\ .$$

In Fig. 6 these results are compared to other fast spectrum Pu breeding blanket designs that are found in the literature. Note that these simple model calculations reproduce the general trend of the other "point designs" quite well. The Pu breeding ratio divided by the blanket multiplication gives a function that is proportional to the amount of Pu produced per

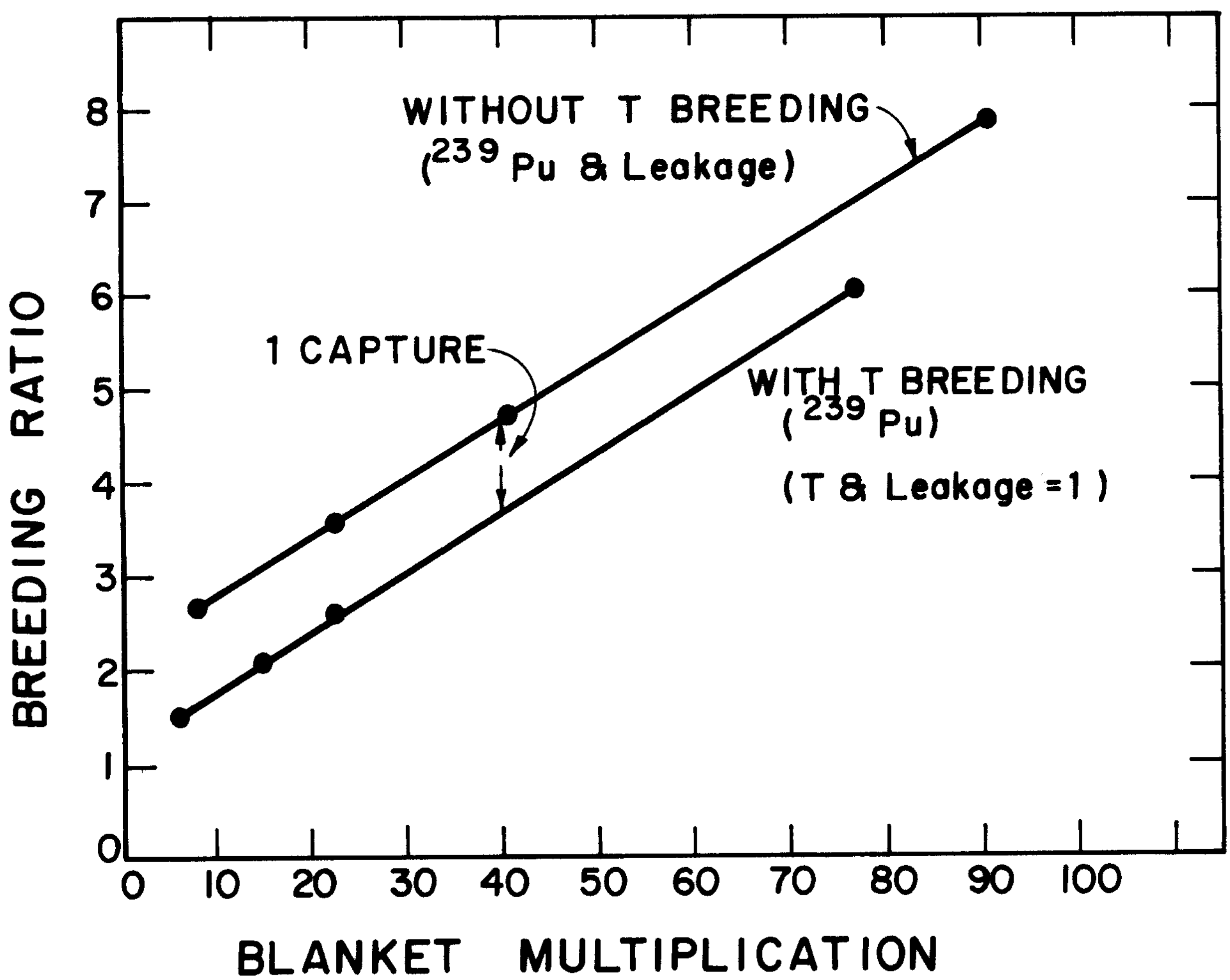
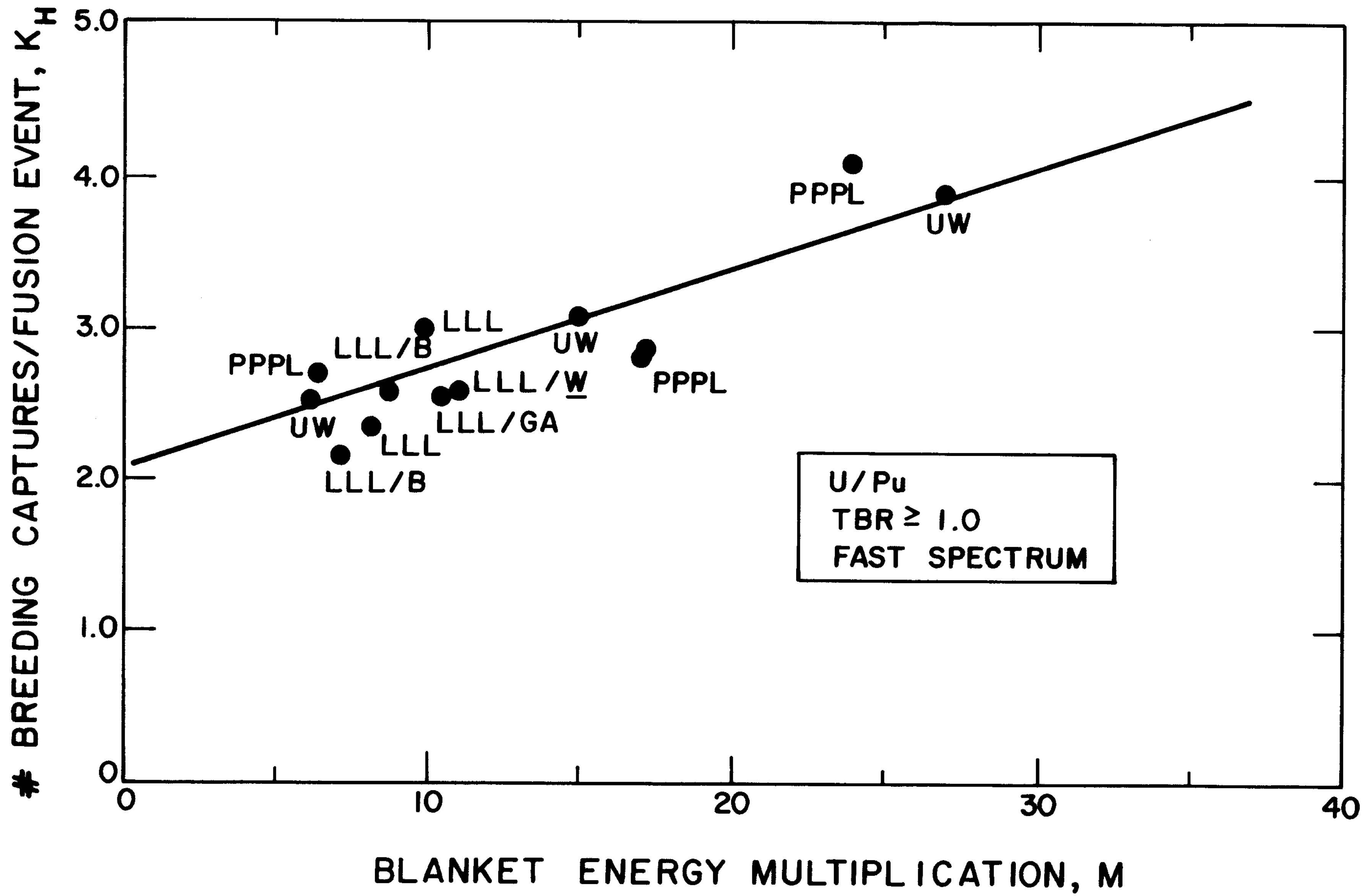

WITHOUT T BREEDING
(^{239}Pu & Leakage)
1 CAPTURE
WITH T BREEDING
(^{239}Pu)
(T & Leakage = 1)
BREEDING RATIO
BLANKET MULTIPLICATION
0
1
2
3
4
5
6
7
8
10
20
30
40
50
60
70
80
90
100

Figure 5



BREEDING CAPTURES/FUSION EVENT, K_H
BLANKET ENERGY MULTIPLICATION, M
5.0
4.0
3.0
2.0
1.0
0
10
20
30
40
PPPL
UW
LLL
LLL/B
LLL/W
LLL/GA
U/Pu
TBR ≥ 1.0
FAST SPECTRUM

Figure 6

thermal megawatt. These are plotted in Fig. 7 for the same cases, with and without T breeding. For blanket multiplication above 40 there is very little change in the amount of fuel produced/MW_t. For multiplications less than 40 this parameter increases sharply. Furthermore, there is growing separation between the T breeding and non-T breeding cases. However, this must be treated with caution because the tritiumless hybrid must obtain its tritium from somewhere else and this external T producing reactor (either fission or fusion) will have a thermal power associated with it. Before any final conclusions can be made, these results must be implemented into an economic model.

The power density in these blankets, normalized to a fusion rate of /neutron/sec, are shown in Figs. 8 and 9. Again, when equal blanket thicknesses are compared, there is not a great amount of difference between the cases when there is T breeding and when there is no T breeding.

Finally, several calculations are done to compare the performance of the metal fuel used in all of the previously reported calculations with oxide and carbide fuel. These comparisons are shown in Table 3. The total breeding plus leakage for oxide fuel is about 0.82 times that of metal fuel when there is no Pu enrichment. For carbide fuel it is about 0.89 times as great. When the fuel is enriched to 3% Pu the ratio of oxide to metal fuel total breeding plus leakage is 0.68. Oxide or carbide fuels would likely be used with the option of long exposure times in the hybrid blanket. Greater fissile enrichment would result from these longer exposure times. This analysis indicates that this would in fact be the least optimum method of fuel production, because the breeding performance of these ceramic fuels grows worse as the fissile enrichment increases.

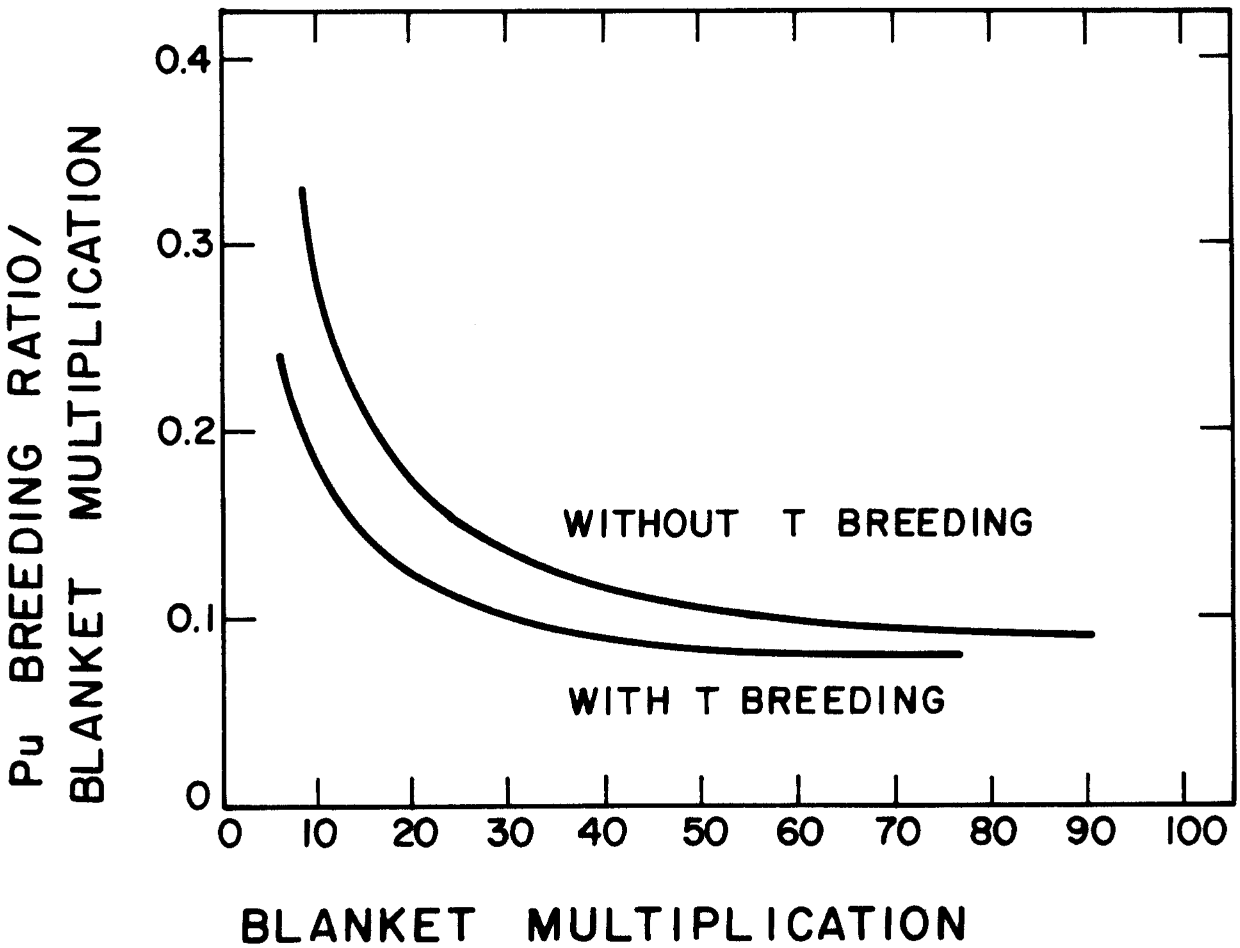

Pu BREEDING RATIO/
BLANKET MULTIPLICATION
0.4
0.3
0.2
0.1
0
WITHOUT T BREEDING
WITH T BREEDING
0
10
20
30
40
50
60
70
80
90
100
BLANKET MULTIPLICATION

Figure 7

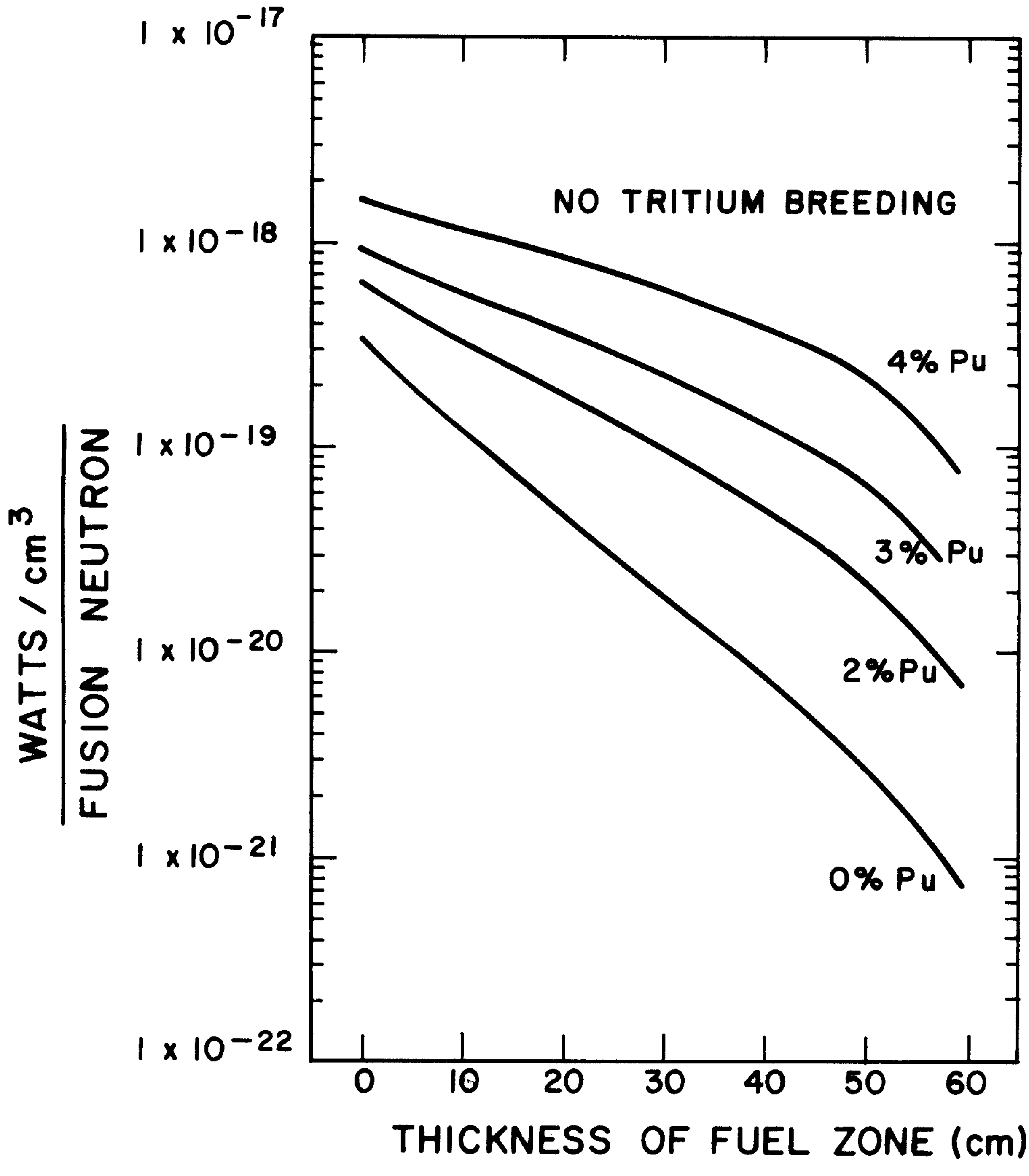

1 x 10^{-17}
1 x 10^{-18}
1 x 10^{-19}
1 x 10^{-20}
1 x 10^{-21}
1 x 10^{-22}
WATTS / cm^3
FUSION NEUTRON
NO TRITIUM BREEDING
4% Pu
3% Pu
2% Pu
0% Pu
0
10
20
30
40
50
60
THICKNESS OF FUEL ZONE (cm)

Figure 8

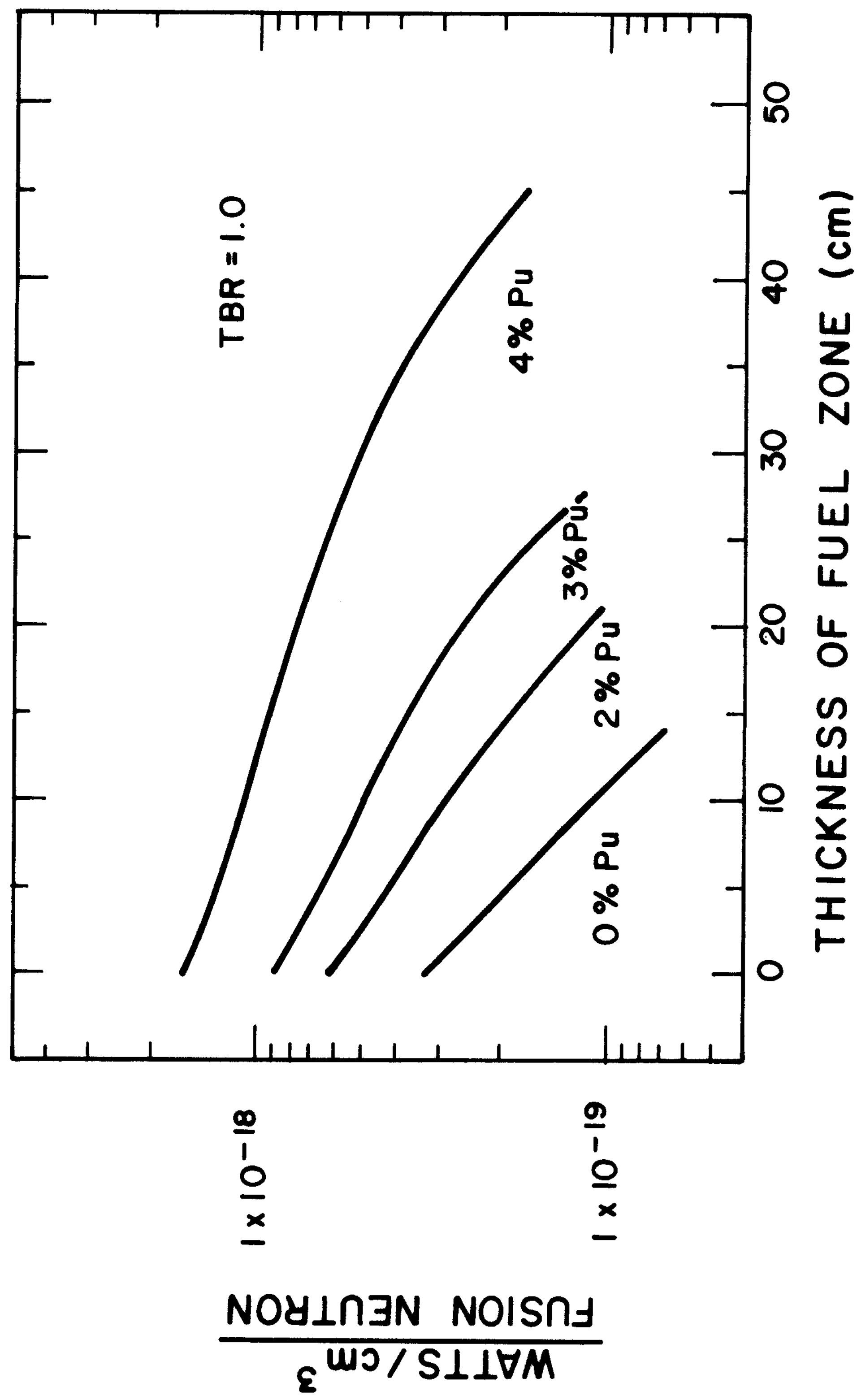

TBR = 1.0
4% Pu
3% Pu
2% Pu
0% Pu
1 x 10-18
1 x 10-19
WATTS/cm³
FUSION NEUTRON
0
10
20
30
40
50
THICKNESS OF FUEL ZONE (cm)

Figure 9

Table 3

Comparison of Metal, Oxide, and Carbide Fuel

	Pb Zone	U/Pu Zone		Li Zone	Total Breeding	^{239}Pu Breeding	T Breeding	Leakage	M	k_{eff}
Metal	10	20	0%	40	2.237	1.854	0.384	0.352	7.06	0.416
Oxide	10	20	0%	40	1.812	1.446	0.366	0.303	4.5	0.302
Carbide	10	20	0%	40	2.005	1.647	0.358	0.309	5.4	0.349
Metal	10	60	0%	-	2.673	2.673	-	0.03	8.28	0.455
Oxide	10	60	0%	-	2.146	2.146	-	0.025	5.4	0.343
Carbide	10	60	0%	-	2.370	2.370	-	0.023	6.4	0.389
Metal	10	20	3%	40	2.710	2.007	0.703	0.662	20.22	0.683
Oxide	10	20	3%	40	1.853	1.349	0.504	0.433	9.5	0.499
Metal	10	60	3%	-	4.433	4.433	-	0.277	40.35	0.809
Oxide	10	60	3%	-	2.412	2.412	-	0.088	14.5	0.608