A Survey of Fusion-Fission System Designs and Nuclear Analysis

M.Z. Youssef and R.W. Conn

June 1979

UWFDM-308

FUSION TECHNOLOGY INSTITUTE

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I. Introduction

The fusion-fission hybrid, when realized as a commercial power plant, will represent a long-term energy option. Utilizing the properties of the energetic neutrons from the D-T reaction in this system to breed fissile fuel (U-233 or Pu-239) by neutron capture in a fertile fuel (Th-232 or U-238) for subsequent use in fission reactors (LWR, HTGR) will substantially extend the naturally occurring fissile fuel supply (U-235) by the use of essentially non-exhaustible fertile materials.

The fusion-fission system can be designed and the neutron spectrum in the breeding blanket of that system can be tailored to produce two sources of revenue: fissile fuel and electric power. As far as breeding fissile fuel is concerned, the fusion-fission hybrid should far exceed the capability of the fast breeder (LMFBR) as a fuel source. In the following we discuss the role which the fusion-fission hybrid can play in this regard. The different philosophies for designing the blanket of the hybrid to perform different functions is reviewed based on the current fusion-fission designs in the published literature.

II. Comparison Between Fast Breeder and Fusion-Fission Systems as Fissile Fuel Breeders

The potential of fast breeders to provide the make-up fissile fuel needs for fission reactor burners and converters has been studied extensively.(1) It has been emphasized that a long-term energy option is achieved if the matched fission reactors have a high conversion ratio (e.g., tight lattice LWR or HTGR) and if the doubling time is sufficient to match expanding industrial needs. However, if the breeding characteristics of the fusion and fast breeder reactors are compared, the fusion-fission system will outperform the fast breeder. Such a comparison has been done by Fortescue[2] where he developed expressions to allow ready comparison of a hybrid fusion-fission plant and fast breeder with respect to the number of thermal reactors that their fissile fuel production could support, both for their fissile fuel needs and for the new inventory needs of an expanding industry.

One type of fusion-fission hybrid allows fission to take place in the fusion blanket and utilizes the 200 MeV per fission to multiply the 14.1 MeV D-T neutron energy. Breeding fissile fuel to be transferred to the fission reactors is also possible. If fissioning is not allowed in the fusion blanket and breeding fissile fuel is emphasized, the number of fission reactors that can be supplied will be enhanced and this integrated fusion hybrid-fission reactor symbiote will perform the fissile fuel breeding function primarily in the fusion blanket. Power is produced primarily in the fission reactor which also recycles fissile fuel in its core. Further, it is expected that complexity in the engineering design to retrieve the bred tritium from the fusion blanket may be simplified if the tritium production function is transferred to the fission reactors.[2] In the present work, several combinations of fusion-fission symbiotic systems are investigated which may include a tritium producer reactor dedicated mainly to tritium production. We will present the preliminary results of this study later.

In Fortescue's comparison, the fusion reactor plays the role of fissile fuel factory (no power is produced in the fusion blanket). The relations he derives[2] are expressed in terms of the neutron multiplication factor obtained in the fusion blanket, and the analogous quantities represented by the conversion ratio of the fast and thermal fission reactors included with the comparison.

The annual growth rate, G_A, that matches the industrial expansion, expressed as fraction of fissile inventory in the fission reactors (no fissile inventory is assumed in the fusion reactor), is[19]

$$G_A(\%) = \frac{10.95(C_{FB}-1)-(1+\alpha)(1-C)P}{\frac{(1+Z)P}{0.383\ RL} + 10.95\ D} \quad , \tag{1}$$

where the fusion reactor is the fissile fuel breeder. The corresponding expression in the case where the fast breeder reactor is used instead is

$$G_A(\%) = \frac{0.383\ R_B L(1+\alpha_B)}{f[1+Z_B+\frac{PR_B}{R}(1+Z)]} \times [(C_B - 1) - (1 - C)\ k\ P] \ , \tag{2}$$

where

$$k = \frac{(1+\alpha)f}{(1+\alpha_B)} \simeq 1$$

$f \equiv$ fast fission factor associated with the fast breeder

α, α_B = capture to fission ratio in the fission reactors and the fast breeder, respectively

C_{FB}, C_B, C = the total atoms (tritium + fissile) produced per DT neutron in the fusion blanket, the conversion ratio in the fast breeder and the fission reactors, respectively

$Z_B, Z = \frac{\text{out-of-core inventory}}{\text{in-core inventory}}$, in the fast breeder and the fission reactor, respectively

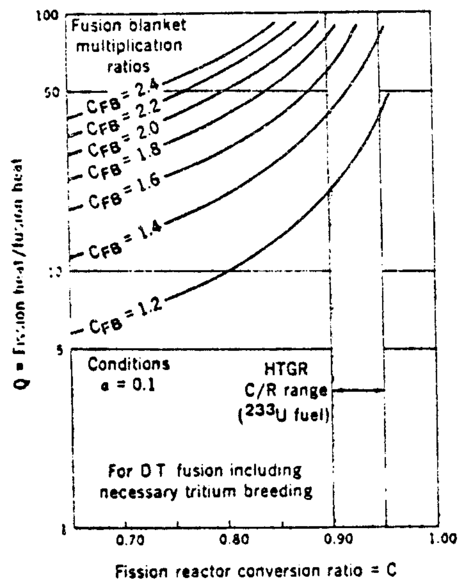
R_B, R = fissile rating per fissile initial inventory, $(\frac{MW_t}{kg})$, in the fast breeder and the fission reactors, respectively

L = Load factor

$$P = \frac{\text{thermal power of fission reactors}}{\text{breeder (fusion or fast reactor) thermal power}}$$

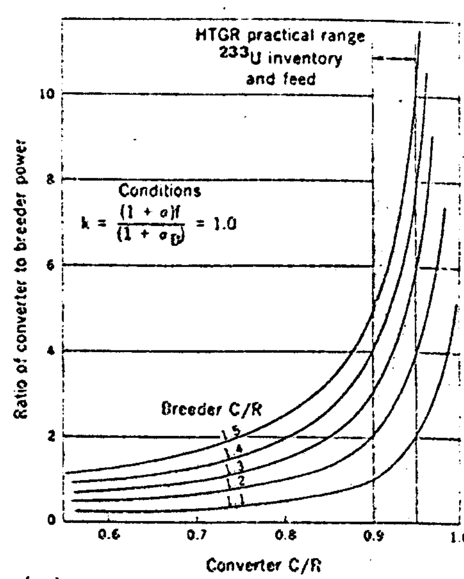
D = delay time in tritium processing (years).

Under steady-state conditions, with no allowance for system growth, we have G_A=0. If there is no delay in tritium processing (continuous extraction and feed), D=0. In Eq. (1), it is assumed that one fission produces ~ 11 times the heat from one fusion. Fig. (1) gives the ratio of the thermal power of the fission reactors to the fusion and the fast breeder thermal power, respectively, for different values of fission reactor conversion ratio and breeding capacity in the fusion reactor (C_{FB}) and the fast breeder C_B. Allowing for industry growth, G_A is represented in Fig. (2) for C=0.9 and for both fusion and fast breeders. The range of C_{FB} is typical of a symbiotic fusion-fission system (C_{FB} ~ 1.4) as will be shown later. The range of C_B values from 1 to 1.5 represents the potential value in a fast breeder. The range C_B = 0.9 - 0.95 is atypical of a HTGR based on U^{233} inventory and feed. Comparing Fig. (1)-a and Fig. (1)-b, it is clear that enormously higher numbers of converter reactors can be fed by one fusion plant of equal fusion power compared to the corresponding value if the fast breeder is used. Some 20 to 50 converter reactors (C=0.9) could be maintained by a fusion plant of equal power while only 2 to 5 could be supported with a fast breeder, a factor of 10 less. Also, notice from Fig. (2) the higher growth rate that can be obtained with the fusion breeder for a given value of breeding ratio. An analysis of this sort



(a) Ratio of the fission heat produced to the fusion heat required to supply the necessary fuel, under steady-state conditions.

G_A, D=0.0

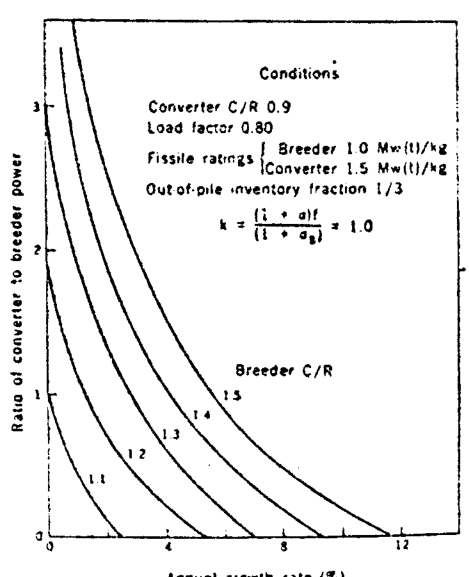


(b) Fast breeder-converter combination under steady-state conditions.

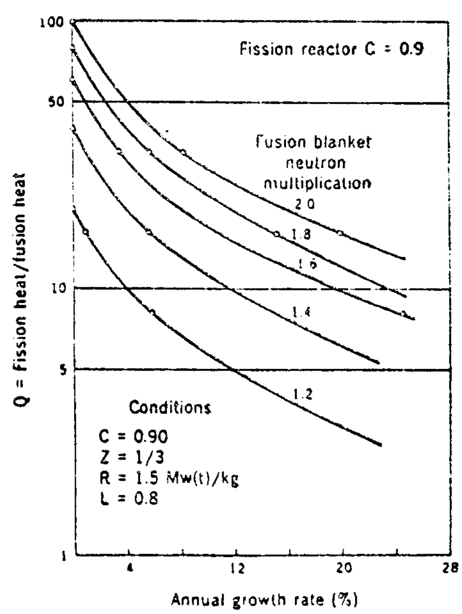
G_A, D=0.0

Fig. 1

Fig. 2



(a) Fast breeder-converter combination for self-sustaining growth.



(b) Growth of fusion-fission reactor association (zero tritium processing delay).

has been carried by Gordon and Harms[3] for such a symbiotic system where they demonstrated that a doubling time of less than a year is attainable.[3,4]

The potential of utilizing a symbiotic system for fissile fuel production using a molten salt containing ThF_4 for U^{233} breeding for subsequent use in a molten salt fission reactor (MSR) was first introduced by Lidsky[5] early in 1969. Recently, Blinkin and Novikov[6] optimized a fusion blanket similar to Lidsky's blanket to make the fuel handling and reprocessing simpler by devoting the MSR to breed only tritium while consuming the U^{233} supplied from the fusion reactor which carries ThF_4-NaF-BeF_2 molten salt. As they claimed, this will avoid the very difficult problems of generating tritium in a highly complicated fusion machine and will make the reprocessing much simpler. As they reported, reprocessing will consist merely of constantly removing U^{233} from the salt circulating in the fusion reactor blanket by fluorination and removing xenon from the fuel salt of the MSR by purging. Doubling times of ~ 4 years could be obtained compared to ~ 10 yrs evaluated by Lidsky while maintaining the same breeding capacity in the fusion blanket ($C_{FB} \sim 1.47$) and the same support ratio ($P \sim 10$).

From a safety point of view, the fusion breeder can be made definitely subcritical under all conditions. It is easier to control the flux level in the fusion breeder than in the fast breeder, because the latter is characterized by a short neutron lifetime. The reason is due to the properties of the energetic D-T neutron which has a "higher value" in terms of producing secondary neutrons in the fusion blanket from (n,2n), (n,3n) and/or fission than do neutrons resulting from fission or (n,3n) reactions. The contribution of these secondary neutrons to different reactions and energy production is small compared to the contribution due to the D-T neutrons and the system would shut itself down very quickly if the source neutrons were removed.

The difference in the energetics between the D-T neutrons and the subsequent secondary neutrons has suggested a computational scheme which treats the latter separately. Appreciable reduction in the computation cost is obtained, with high accuracy, when simple approximations are used (e.g., diffusion theory) to treat the secondary neutrons. This point will be elaborated and the results of this computational scheme will be presented in a later section of the present study.

The low fissile fuel inventory in the fusion-fission system is another advantage which renders this system a "safer-breeder". Problems associated with reforming critical assemblies upon core meltdown are minimized in a fusion breeder loaded with a fertile fuel.

It should be noted, however, in comparing the fast breeder and the fusion reactor as a "fissile fuel factory" that the fusion plant will not necessarily be a more economic proposition because there is no clear-cut knowledge about the cost of a fusion plant. Some economic studies have been done by Lawrence Livermore Lab. (LLL) in a joint effort with Bechtel Corp. for designing and estimating the cost of a laser driven fusion-fission system which produces Pu^{239} as well as power. Bechtel estimates the cost of such a system to be 2-3 times higher than a LWR of an equal power.(7) For other fusion drivers, the cost is expected to be within this range. Some other economic considerations can be found elsewhere.[8]

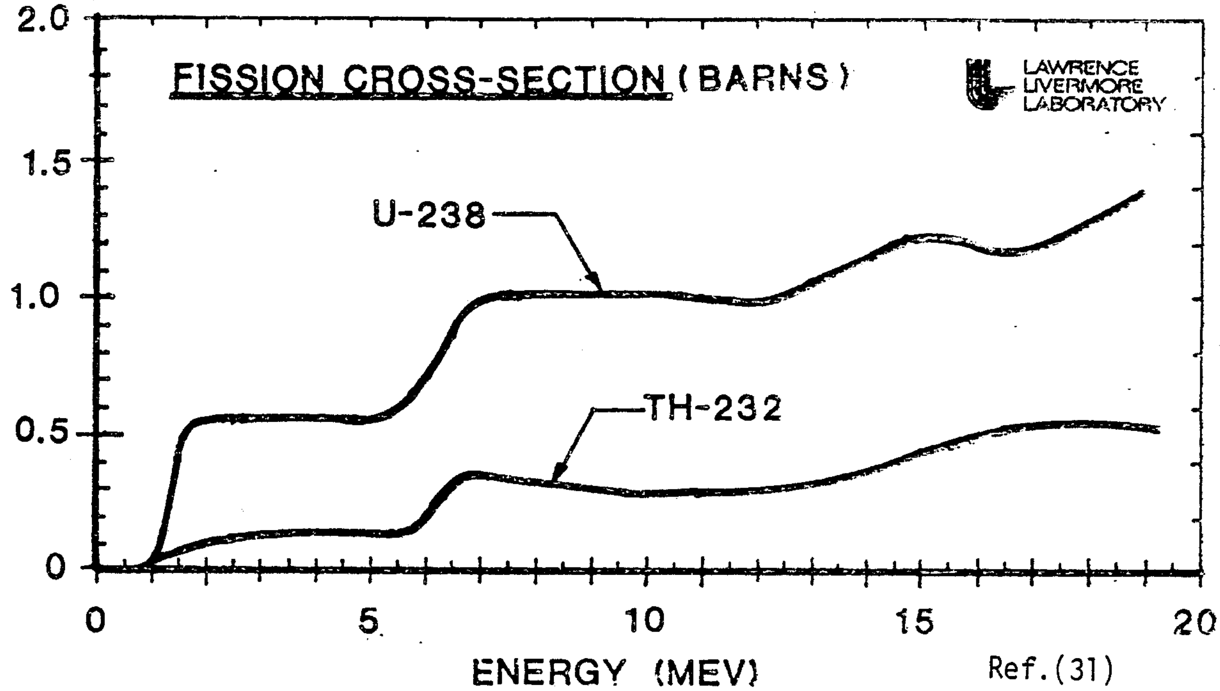
In the following section, the nuclear considerations in designing different fusion blankets to perform different functions are reviewed based on the current designs of fusion-fission systems.

III. Important Considerations in Fusion Hybrid Blanket Designs

In its simplest form, the fusion-fission system consists of a fusion component producing and containing a fusion plasma surrounded by a blanket which intercepts the fusion neutrons. The principal approaches to fusion being pursued are the tokamak[10] and mirror[11] magnetic confinement fusion, and laser[12] and electron beam heated inertial confinement fusion.[13] While other neutron producing fusion reactions are possible, namely, D-D, the much higher cross section for D-T at relatively low temperatures makes D-T fusion the practical choice for fusion-fission systems.

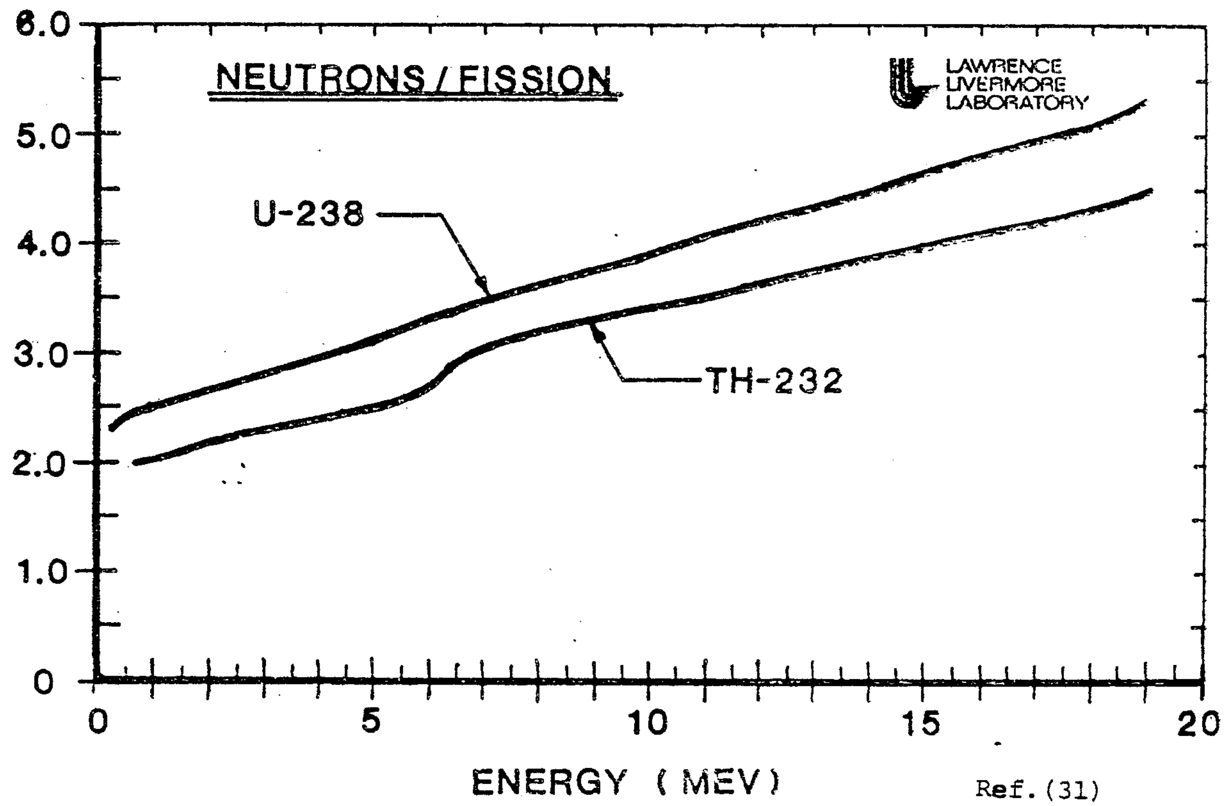
The purpose of the blanket is to produce tritium needed for the D-T reaction, produce fissile fuel from neutron capture in fertile material and/or to produce energy. The role of the blanket can vary from just producing fissile fuel[2],[5-6],[14-16] (symbiotic) to a nearly critical fission assembly[17] ($K_{eff} \simeq 0.94$).

To emphasize the potential of the D-T neutrons to be multiplied in the blanket, we show in Fig.(3) the fission cross section for U^{238} and Th^{232} as function of neutron energy. At 14.1 MeV, U^{238}(n,fiss) $\simeq$ 1.15 barns while Th^{232}(n,fiss) $\simeq$ 0.37 barns. Because of the 200 MeV released per fission,



Ref.(31)

Fig. (II.3)



Ref.(31)

Fig.4

the energy of the D-T neutrons is multiplied if fissioning occurs in the blanket. The fissioning in the blanket can be enhanced by noticing that most of the neutrons released per fission are themselves above the fast fission threshold. We give in Fig. (4) the average number of neutrons generated per fission vs. incident neutron energy. For 14.1 MeV neutron induced-fission, the number of neutrons generated in U^{238} is 4.5 and in Th^{232} is 3.87. The resultant energy multiplication will reduce fusion technology requirements (e.g., lower the value of $n\tau$, lower the magnetic fields, lower the beam energies, lower the fusion gain, ...). This relaxation in the design parameters may allow earlier commercialization of fusion and thus, an earlier return on investment in fusion research.[18-20]

The potential of using a fertile material for fissile fuel breeding and energy multiplication is shown in Table (1) which gives the energy multiplication M (defined as the ratio of the total energy deposited in the blanket per 14.1 MeV D-T neutron) and the breeding reaction (n,γ) per incident D-T neutron (this entry is called the fissile fuel conversion ratio C) in an infinite medium containing natural uranium, U^{238}, and Th^{232}. An energy multiplication $M \simeq 22$ can be obtained with a high fissile fuel conversion ratio ($C \sim 5$). Because of the lower fission cross section for Th^{232}, the capacity to breed U-233 is lower ($C \simeq 2.7$).

In a realistic blanket design, several considerations must be addressed in order to have a consistent blanket design.[21] Among those considerations are: nuclear performance, blanket geometry, refueling and replacement, tritium handling, heat removal, structural integrity, materials, safety, and

Table 1

Results for Infinite Medium (per D-T neutron)

(Ref. 21, 31)

Material	Energy Deposited MeV	M	Breeding Reaction (n,γ) ;(f/n)
Natural Uranium	309	22	5
Uranium-238	233	16	4.4
Thorium	64	4.5	2.7

Table 2

Blanket* vs. Infinite Medium and Fuel Type

(per 14.1 MeV D-T Neutron)

(Ref. 21, 31)

	T	f/n	(MeV)	M
Infinite Medium (natural uranium)	0.0	5.0	309	22
Blanket* With U	1.1	2.2	200	14
Blanket* With UO_2	1.1	1.1	100	7.1
Blanket* With UC	1.1	1.4	130	9.3

*Blanket:

Zone 1: 69% U + 10% SS + 16% Li

Zone 2: 86% Li + 9% SS

environmental issues. Tritium breeding to sustain the D-T reaction and high fissile breeding to supply several fission reactors are two major nuclear considerations. The blanket geometry should conform to the D-T neutron source and allow for penetration, blanket refueling, and replacement. Tritium removal and containment methods are important considerations because choices made affect tritium breeding rates needed. Tritium is provided through $^{6}Li(n,\alpha)T$ and/or $^{7}Li(n,\alpha,n')T$ reactions. Natural or enriched lithium can be used as a coolant for the front zone and/or the fuel zone containing the fissile fuel breeding zone. As a rule, breeding tritium from ^{6}Li is performed in zones which have substantially-moderated neutrons where the $^{6}Li(n,\alpha)T$ reaction is high. The total tritium breeding ratio, T, defined as the number of tritium atoms produced in the blanket per D-T neutron, should be slightly greater than 1.0 to provide the necessary fuel for the D-T reaction and to substitute for tritium losses and decay. As mentioned before, in some designs, the tritium breeding function may be eliminated from the blanket and be performed in the fission reactors.

High power density, particularly in the front zone of the blanket facing the high 14.1 MeV neutron flux, represents a design challenge and an appropriate heat removal system is needed to remove the excessive heat from this zone and the rest of the blanket. Structural integrity dictates the type of structural material to be used in the blanket. The performance of the blanket is quite sensitive to the ratio of structural material to fuel due to neutron competition. The type of fuel used (metal, carbides, oxides) is both important in nuclear design as well as fuel integrity. For example, uranium metal has swelling problems at burnups and temperatures that are too low to be of much interest for hybrids which emphasize power production.[12,21] It is a suitable fuel material for fusion systems which emphasize fuel production. In Table (2), we show how the blanket requirements and design trade-offs may affect the blanket performance. It is clear from this table that requiring a natural uranium blanket to breed tritium containing structure results in a plutonium breeding ratio of ~ 2 compared to 5 in the case of infinite medium. Also, using ceramic uranium fuel will reduce the nuclear performance of the blanket (e.g., UO_2 performance is 1/2 U-nat. performance and UC performance is 2/3 U-nat. performance). Ceramic fuel may be used in blankets which have high power densities ($> 200 \frac{W}{cm^3}$) with a high flux influence. Due to the buildup of fission products and high radioactivity levelsthroughout the blanket, remote handling is necessary to replace the first wall (subjected to radiation damage due to long irradiation) and for refueling.

IV. Classes of Hybrid Blankets

Several different fusion-fission system designs have been investigated and their performance is reported in the literature. These systems have been extensively investigated during the last four years although the early work in that field appears in 1953.[22] Leonard has recently provided a bibliography of fusion-fission publicationslisting some 160 entries.[23] Refs. (15) and (24) give a recent review of the fission-fusion systems. Different designs can be extracted from Refs. 25-28.

Based on the different designs, the type of blanket can be classified as: uranium fast fission blanket, thorium fast fission blanket, thermal fission blanket, plutonium and U^{238} enriched fast fission blankets, and the non-fissioning blanket. Some of these blankets have been designed to emphasize fissile fuel production while others emphasize power production. As mentioned before, in the non-fissioning blanket, fissioning in both the fertile and the bred fissile fuel is suppressed. In a thermal fissioning blanket, fissioning is encouraged in the bred fuel and the blanket is devoted to power production. In other types of blankets fissile fuel production is emphasized with power produced as a by-product to improve the economics of the system. Except for the non-fissioning blanket, the fusion-fission system is termed a hybrid.

IV.1. Front Zone Neutron Multiplier

It has been argued that including a front zone to multiply the D-T neutrons will improve the blanket performance.[29-30] For example, in a hybrid system, including a "fission plate" containing depleted uranium with a concentration 53% U^{238}, 35% sodium, and 12% stainless steel will multiply the fusion neutron energy by a factor of 3-7 and the number of neutrons leaving the fast fission blanket will be 1.6-1.9 times larger than the number of incident fusion neutrons if the thickness of this front zone varies from 4-15 cm.[29] Pu^{239} can be added to this fission plate to enhance the neutron multiplication through fissioning.[17] In general, the material choices for this zone are:[19]

A. Fuels:
   1. Type: metals, oxides, or carbides
   2. Fertile isotopes: Th^{232} or U^{238}
   3. Fissile isotopes: U^{233}, U^{235}, or Pu^{239}
   4. Cladding: stainless steel or refractory metals.

B. Coolants: gas or liquid metal

C. Structure: Stainless steel or refractory metals.

If fissioning is suppressed throughout the blanket, the front zone may include materials (Be, Mo, Nb, Pb, ...) to multiply neutrons through (n,2n) reactions. TZM material can also be used (as in Lidsky's [5] symbiotic system).

The most crucial issues affecting the technical and economic feasibility of fusion-fission hybrid systems occurs in the blanket region closest to the fusion neutron source because of the exposure to the high levels of energetic neutron flux and if fissioning takes place in the neutron multiplier zone, the power density in the front zone may be excessively high.[17] Therefore, important performance indicators such as first wall flux, blanket power density, and blanket lifetime will be determined by the conditions in the region closest to the fusion source.[29]

IV.2. Different Options Possible After the Front Zone

The different blanket types are named in the literature according to the function performed in the region next to the front zone. Hybrid blankets can be designed to maximize breeding (tritium and/or fissile material) or fusion-neutron energy multiplication. In general, fissile production per source neutron (f/n) is maximized in uranium or thorium fast fissioning blankets. Thus, fissioning in the fertile material is used for enhancing the neutron multiplication for breeding purposes, and fissioning in the bred fuel is minimized. On the other hand, energy multiplication is maximized in thermal-fission blankets containing heterogeneous lattices of fissionable material and moderators by enhancing fissioning in the bred fuel. Substantial portions of the spectrum of different fission reactor technologies can be employed in designing these blankets. Thus, the fusion-fission systems based on these options will minimize the changeover of the present fission reactor technology. As far as material choices are concerned, for thermal-fission blankets we have:[(19)]

A. Fuels

1. Type: oxides or carbides
2. Fertile isotopes: Th^{232} or U^{238}
3. Fissile isotopes: U^{233}, U^{235}, Pu^{239}
4. Cladding: graphite, zirconium, or stainless steel

B. Moderators: graphite or hydrides

C. Coolant: gas or liquid metal

D. Structure: materials with low thermal-neutron absorption.

The material choices for U- or Th- fast fission blankets may be the same as in the front zone fission plate.

One of the options that can be adopted is to use the fusion-fission system to breed tritium. Although this option has not been studied in the literature, we mention it here for completeness. In this case, lithium behind the front zone neutron multiplier is used to breed excess tritium for the startup of pure fusion reactors.

In the symbiotic system, the zone following the front zone neutron multiplier is used for breeding fissile fuel without allowing fissioning to occur in both the fertile and the bred fuel. An intermediate zone containing a low-Z material may be added between this zone and the front zone to serve as a moderator. The neutrons reaching the fuel zone will be thermalized and neutron capture in the fuel zone is enhanced. Because of the competition between (n,γ) reactions and $^6Li(n,\alpha)T$ reactions at low neutron energies, the amount of ^6Li present in the fuel zone should be minimized to assure high fissile fuel production rates.

IV.3. Reflector and Tritium Breeding Zone

Most of the neutrons are thermalized when they reach the back edge of the blanket. It is in this region where the $^6Li(n,\alpha)T$ reaction rate is high and thus serves for tritium breeding. A reflector zone is included to lessen neutron leakage. We summarize in Fig. (5) the role of the blanket in fusion-fission systems.

V. Typical Performance for Different Fusion-Fission Systems

Based on a literature survey, we give in Appendix A a summary of some of the current conceptual designs of fusion-fission systems with a schematic representation of the blanket used in each design. These designs reflect the different philosophy of designing a hybrid system (e.g. U-fast fission, Th-fast fission, etc. ...). The important parameters of interest are: energy multiplication (M), fissile atoms produced per D-T neutron (f/n), and tritium atoms produced per D-T neutron, T.

We summarize in Fig. (6) the performance parameters of fusion-fission systems which breed U-233. One notices the following:

- The energy multiplication, M, is almost a linear function with the fissile breeding ratio f/n for Th-non-fissioning blankets and Th-fast fissioning blankets. Allowing for fissioning in the latter, f/n and M are slightly higher than the corresponding values in the Th non-fissioning blanket. In both cases, the value

Figure (5) Schematic Diagram of the Blanket of Different Fusion-Fission Systems

D-T Source

First Wall

• Blanket Type: • Zones:	(Symbiosis)	(Hybrid)	
Zone A: Multiplying Medium Front Zone	$^{9}Be(n,2n)$ and/or $Pb(n,2n)$ and/or $Mo(n,2n)$ and/or $Nb(n,2n)$ and/or $^{7}Li(n,n',\alpha)T$	Fast Fission Plate	$U^{238}+Pu^{239}$ $U^{238}(n,fiss),(n,2n)$ $(n,3n)$ $Pu^{239}(n,fiss)$, $(n,2n)$ $(n,3n)$ $U^{238}(n,\gamma)$
Zone B: Intermediate Moderator Zone	Excess Neutrons	Excess Neutrons	
Zone C: Breeding and/or Energy Production Zone	$^{6}Li(n,\alpha)T$ $Th^{232}(n,\gamma)$ U^{233} (Zone B may be graphite)	• Thermal Lattice to Produce Power • Uranium Fast Fission $^{238}U(n,fiss)$ $^{238}U(n,\gamma)$ Produces Pu^{239} • Thorium Fast Fission $^{232}Th(n,fiss)$ $^{232}Th(n,\gamma)$ Produces ^{233}U • Tritium Production $^{6}Li(n,\alpha)T$ $^{7}Li(n,n',\alpha)T$ Lithium to Produce Excess Tritium	
Zone D: Moderator and Tritum Breeding Zone	(^{233}U to HTGR and/or Tritium)	(^{239}Pu and/or ^{233}U and/or Tritium)	

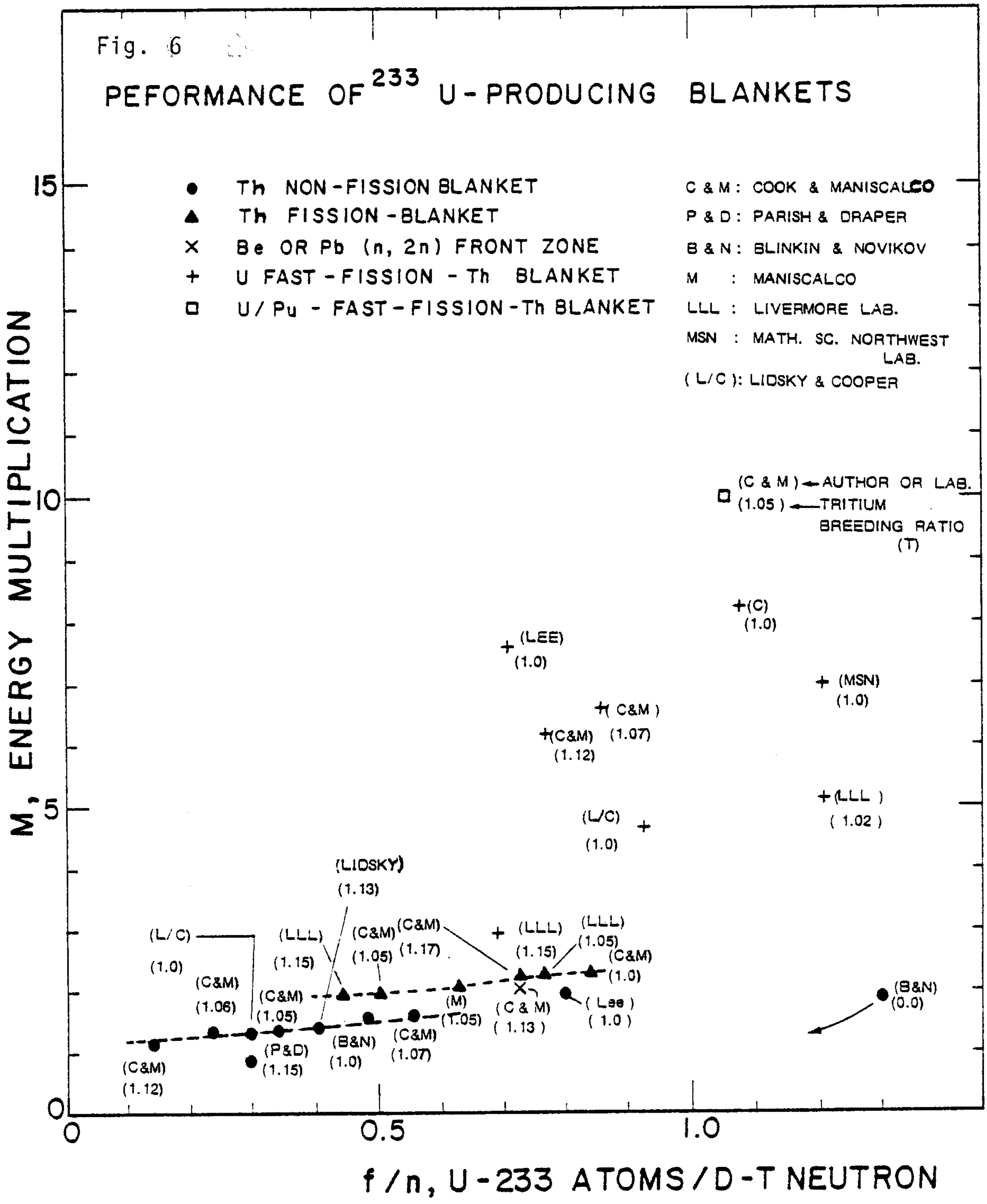


Fig. 6
PEFORMANCE OF 233 U-PRODUCING BLANKETS
Th NON-FISSION BLANKET
Th FISSION-BLANKET
Be OR Pb (n, 2n) FRONT ZONE
U FAST-FISSION - Th BLANKET
U/Pu - FAST-FISSION-Th BLANKET
C & M: COOK & MANISCALCO
P & D: PARISH & DRAPER
B & N: BLINKIN & NOVIKOV
M : MANISCALCO
LLL : LIVERMORE LAB.
MSN : MATH. SC. NORTHWEST LAB.
(L/C): LIDSKY & COOPER
(C & M) ← AUTHOR OR LAB.
(1.05) ← TRITIUM BREEDING RATIO (T)
M, ENERGY MULTIPLICATION
f/n, U-233 ATOMS/D-T NEUTRON

of the total breeding capture (C_{FB} = f/n + T) $\simeq$ 1.2 - 1.8 while $M \simeq 1 - 3$.

- In blankets which utilize fast fission in U-238 as a neutron and energy multiplier in the front zone, the values of f/n and M are higher. M is a factor of 3-7 higher than the no-fission-front zone blankets and f/n is a factor of 1.3-1.9 higher. Including Pu in the front zone will enhance neutron and energy multiplication throughout the blanket. Values of f/n $\sim$ 4 and M $\sim$ 80 can be obtained in this case as in the Su & McCormick design.[(17)] This will be at the expense of having higher power density in the front zone and the presence of fissile inventory in the blanket.
- As the spectrum gets softer in the blanket and fissioning of the bred U-232 is allowed, the energy multiplication (M) does not follow a linear relation with f/n as in the non-fissioning blankets.
- If fissioning is suppressed in the blanket, including a Be (or Pb) front zone multiplier will improve the blanket performance due to neutron multiplication through (n,2n) reactions.

In Fig. (7) the performance of blankets which breed Pu^{239} is presented (see Appendix A). Higher energy multiplication and breeding ratios are obtained in these blankets compared to the U-233 breeding blanket. This is due to the higher fission cross section and neutron yield for U^{238} compared to Th^{232} at 14.1 MeV neutron energy. The dispersion between the values of M and f/n for different designs reflects the deviation from linearity as more fissioning is encountered in the blanket. However, this is less noticeable in the fast

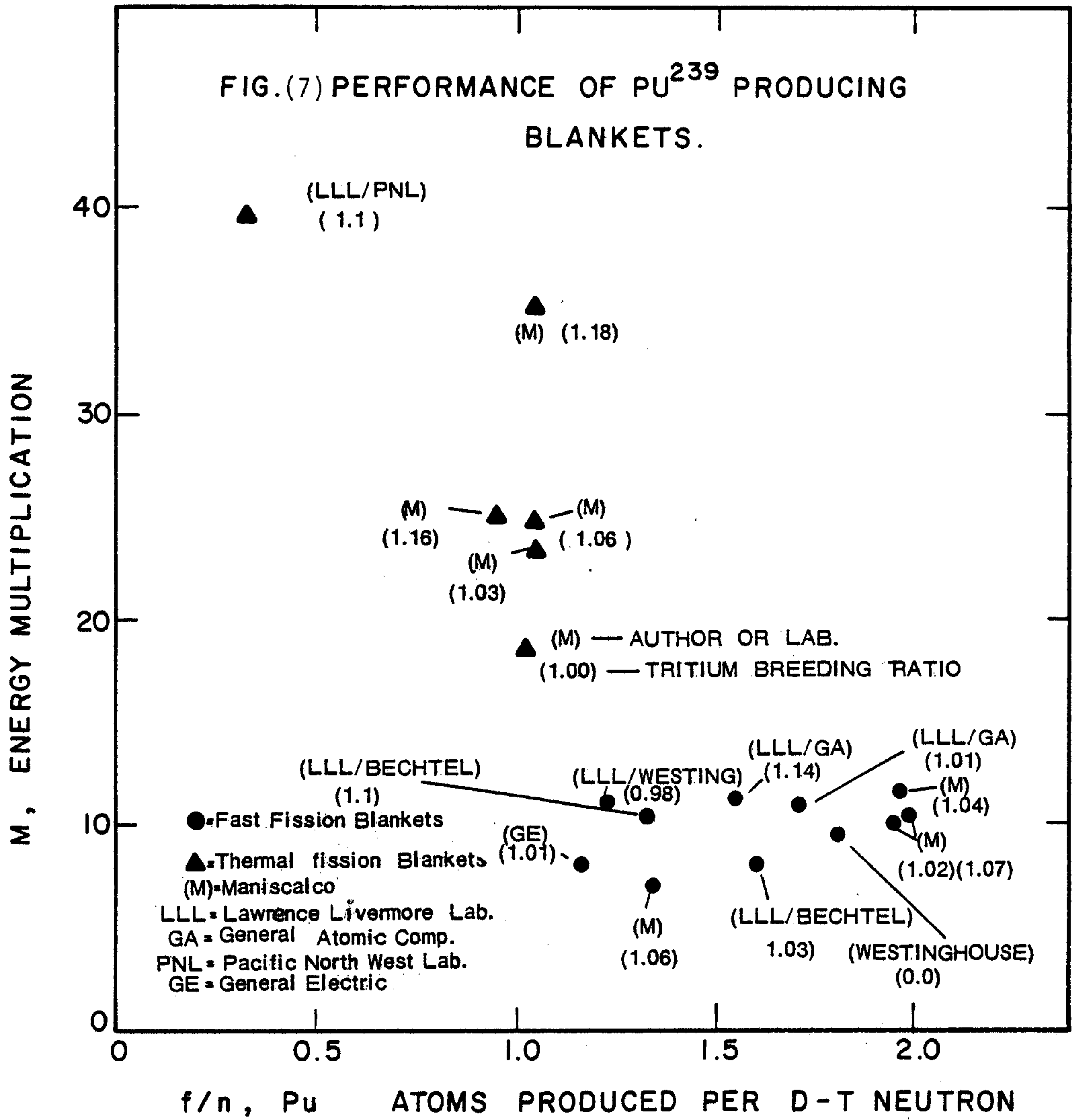


FIG.(7) PERFORMANCE OF PU239 PRODUCING BLANKETS.
M, ENERGY MULTIPLICATION
f/n, Pu ATOMS PRODUCED PER D-T NEUTRON
0
10
20
30
40
0.5
1.0
1.5
2.0
(LLL/PNL) (1.1)
(M) (1.18)
(M) (1.16)
(M) (1.06)
(M) (1.03)
(M) — AUTHOR OR LAB.
(1.00) — TRITIUM BREEDING RATIO
(LLL/BECHTEL) (1.1)
(LLL/WESTING) (0.98)
(LLL/GA) (1.14)
(LLL/GA) (1.01)
(M) (1.04)
(M) (1.02)(1.07)
(GE) (1.01)
(M) (1.06)
(LLL/BECHTEL) 1.03)
(WESTINGHOUSE) (0.0)
●=Fast Fission Blankets
▲=Thermal fission Blankets
(M)=Maniscalco
LLL = Lawrence Livermore Lab.
GA = General Atomic Comp.
PNL = Pacific North West Lab.
GE = General Electric

fissioning blankets. In fact, having higher multiplication will be on the expense of fissile fuel bred. Table (3) summarizes the expected values of energy multiplication and fissile breeding ratio for different types of blankets based on Th/U and U/Pu fuel cycle.

VI. Comparison Between Th/U and U/Pu Fuel Cycles

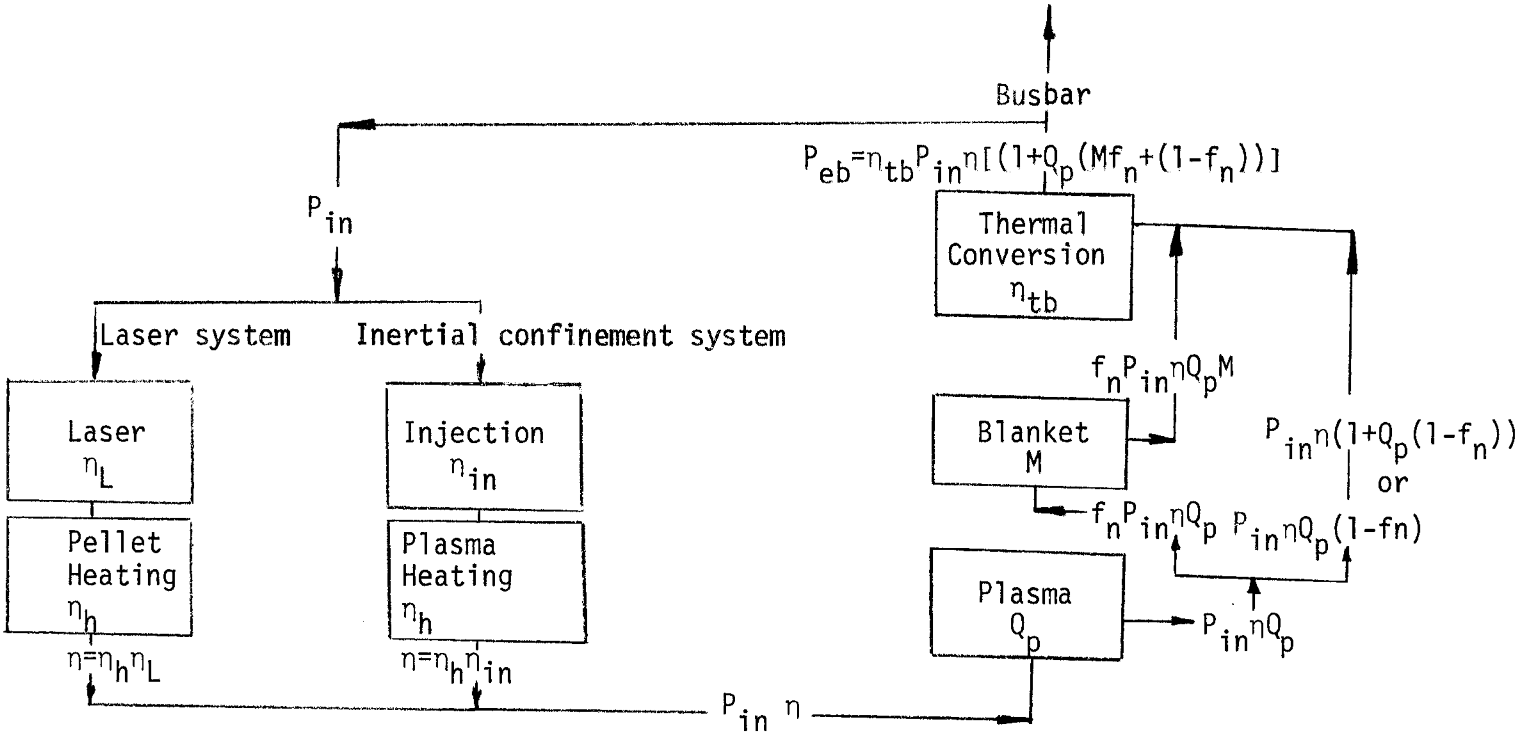
Because of the higher energy multiplication in U/Pu blankets, the fusion driver requirements are relaxed in designs using these blankets from those using Th/U blankets for the same fraction of recirculating power needed for the fusion driver. This can be seen from Fig. (8) where Fig. (8)-a describes the energy balance and Fig. (8)-b shows the effect of the blanket energy multiplication on the fusion gain (fusion gain = $\frac{\text{thermal fusion power}}{\text{electric input to the fusion driver}}$) for different values of recirculating power fraction. As shown, the fusion gain decreases drastically as M increases. In particular, for a fusion-fission system which operates as a fissile fuel producer (recirculating fraction ~ 100%), the uranium-fueled fast fission blankets can produce fissile fuel in an energy breakeven mode with fusion energy gains that are 2-6 times lower than those required for similar thorium-fueled blankets and 30 times lower than those required for a pure fusion power plant where M=1.[19] For systems which are energy producers (recir. power $\lesssim$ 25%) and in the case of the U/Pu fast fission-thermal blanket ($|U+Pu|Th+U^{233}+Li|$) the fusion gain is 20 times lower than for a thorium-fast fission blanket with the same recirculation fraction.

However, blankets based on Th/U fuel cycles will breed U-233 which is a superior fuel for a near breeder reactor (HTGR, MSR) because of its low capture to fission ratio (~ 0.1 compared to 0.3 in a LWR burner based on a U/Pu cycle). With the fusion-fission reactor as a fissile fuel factory,[32] the supporting ratio (defined as the number of fission reactors of equal thermal power to the hybrid reactor which are supported by their fissile

Table 3

Different Blanket Types and the Expected Performance

Cycle	Type of Blanket	Representation of the Blanket	M	f/n	C_{FB}=f/n+T
Th^{232}/U^{233}	Non-Fissioning	$\vert Li \vert Th \vert$	1.0-1.2	0.13-0.8	1.2-1.8
	Be Front Zone	$\vert Be+Li+Th \vert$	$\sim$ 1.6	$\sim$ 0.8	$\sim$ 1.8
	Th-Fast Fission	$\vert Th \vert Li \vert$	1.5-2.5	0.5-0.9	1.5-1.9
	U-Fast Fission	$\vert U \vert Th+Li \vert$	3-10	0.8-1.5	1.8-2.5
	U/Pu-Fast Fission	$\vert U+Pu \vert Th+Li \vert$	> 10	> 1.2	> 2.2
	U/Pu-Fast Fission (Thermal Th-Blanket)	$\vert U+Pu \vert Th+U^{233}+Li \vert$	> 30	-	1.0
U^{238}/Pu^{239}	U-Fast Fission	$\vert U \vert Li \vert$	7-11	1.1-2	2.2-3
	U/Pu-Thermal Blankets	$\vert U+Pu \vert U+Li \vert$	> 20	$\sim$ 1.0	2.1



Circulating power fraction $= \dfrac{P_{in}}{P_{eb}} = \dfrac{1}{\eta_{tb}\eta[1+Q_p(0.8M+0.2)]}$ $\left\{\begin{array}{l}\text{power to heat the}\\ \text{plasma is retained,}\\ f_n=0.8\end{array}\right.$

$= \dfrac{1}{\eta_{tb}\eta Q_p(0.8M+0.2)}$ $\left\{\begin{array}{l}\text{power to heat the}\\ \text{plasma is not retained}\\ f_n=0.8\end{array}\right.$

η_L = laser system efficiency

η_{in} = injection efficiency

η_h = plasma (or pellet) heating efficiency

P_{in} = electric power input to injectors (or to laser)

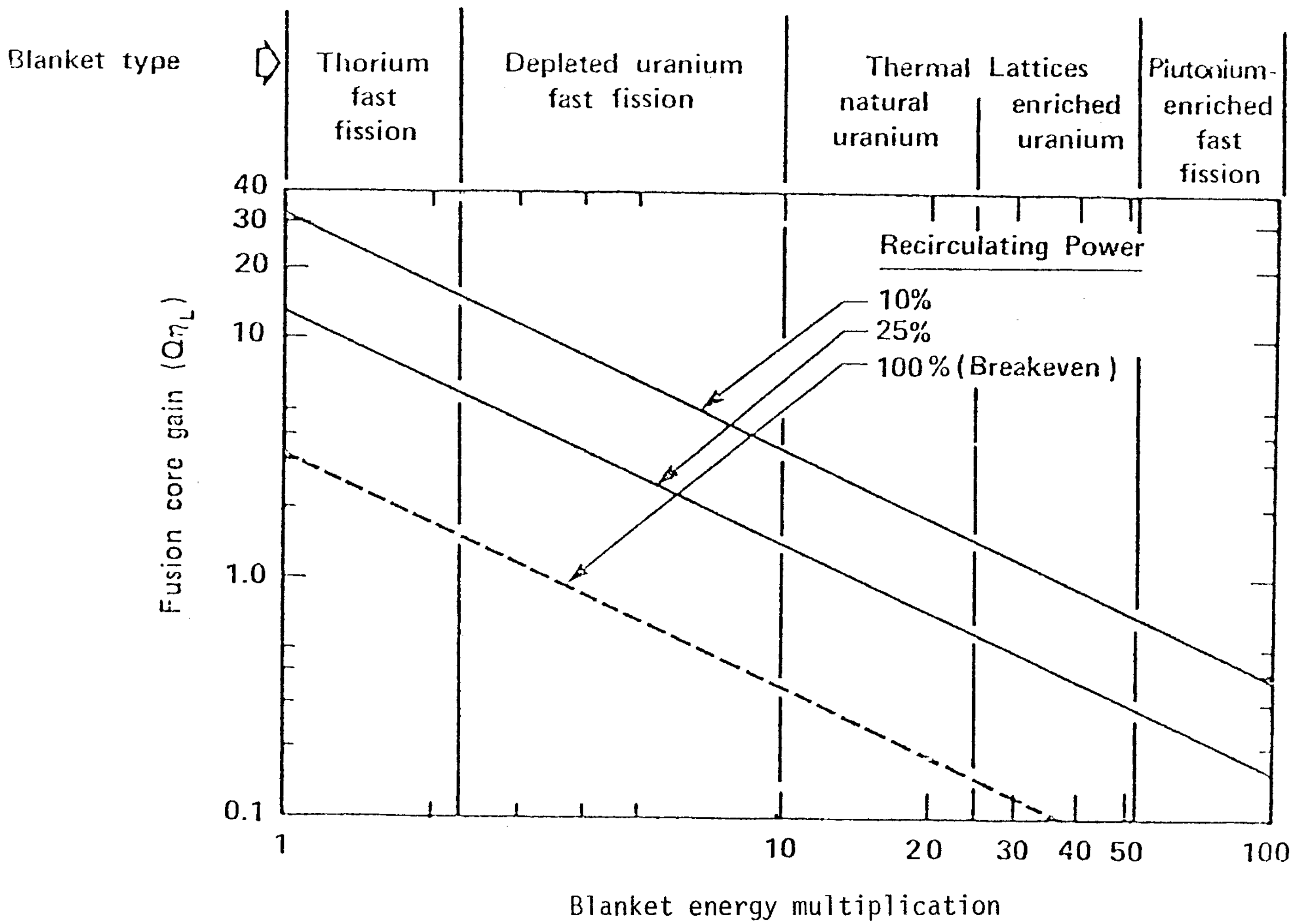
η_{tb} = thermal to electrical power conversion of the fusion-fission breeder $\simeq$ 0.35

f_n = fraction of energy carried by the D-T neutrons (~ 0.8)

Q_p = plasma power multiplication (or pellet gain)

ηQ_p = fusion gain = $\dfrac{\text{fusion power}}{\text{electric input to the fusion driver}}$

Fig. 8-a



Laser Fusion Core Gain Requirements for Hybrid Fusion/Fission Systems (Ref. 19).

Fig. 8-b

fuel needs from the hybrid) will be higher with Th/U fuel cycles than with U/Pu fuel cycles due to the higher conversion ratio, C, of the near breeder fission reactors ($C \sim 0.95$) compared to the burners ($C \sim 0.6$). Thus, with an internationally-controlled "secure fence" consisting of a fusion-fission fuel factory and a reprocessing plant, most of the power will be produced outside this fence and the integrated system will be a proliferation-controllable one.(33)

A final choice of whether the optimized fusion-fission system will be a power producer (on-line) or a fuel factory (no electricity is generated in the hybrid) and whether it is preferable to use U/Pu or Th/U fuel cycles should be based on an economic evaluation. Bender has studied these options in a mirror-system coupled to fission reactors through fissile fuel and power linkages.(34) He showed that working in an "on-line" mode (i.e., the hybrid produces fissile fuel + electricity) is more favorable economically than working in a fuel factory mode for both fuel cycles although the penalty obtained is less when the Th/U fuel cycle is used. Also, he demonstrated that the relative total system capital cost, R_c (defined as the capital cost of the combined system ($/kWe) relative to the capital cost of the fission reactors only), is insensitive to the hybrid capital cost even if the fusion power amplification, Q_p, is low (~ 0.5). With Th/U cycles and fixing R_c to be 1.25 (i.e., 25% increase in the electricity cost when the fusion and fission reactors are coupled), he demonstrated that the allowable relative unit capital cost K (defined as the ratio of the unit capital cost of the hybrid($/kWth) to the unit capital cost of the fission reactors) is higher for Th/U cycles than for U/Pu cycles; that is, more expensive Th/U-hybrids can be built without severely affecting the total electricity cost when compared to U/Pu-hybrid. Bender uses his economic model to compare different types of blankets that breed U-233.[35] He shows that system economics are dominated by the value of f/n, i.e. $R_c \sim 1/(f/n)$ while the support ratio, R_t, is dominated by the ratio (f/n)/M. These conclusions apply for $Q_p > 1$. In Bender's model, the electricity cost is considered to be dominated mainly by the capital cost.

VII. Conclusions and Remarks

It is demonstrated from the survey presented in this part that the fusion-fission hybrids can be designed to meet a broad spectrum of fissile fuel and energy requirements. Minimum extension in the present technology encountered in the LWRs and fast breeders is predicted to meet these requirements. Thus, an early introduction of the fusion-fission systems as a long term option is expected.

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Appendix A

Review of Different Fusion-Fission System Designs and Schematic Representation of Their Blankets

Table (A-1): Selective Designs of Fusion-Fission Systems Which Breed U-233

Authors or group	Lidsky	Blinkin & Novikov	LLL		Su & McCormick	Woodruff & Quimby Math. Sci. NW		Lidsky (MIT) Cooper (Physics Int.)	
Year of study	1969	1978	1978		1975	1976		1976	
Ref. # Blanket #	1,2 2	3 3	13 8-a	 8-b	6 13	12 9		26 15	
Type of machine	Tokamak R = 3.8 m r = 1.25 m toroidal field 2T, Ti 20 keV	Tokamak R = 11.4 m r = 5 m	Laser		Tokamak 20 m hybrid	Laser solenoid 330 m # of tube 4	 100 m 4	Electron beam heated linear solenoid 300 m length Mult. mirror isolated plas.	 Free stream gas blanket
Criteria for blanket design	U^{233} breeding for molten salt fission reactor (MSR) (symbiotic system) non-fissioning blk.	As in Lidsky but fission reactors breed tritium only	U^{233} breeding from Th^{233} metal + power Th-fast fission without U-multiplier	 Th-fast fission with U mult.	Power + high gain U^{233} breeding U+Pu fast fission multiplier Th blanket	Pu^{239} m conv. U^{233} B-Z (Low M)	Breed Pu^{239} U^{233} + some power (high M)	Breed U^{233} from molten salt Non-fissioning blanket	 Fissioning blk. with U^{238} front zone
Neutron spectrum in the blanket	Thermal	Thermal	Fast		Thermal + epithermal	Fast	Fast	Thermal	Fast + epithermal
Fuel	Molten salt ^7LiF-BeF_2+ThF_4 71%-2%-27%	Molten salt NaF-BeF_2-ThF_4 71%-2%-27%	Th^{233} metal	Th^{233} metal U-depleted in the front zone mult.	ThC to breed U^{233} in breeding zone U^{238}&8% Pu in front zone	Conv. U metal | B.Z. ThC	Conv. U + 4% Pu | B.Z. ThC	LiF-BeF_2-ThF_4 salt	
Structure	TZM(Mo)	Nb	s.s.	s.s.	Nb	Structure Nb		Nb	
Coolant	Li	LiF	Li (natural) + Na		Li in B.Z. Na in Front Z.	He Li	He Li	Li	Li
Mat. to breed tritium	Li	Na-F salt	Natural lithium		Li	Li	Li	Li (nat.)	
T(TER)	1.126	0.0	1.05	1.15	1.05	> 1.0	> 1.0	1.0	1.0
Fissile production	f/n=0.325	f/n=1.47	f/n=0.77 U^{233}=1.9 $\frac{kg}{MWt\text{-}yr}$	0.62(U^{233}) U^{233}=1.1 $\frac{kg}{MWt\text{-}yr}$ Pu=0.61 "	f/n=3.54(U^{233}) U^{233}=2417 kg/yr fuel doubling time= 12 yr	1.2(U +Pu) 3000 $\frac{kg}{yr}$	2(U +Pu) 1500 $\frac{kg}{yr}$	f/n=0.31 U^{233} 1176 kg/yr	f/n=0.92 (U^{233}) $\frac{kg}{yr}$ 5500 (U+P)
Energy (M) Multiplier	1.5	~ 1.6	1.77	2.53	80.9	7.0	23	1.01	4.25
P (MW_e)	130	92	Gross: 828 MWe, laser power=433 MWe, avg. power = 60 MWe, net = 385 MWe	1184 433 80 671 (net)	4000×η = 0.41 thermal	None	None	Cir: 309 net: 117	2448 -147
P (MW_{th})	295	208	2300, η_t =0.36	3290, η_t =0.36	10,000	4000	4000	Fiss. power < 10	3680 2076
Wall loading	1 MW/m^2	1 MW/m^2	2.35 $\frac{MW}{m^2}$	2.35 $\frac{MW}{m^2}$	0.5 MW/m^2	2.7 $\frac{MW}{m^2}$	2.7 $\frac{MW}{m^2}$	plasm. I/P 114 4 $\frac{MW}{m^2}$	4
Burnup						0.11% yr	0.42% yr		
Fusion power	~ 197 MWt Q = 0.57	130	Fusion gain ηQ=3.0 laser eff. = 3% pellet gain = 100					1080, fusion gain 3.5	866, 0.33
Fuel power density W/cm	Fiss. reactor power/fusion reactor power = 10	Fiss. reactor power/fusion reactor power = 10.8	fusion power: 1300 MW_{th}	 1300 MW_{th}	Av. 210 $\frac{kW}{\ell}$				

Table A-2: Performance Parameters of Fusion-Fission Systems Which Breed U-233

Type of Blanket	Authors or Lab & Year	Ref. #	Function	T	f/n	M	(n,fission)
Non-fissioning blanket (symbiotic)	Cook & Maniscalco (blk. #1) (1976)	34	• Th^{232} to breed U^{233} (no front zone multiplier)	1.07 1.05 1.06 1.12	0.55 0.33 0.233 0.14	1.6 1.3 1.3 1.1	0.029 0.015 0.009 0.004
	Lidsky (blk. #2) (1969)	1&2	• Molten salt ThF_4 to breed U^{233} (Mo multiplier	1.13	0.33	1.5	< 0.002
	Blinkin & Novikov (blanket #3) (1978)	3	• Molten salt ThF_4 to breed U^{233} (Nb multiplier)	0.0	1.47	1.6	< 0.002
	Lee	8	• Molten salt ThF_4 to breed U^{233} (Be multiplier)	~ 1.0	0.8	~ 1.6	
	Parish & Draper (1973) (blk. #4)	11	• Nb multiplier	1.15	0.31	0.8	
Th fast fission blanket without front zone multiplier	Maniscalco & Cook (Blk. #5) (1976)	34	Δ $Th^{232} \rightarrow U^{233}$ metal	1.0 1.17 1.05	0.84 0.72 0.5	2.3 2.3 2	0.097
	Maniscalco (blk. #6) (1975)	35	Δ $Th^{232} \rightarrow U^{233}$ metal	1.05	0.61	2.2	
	Maniscalco (LLL) (1978) (blk. #7) scoping design	13	Δ $Th^{232} \rightarrow U^{233}$ metal	1.15	0.443	2.03	
	Maniscalco (LLL) (1978) (blk. #8-a) final design	13	Δ $Th^{232}_{metal} \rightarrow U^{233}$	1.05	0.77	1.77	
U^{238}-Pu^{239} fast fission multiplier	Math. Sc. NW (blk. #9) (1976)	12	+ $ThC \rightarrow U^{233}$ front zone contains depleted U	> 1.0	1.2 (U^{233} + Pu^{239})	7	
	Math. Sc. NW (blk. #9) (1976)	12	⊕ $ThC \rightarrow U^{233}$ front zone includes 4% P_u of the fuel content	> 1.0	2 (U^{233} + Pu^{239})	23	
	Maniscalco (LLL) (1978) (blk. #10) scoping design	13	+ $Th^{232}_{metal} \rightarrow U^{233}$ depleted U front zone followed by Be zone	1.02	1.23 (U^{233})	4.88	
	Maniscalco (LLL) (1978) (blk. #8-b) final design	13	+ $Th^{232}_{metal} \rightarrow U^{233}$ depleted U front zone followed by Be zone	1.15	0.62 (U^{233})	2.53	
	Cook & Maniscalco (1976) (blk. #11)	34	+ $Th^{232}_{metal} \rightarrow U^{232}$ depleted U front zone followed by Li zone	1.12	U^{233} 0.763 Pu^{239} 0.622	6.2	0.019
	Cook & Maniscalco (1976) (blk. #12-a)	34	+ $Th^{232}_{metal} \rightarrow U^{232}$ depleted U front zone followed by Li zone	1.07	U^{233} 0.85 Pu^{239} 0.69	6.6	
	Cook & Maniscalco (1976) (blk. #12-b)	34	⊕ $Th^{232}_{metal} \rightarrow U^{232}$ depleted U with 2% Pu in the front zone	1.05	U^{233} 1.05 Pu^{239} 0.85	10	0.043
	Su & McCormick (blk. # 13)	6	⊕ $Th^{232} \rightarrow U^{232}$ $U^{238} \rightarrow Pu$	1.05	3.54	80.9	
Be front zone multiplier	Cook & Maniscalco (1976) (blk. #14)	34	Th^{232} metal $\rightarrow$ U^{233} Be front zone	1.13	0.71	2.2	0.076

⊕ Thermal blanket with U/Pu fast fissioning front zone

\+ Th fast fission blanket with U fast fission front zone

Δ Th fast fission with no front zone

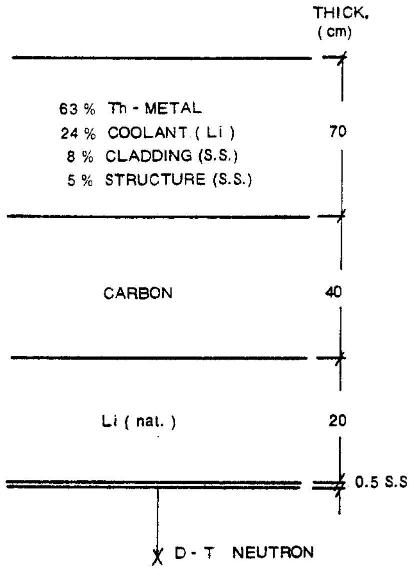
• Non-fissioning Th-blanket (symbiotic)

Table A-3: Selective Designs of Fusion-Fission Systems Which Breed Pu-239

Author or group	LLL/Bechtel	LLL/Westing-house	LLL/Bechtel	LLL/GA	LLL/GA		LLL/PNL	PNL	GE	Westing-house
Year of study	(1977/78)	(1977/78)	(1976)	(1977/78)	(1976)		(1974)	(1972)	(1978)	(1977)
Ref. # Blanket #	5,14-16 16	5,14,16,17 17	18,19 18	32,33 19	27-31 20		2,20,22 21	7, 23-25 22	36	37
Type of machine	2nd generation, laser driven (operates for 3 years) cost ~ 3 x LWR	Laser driven (operates for 2.5 years) cost ~ 2 x LWR	1st generation, laser driven (operates for 3.75 years)	Standard minimum B mirror (Ying-Yang) (operates for 3.8 years)	Standard minimum B mirror (Ying-Yang) Conductor field 8T mirror ratio: 2.5	12T 2.75	Mirror (Ying Yang)	Tokamak, 50 m, aspect ratio 5, T=10 keV $n\tau=3.5\times10^{13}/cm^{-3}sec$	Laser Driven	Tokamak
Criteria for Blanket Design	Breed Pu^{239} From Depleted Uranium Metal + Produce Power	Produces Power + Breed Pu^{239} From Spent LWR's Fuel	Breed Pu^{239} From Depleted Uranium Metal	Breed Pu^{239} From U_3Si (depleted) (blanket coverage is 86.5%)	Breed Pu^{239} From U-7% Mo Depleted-U Blanket Coverage 0.86	Breed U^{233} from Th^{232} metal Blanket Coverage 0.77	Produce electricity + breed Pu^{239}	- Produce Electricity	Breed Pu^{239} and use it directly in LWR without reprocessing	Breed Pu^{239} from U-238 and the breeding zone covers the outside region
Neutron spectrum in the blanket	Fast	Fast + epithermal	Fast	Fast	Fast	Fast	Thermal	Thermal	Fast	Fast
Fuel type	Depleted uranium metal	Spent fuel from LWR's in carbide from (UPu)-C	Depleted uranium metal	Depleted uranium in U_3Si	(U-Mo) 7%W-Mo	Th^{232} metal	Depleted UO_2 plates in converter UO_2 with 1.35% U^{235} (in fission lattice)	UC (nat. uranium) for front zone-U (nat.) metal for breeding zone	UO_2 (Depleted uranium)	$U_{nat}C$
Structure	316SS	316SS	316SS	Inconel 718	Inconel 718			Nb	SS	SS
Coolant	Na in fuel zone Li in top, bottom and radial blks.	Natural lithium	Na in fuel zone Li in top, bottom and radial blks.	Helium gas	He	He	He	He	Na	He
Material to breed tritium	Li (nat.)	Li(nat.)	Li(50% Li^6)	LiH	Li^6 aluminate	Li^6 aluminate	Li (nat.)	Li (nat.)	Li(50% Li^6)	--
T(TBR)	0.99→1.07 av.=1.03	Fresh 0.8; av. 0.98	1.1 (total)	0.97→1.37 av.=1.01	~ 1.14	~ 1.09	~ 1.1	1.06	1.1	--
Fissile production	1→0.84 kg/MWt-yr, av. 0.88 kg/MWt-yr, 3500 kg/yr, f/n=1.6	Pu(net)=fresh 1.15, av. 0.63 Kg/MWt-yr f/n=1.23	~ av. 1300 kg/yr f/n=1.17	f/n: 1.86→1.63, av. 1.74, Pu^{239} (net)=1980 kg/yr	Pu^{239} 2360 kg/yr, f/n= 1.55	U 2590 kg/yr, f/n = 0.54	f/n=1.33	f/n=2.6 ?	f/n =1.17, ~1300 kg/yr Pu^{239}	f/n=1.79 1800 kg/yr of Pu-239
Energy multiplication (M)	6→8.3 av. 7.15	Fresh: 6.6 av. 11	Av. 8.7	9.14→17.7 Av. 10.9	Av. 11.1	Av. 2.8	39.8 $k_{eff}=0.9$	~ 35	8.6	9.4
P(MWe)	Gross: 1520 net: 1195→1232 Av. ≅ 1210		Av. {Gross: 535 net: 400	Net: 525	1040	-40	Net=663.8 $\eta_t=39\%$ $\eta_{net}=32\%$	Gross=400 MWt $\eta_{th}=0.4$ net~335	Gross: 535 Net: 400	
$P(MW_{th})$	4000	1380 (3 units running)	1400	~ 3600 capacity factor=0.74	4220	3340	2045.4	1000	1400	2300
Wall load (MW/m^2)	2→1.3 Av. 1.65	10	1	1.9	1.3 duty factor 0.75	4.2 duty factor 0.73	~0.2 MW/m^2	0.05 MW/m^2	1 MW/m^2	1.55 MW/m^2
Burnup	~ 0.6% after 1.5 years	Fresh=1.1% Av. 5.8%	Av. 1.5%	~ 1.16%	1.0 Blanket Exposure 4.1 $MW\text{-}yr/m^2$	~ 0.5% 9.2				
Fusion Power(MW)	P_f=850→530 Av. 700 Recirculat. 22→19%, Av. ~ 20% fusion gain≅2	P_f=125, fusion gain ≥ 1, recirculation 25%	P_f=200, fusion gain 2, recirculation 25%	P_f = 402 Q = 0.63 $P_{injected}$=638	P_f=470 P_{inj}= 100 keV Q=0.68	1500 P_{inj}= 100 keV Q=0.75	64.2 MW_t P_{inj} 68.3 MW_e Q= 0.94	P_f =31.4 MWth P_{inj} = 65 MWth Q=0.48	P_f=200 Q=1.5	P_f= 122 MW Q=1.25
Power density W/cm^3	Av: 78.4→91.3 Av.≅84.9 Max: 1.89→220 Av. ≅ 204	Av: Fresh 170 Av. 330 Max: (2.5 yr)= 640	Av. ~ 16.8	In fuel zone: 193→34 Av. 270	150	110	4.3	0.75 W/cm^3 in fuel zone	Av. 16.8 W/cm^3	

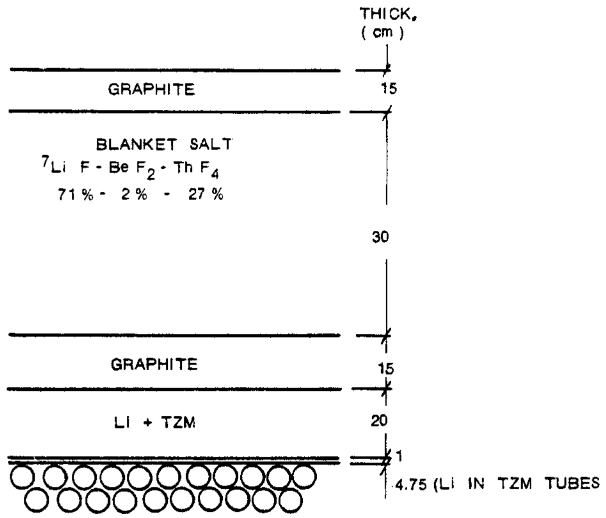
Table A-4: Performance Parameters of Fusion-Fission Systems Which Breed Pu-239

Type of Blanket	Authors or Lab. & Year	Ref. #	Function	T	f/n	M
U fast fission blanket	LLL/Bechtel (blk. #15) (1977)	5,14-16	Breed Pu^{239} from depleted uranium	1.03	1.61	7.15
	LLL/Westinghouse (blk. #17) (1977-1978)	5,14,16,17	Produce power + Pu^{239}	0.98	1.23	11
	LLL/Bechtel (blk. 18) (1976)	18,19	Breed Pu^{239} + power	1.1	1.17	8.7
	LLL/GA (blk. #19) (1978)	32,33	Breed Pu^{239} + power	1.01	1.71	10.9
	LLL/GA (blk. #20) (1976)	27-31	Breed Pu^{239} + power	1.14	1.55	11.1
	Maniscalco (blk. # 23-a) (1975)	35	Breed Pu^{239} + power	1.02 1.07 1.04 1.06 1.16	1.95 1.99 1.96 1.34 4.58	10 10.4 11.4 7.00 80.2
U fast fission U thermal blanket	LLL/PNL (blk. #21) (1974)	2,20-22	Produce power + Pu^{239}	1.1	0.33	39.8
	PNL (blk. #22) (1972)	7,23-25	Produce power + Pu^{239}	1.06	2.63	35
	Maniscalco (blk. # 23-b) (1975)	35	Produce power + Pu^{239}	1.06 1.03 1.16 1.00 1.18	1.04 1.05 0.94 1.03 1.05	25 23.6 25.1 18.7 35.4



(1) MANISCALCO'S Th - NON - FISSIONING BLANKET

M = 1.6
f/n 0.55
T = 1.07



(2) LIDSKY'S Th - NON - FISSIONING BLANKET

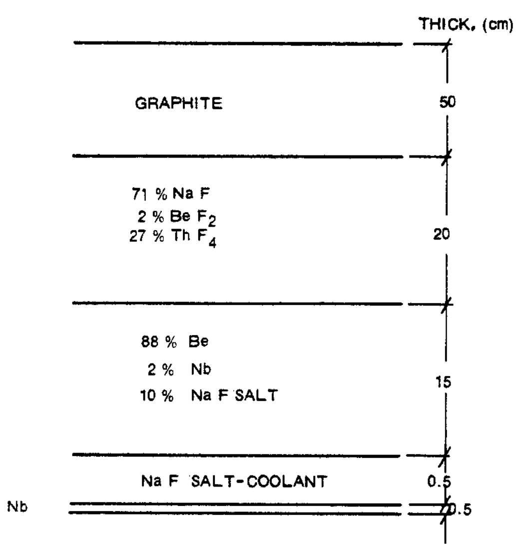
M = 1.5
f/n = 0.33
T = 1.13

CHARACTERISTIC OF THE MOLTEN SALT REACTORS (MSR)
COUPLED WITH THE FUSION REACTOR

Pe = 1000 MWe
Pt = 4450 MWth
C/R = 0.96

COMBINED SYSTEM:

Pe (net) 1690 MWe
η (net) 0.36 10 MSR, τ = 10 YR.

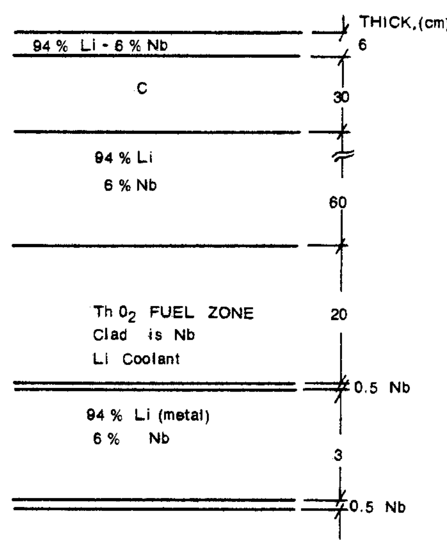


(3) BLINKIN + NOVIKOV'S
Th - NON - FISSIONING BLANKET

CHARACTERISTIC OF THE MOLTEN SALT (MSRT) REACTOR WHICH BREED TRITIUM AND COUPLED TO THE FUSION REACTOR.

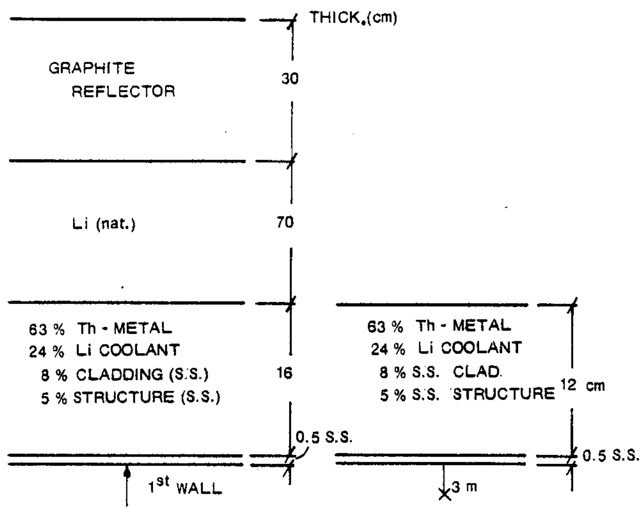
P_{th} = 2250 MW (th)
P_e = 1000 MW_e
C/R = 0.85 (TRITIUM)
COMPOSITION OF FUEL SALT 50 % LiF
50 % BeF
0.1 % $U^{233}F$

OF MSRT ≃ 11
τ ≅ 5 YR.
T 0.0
f/n 1.47
M 1.6



(4) PARISH + DRAPER'S Th - NON - FISSIONING BLANKET

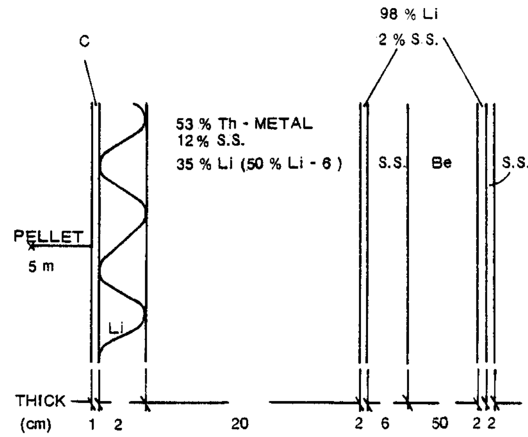
T = 1.15
f / n = 0.31
M = 0.8



(5) (6)

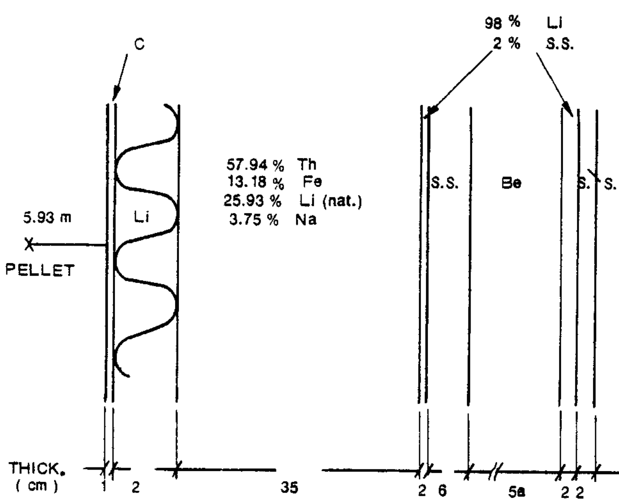
MANISCALCO'S Th - FAST FISSION BLANKETS

(5)	(6)
M = 2.3	M = 2.2
T = 1.0	T = 1.05
f / n = 0.84	f / n = 0.61



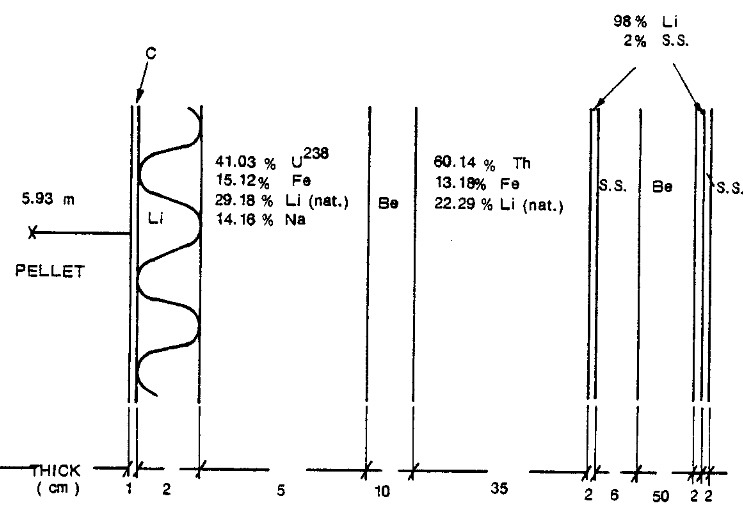
(7) LLL Th - FAST FISSION BLANKET

f/n = 0.443
T = 1.149
M = 2.03
U - 233 = 0.95 $\frac{\text{kg}}{\text{MWt - Yr}}$



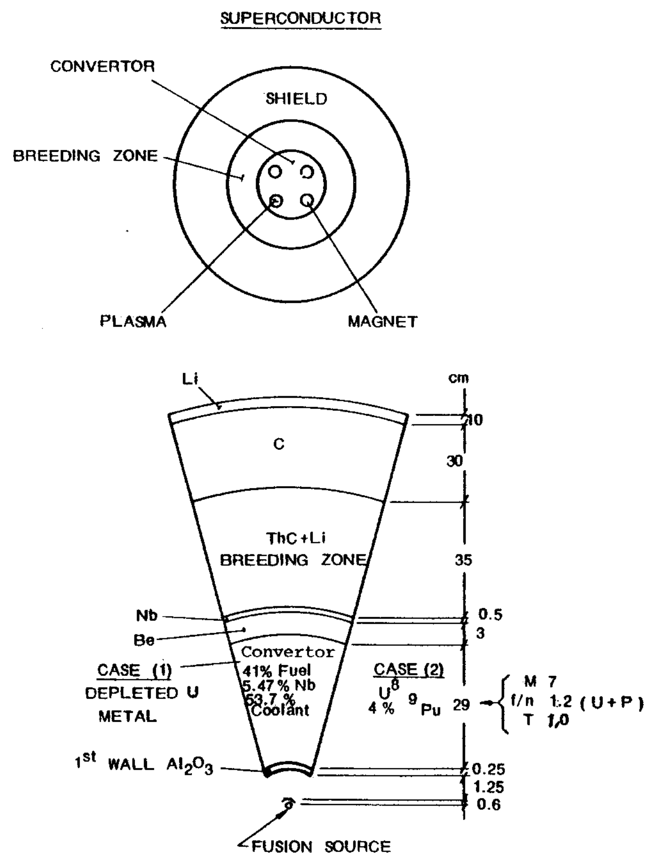
(8) - a LLL Th - Fast Fission BLANKET

M = 1.77
f/n = 0.77
T = 1.05

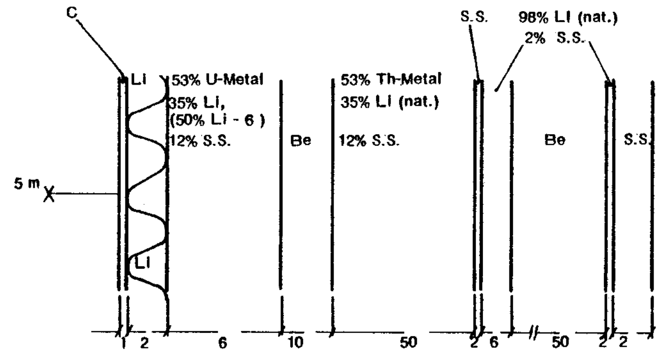


(8) - b LLL U - Th Fast Fission BLANKET WITH U FRONT ZONE MULTIPLIER

M = 2.53
f/n = 0.62 (U233)
T = 1.15



(9) MAT. SC. NORTH WEST INC.

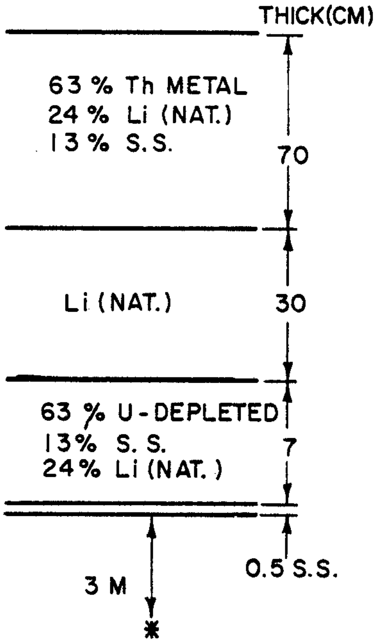


(10) LLL U FAST FISSION Th BLANKET
(U FRONT ZONE MULTIPLIER)

M = 4.88
$f/n\ (U^{233})$ = 1.23
T = 1.02

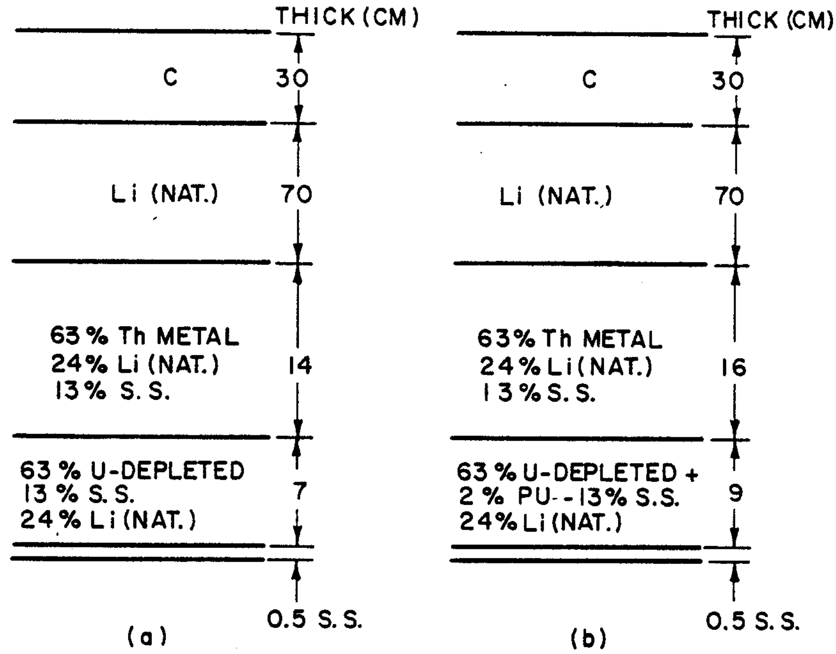
$U^{233} = 1.1 \frac{kg}{MW_t\text{-Yr.}}$

$Pu^{239} = 0.7 \frac{kg}{MW_t\text{-Yr}}$



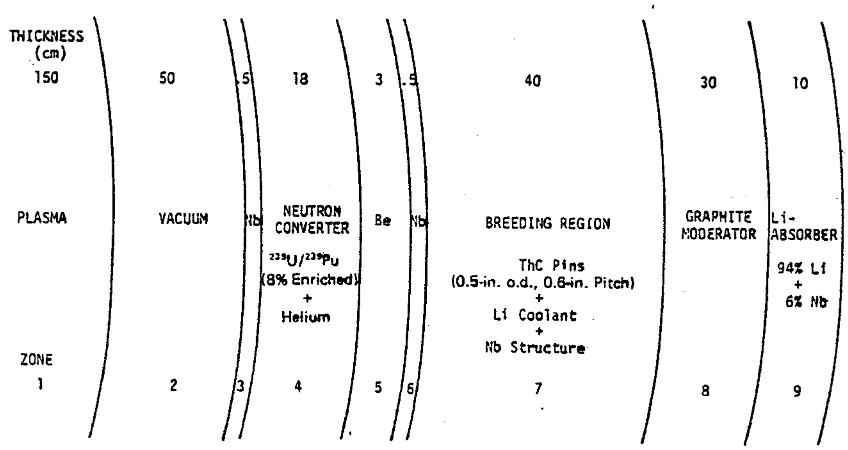
(11) MANISCALCO'S U-FAST FISSION Th-BLANKET

M = 6.2
T = 1.12
f/n = 0.763 (U-233)



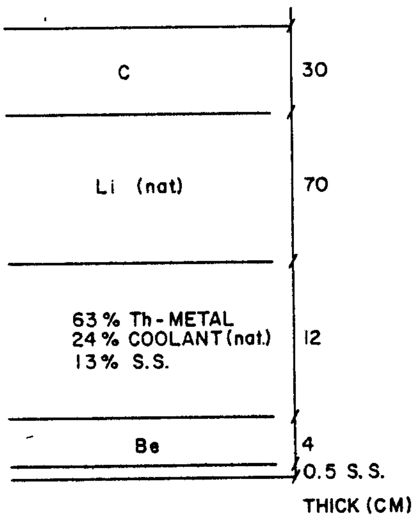
(12) MANISCALCO'S U-FAST FISSION Th-BLANKET

(a)	(b)
M = 6.6	M = 10
T = 1.07	T = 1.05
f/n = 0.85 (U-233)	f/n = 1.05



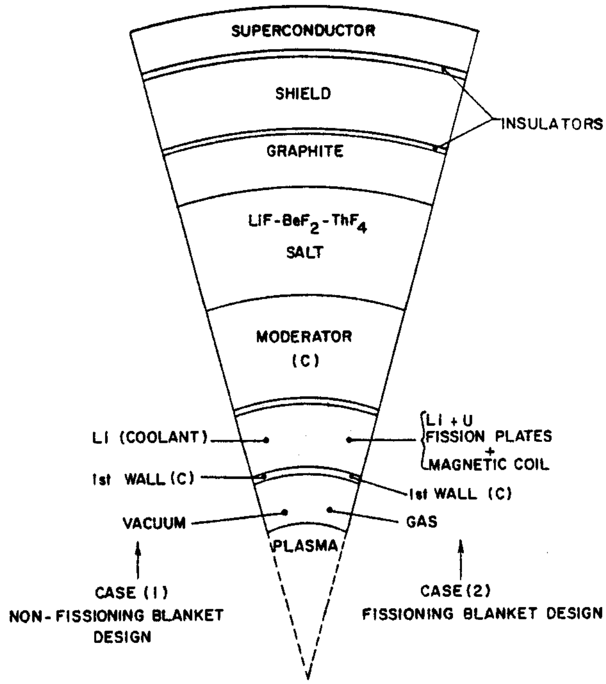
(13) Su & McCORMICK U/Pu FAST FISSION Th-BLANKET

M = 80.0
f/n = 3.54 (U^{233})
T = 1.05



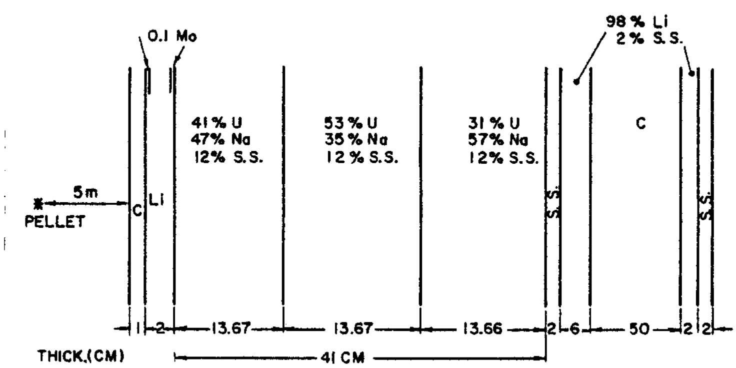
(14) MANISCALCO'S Be-FRONT ZONE Th-BLANKET

M = 2.2
T = 1.13
f/n = 0.71



(15) PIC + MIT BLANKET

Case (1)	Case (2)
M = 1.01	M = 4.25
f/n = 0.31	f/n = 0.92
T = 1.0	T = 1.0



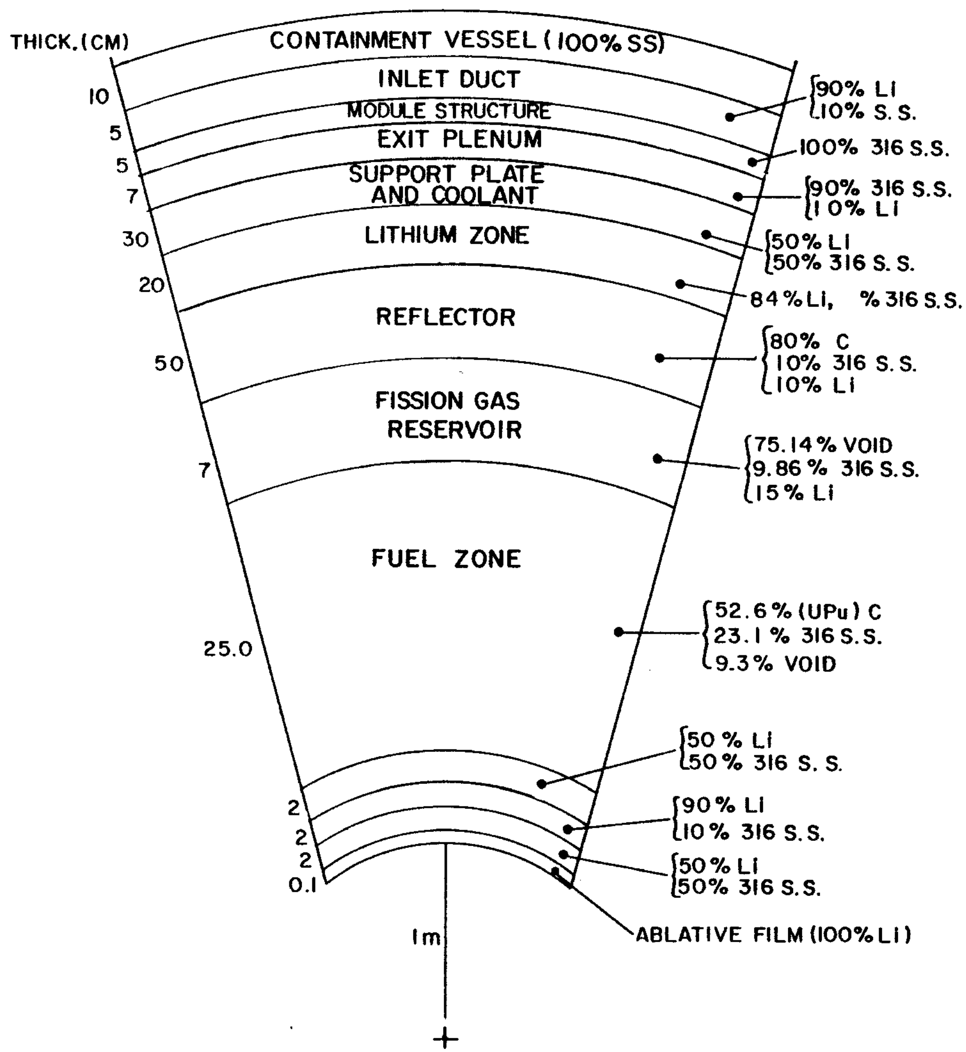
RADIAL BLANKET

(16) LLL/BECHTEL - 2nd DESIGN

M = 7.15
f/n = 1.61
T = 1.03

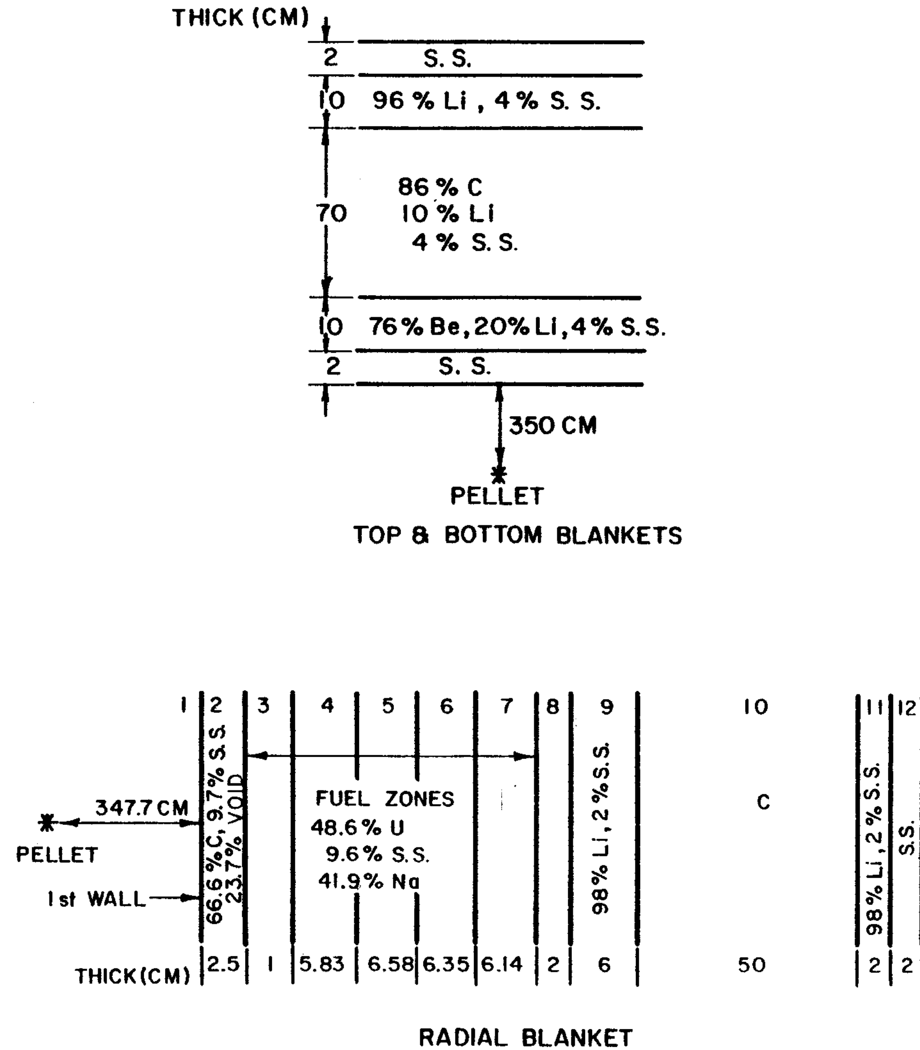
AVERAGE VALUES AFTER 3 YEARS

TOP AND BOTTOM BLANKETS AS THE 1st DESIGN
BUT LOCATED AT 3.5 M FROM THE LASER PELLET



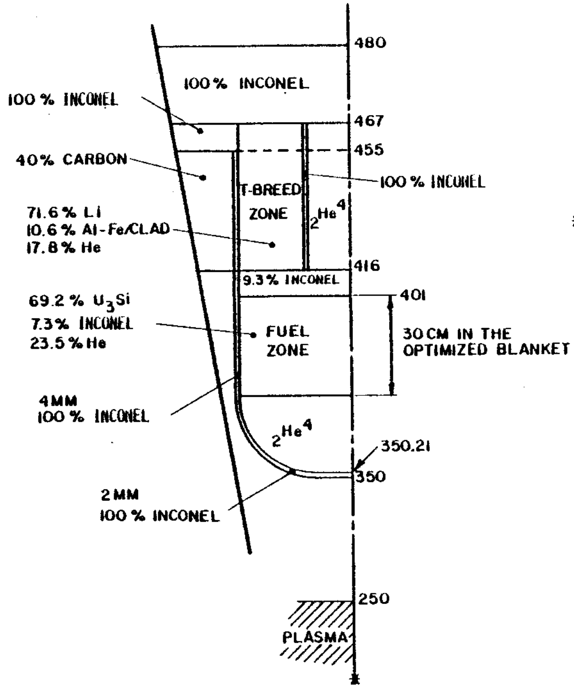
(17) LLL/WESTINGHOUSE (1977/1978)

M = 11
f/n = 1.45
T = 0.98



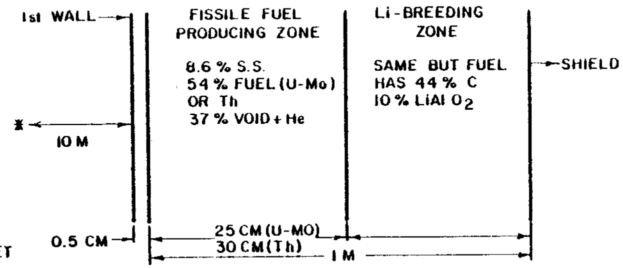
(18) LLL-LASER/BECHTEL, 1st DESIGN

M = 8.7
f/n = 1.46
T = 1.1



(19) LLL/GA (1977/78)

M = 10.9
f/n = 1.74
T = 1.01

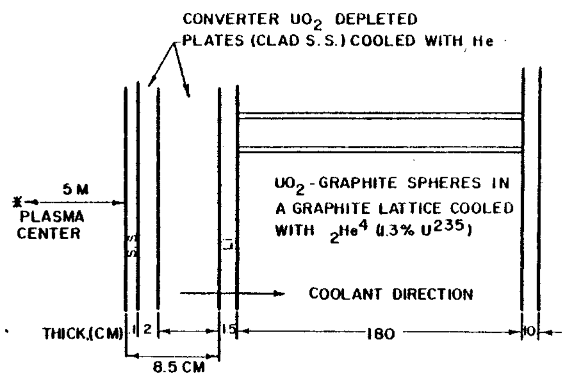


(20) LLL/GA (1976)

U-7% MO	Th
M = 11.1	2.8
f/n = 1.55	0.54
T = 1.14	1.09

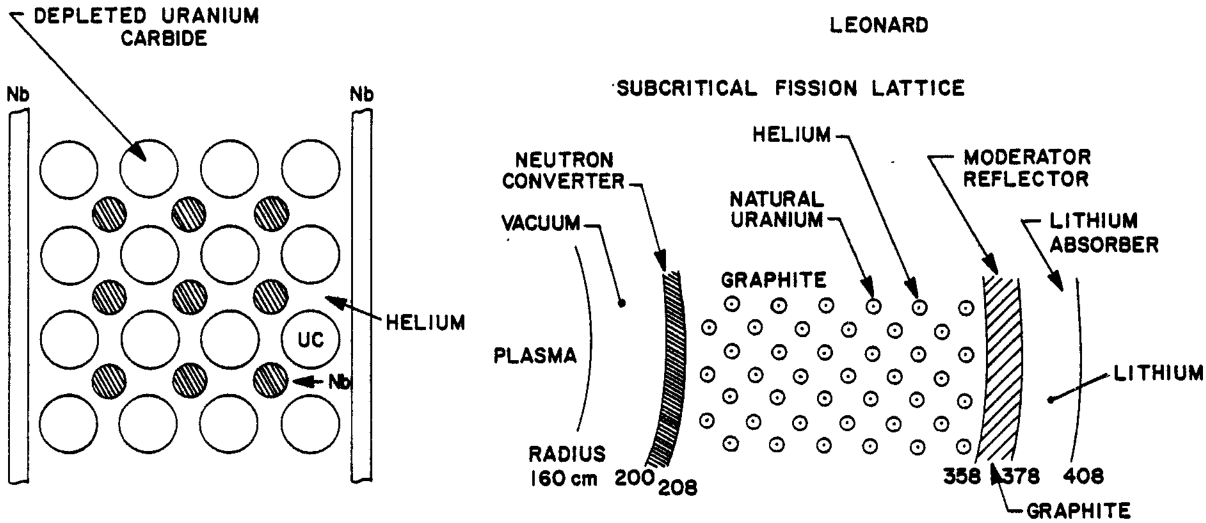
FOR Th BLANKET THE 54 % FUEL IS

45.9 % Th
5.4 % LiO_2 (Li-6)
2.7 % VOID



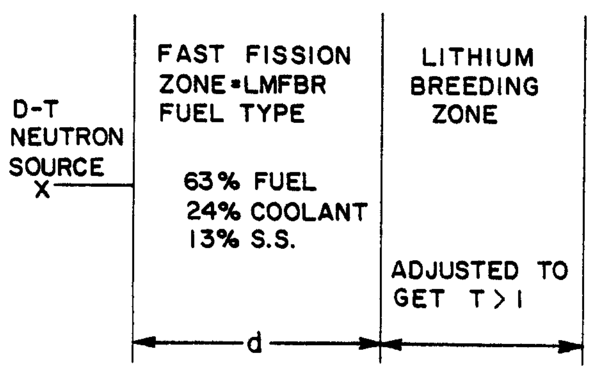
(21) LLL/PNL (1974)

M = 39.8
f/n = 1.1
T = 1.0



(22) PNL (LEONARD ET AL)
(1972)

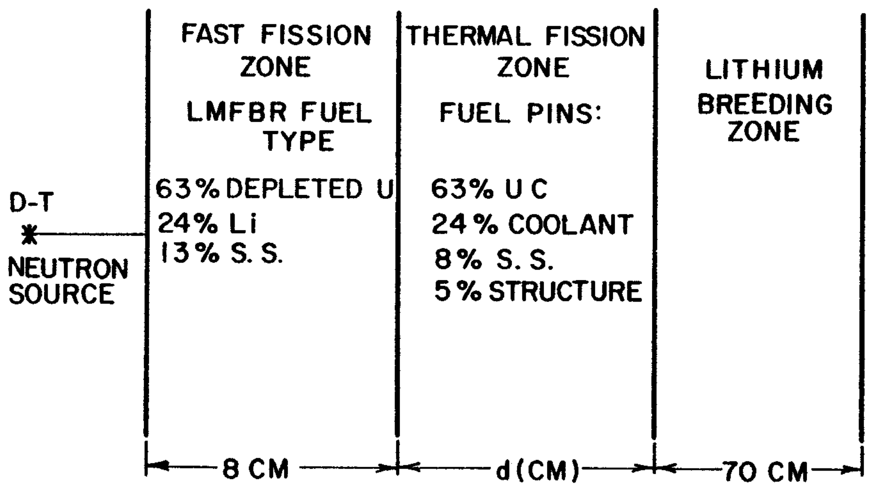
M = 35
f/n = 2.6
T = 1.06



DIFFERENT FAST FISSION BLANKETS

	A	B	C	D	E
FUEL TYPE	METAL			CARBIDE	
FERTILE FUEL	U - 238				
FISSILE FUEL	^{235}U (0.25%)		^{235}U(0.72)	^{235}U(0.25)	^{239}Pu(9.5)
COOLANT	Li	Li	He	Li	Na
THICK (cm)(d)	21	21	22	18	20/40
M	10	10.4	11.4	7.00	80.2
f / n	1.95	1.99	1.96	1.34	4.58
T	1.02	1.07	1.04	1.06	1.16

(23-a) MANISCALCO'S FAST FISSION BLANKETS



DIFFERENT THERMAL LATTICES IN FISSION ZONE

	A	B	C	D	E
U^{235} (% WT.)	NAT.	NAT.	NAT.	NAT.	1.0
COOLANT	He	Li-7	He	He	He
MODERATOR	ZrH	ZrH	Li^7H	ZrH	ZrH
STRUCTURE	←ZIRCALOY		→	S. S.	ZIRCALOY
THICK CM (d)	26	26	26	24	26
M	25	23.6	25.1	18.7	35.4
f/n	1.04	1.04	0.94	1.03	1.05
T	1.06	1.03	1.16	1.00	1.18

(23) b MANISCALCO'S THERMAL FISSION BLANKETS

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