A Survey of Fusion-Fission System Designs and Nuclear Analysis

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A SURVEY OF FUSION-FISSION SYSTEM

DESIGNS AND NUCLEAR ANALYSES

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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 out-
perform 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)

\[
G_A(\%) = \frac{10.95(C_{FB}-1)(1+\alpha)(1-C)P}{(1+Z)P + 0.383 \cdot RL + 10.95 \cdot D},
\]

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

\[
G_A(\%) = \frac{0.383 \cdot RBL(1+\alpha_B)}{PR_B \cdot f[1+Z_B^+ + \frac{R}{R} (1+Z)]} \cdot \left[(C_B - 1) - (1 - C) \cdot k \cdot P\right],
\]

where

\[
k = \frac{(1+\alpha)f}{(1+\alpha_B)} \approx 1
\]

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

$\alpha, \alpha_B =$ capture to fission ratio in the fission reactors and the fast breeder, respectively

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

$Z_B, Z = \frac{\text{out-of-core inventory}}{\text{in-core inventory}}$, in the fast breeder and the fission reactor, respectively
\[ R_B \cdot R = \text{fissile rating per fissile initial inventory} \left( \frac{\text{MW}_t}{\text{kg}} \right), \text{in the fast breeder and the fission reactors, respectively} \]

\[ L = \text{Load factor} \]

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

\[ D = \text{delay time in tritium processing (years)}. \]

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 \( \sim 11 \) 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 \( (C_{FB} \sim 1.4) \) 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. \( C_A, D = 0.0 \)

(b) Fast breeder-converter combination under steady-state conditions. \( G_A, D = 0.0 \)

**Fig. 1**

**Fig. 2**


(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\(^{(17)}\) \(K_{\text{eff}} \approx 0.94\).

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 \(^{238}\text{U}\) and \(^{232}\text{Th}\) as function of neutron energy. At 14.1 MeV, \(^{238}\text{U(n,fiss)} \approx 1.15\) barns while \(^{232}\text{Th(n,fiss)} \approx 0.37\) barns. Because of the 200 MeV released per fission,
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 $\text{U}^{238}$ is 4.5 and in $\text{Th}^{232}$ is 3.87. The resultant energy multiplication will reduce fusion technology requirements (e.g., lower the value of $n_T$, 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 $(n,\gamma)$ per incident D-T neutron (this entry is called the fissile fuel conversion ratio $C$) in an infinite medium containing natural uranium, $\text{U}^{238}$, and $\text{Th}^{232}$. An energy multiplication $M \approx 22$ can be obtained with a high fissile fuel conversion ratio ($C \approx 5$). Because of the lower fission cross section for $\text{Th}^{232}$, the capacity to breed U-233 is lower ($C \approx 2.7$).

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)

<table>
<thead>
<tr>
<th>Material</th>
<th>Energy Deposited (MeV)</th>
<th>M</th>
<th>Breeding Reaction $(n,\gamma) ; (f/n)$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural Uranium</td>
<td>309</td>
<td>22</td>
<td>5</td>
</tr>
<tr>
<td>Uranium-238</td>
<td>233</td>
<td>16</td>
<td>4.4</td>
</tr>
<tr>
<td>Thorium</td>
<td>64</td>
<td>4.5</td>
<td>2.7</td>
</tr>
</tbody>
</table>

Table 2
Blanket* vs. Infinite Medium and Fuel Type
(per 14.1 MeV D-T Neutron)
(Ref. 21, 31)

<table>
<thead>
<tr>
<th>Infinite Medium (natural uranium)</th>
<th>$T$</th>
<th>$f/n$</th>
<th>$(\text{MeV})$</th>
<th>M</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0.0</td>
<td>5.0</td>
<td>309</td>
<td>22</td>
</tr>
<tr>
<td>Blanket* With U</td>
<td>1.1</td>
<td>2.2</td>
<td>200</td>
<td>14</td>
</tr>
<tr>
<td>Blanket* With UO$_2$</td>
<td>1.1</td>
<td>1.1</td>
<td>100</td>
<td>7.1</td>
</tr>
<tr>
<td>Blanket* With UC</td>
<td>1.1</td>
<td>1.4</td>
<td>130</td>
<td>9.3</td>
</tr>
</tbody>
</table>

*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 $^6\text{Li}(n,\alpha)\text{T}$ and/or $^7\text{Li}(n,\alpha,n')\text{T}$ 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 $^6\text{Li}$ is performed in zones which have substantially-moderated neutrons where the $^6\text{Li}(n,\alpha)\text{T}$ 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.\textsuperscript{(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 \( \sim 2 \) 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 level throughout 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.\textsuperscript{(22)} Leonard has recently provided a bibliography of fusion-fission publications listing some 160 entries.\textsuperscript{(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-fissiioning 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. 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\textsuperscript{232} or U\textsuperscript{238}
3. Fissile isotopes: U\textsuperscript{233}, U\textsuperscript{235}, Pu\textsuperscript{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 \((n,\gamma)\) reactions and \(^6\text{Li}(n,\alpha)\text{T}\) reactions at low neutron energies, the amount of \(^6\text{Li}\) 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 \(^6\text{Li}(n,\alpha)\text{T}\) 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

Blanket Type: (Symbiosis) (Hybrid) First Wall

Zones:

Zone A: Multiplying Medium Front Zone
- $^9\text{Be}(n,2n)$
- and/or $^\text{Pb}(n,2n)$
- and/or $^\text{Mo}(n,2n)$
- and/or $^\text{Nb}(n,2n)$
- and/or $^\text{Li}(n,n',\alpha)T$
- Fast Fission Plate
  - $^\text{U}^{238}+^\text{Pu}^{239}$
  - $^\text{U}^{238}(n,\text{fiss}),(n,2n) (n,3n)$
  - $^\text{Pu}^{239}(n,\text{fiss}),(n,2n) (n,3n)$
  - $^\text{U}^{238}(n,\gamma)$

Zone B: Intermediate Moderator Zone
- Excess Neutrons
- Excess Neutrons

Zone C: Breeding and/or Energy Production Zone
- $^\text{Li}(n,\alpha)T$
- $^\text{Th}^{232}(n,\gamma) U^{233}$
- (Zone B may be graphite)
- Thermal Lattice to Produce Power
- Uranium Fast Fission
  - $^\text{U}^{238}(n,\text{fiss})$
  - $^\text{U}^{238}(n,\gamma)$
  - Produces $^\text{Pu}^{239}$
- Thorium Fast Fission
  - $^\text{Th}^{232}(n,\text{fiss})$
  - $^\text{Th}^{232}(n,\gamma)$
  - Produces $^\text{U}^{233}$
- Tritium Production
  - $^\text{Li}(n,\alpha)T$
  - $^\text{Li}(n,n',\alpha)T$
- Lithium to Produce Excess Tritium

Zone D: Moderator and Tritum Breeding Zone
- (U$^{233}$ to HTGR and/or Tritium)
- (Pu$^{239}$ and/or U$^{233}$ and/or Tritium)
Fig. 6

PERFORMANCE 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: MATHER SC. NORTHWEST LAB.
L/C: LIOSKY & COOPER

$M_{\text{E}}$, ENERGY MULTIPLICATION

TRITIUM BREEDING RATIO ($T$)

$\frac{f}{n}$, U-233 ATOMS/D-T NEUTRON
of the total breeding capture \((C_{FB} = f/n + T) \approx 1.2 - 1.8\) while \(M \approx 1 - 3\).

- 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 \(f/n \approx 4\) and \(M \approx 80\) 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 $\text{Pu}^{239}$ PRODUCING BLANKETS.

- (LLL/PNL) (1.1)
- (M) (1.18)
- (M) (1.16) (1.06)
- (M) (1.03)
- (M) — AUTHOR OR LAB.
- (1.00) — TRITIUM BREEDING RATIO

$M$, ENERGY MULTIPLICATION

$f/n$, $\text{Pu}$ ATOMS PRODUCED PER D-T NEUTRON

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 \( \sim 100\% \)), 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 \( \leq 25\% \)) 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 (\( \sim 0.1 \) 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>
<thead>
<tr>
<th>Cycle</th>
<th>Type of Blanket</th>
<th>Representation of the Blanket</th>
<th>M</th>
<th>f/n</th>
<th>C_{FB}=f/n+T</th>
</tr>
</thead>
<tbody>
<tr>
<td>Th^{232}/U^{233}</td>
<td>Non-Fissioning</td>
<td></td>
<td>Li</td>
<td>Th</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Be Front Zone</td>
<td></td>
<td>Be+Li+Th</td>
<td></td>
<td>~ 1.6</td>
</tr>
<tr>
<td></td>
<td>Th-Fast Fission</td>
<td></td>
<td>Th</td>
<td>Li</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U-Fast Fission</td>
<td></td>
<td>U</td>
<td>Th+Li</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U/Pu-Fast Fission</td>
<td></td>
<td>U+Pu</td>
<td>Th+Li</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U/Pu-Fast Fission (Thermal Th-Blanket)</td>
<td></td>
<td>U+Pu</td>
<td>Th+U^{233} +Li</td>
<td></td>
</tr>
<tr>
<td>U^{238}/Pu^{239}</td>
<td>U-Fast Fission</td>
<td></td>
<td>U</td>
<td>Li</td>
<td></td>
</tr>
<tr>
<td></td>
<td>U/Pu-Thermal Blankets</td>
<td></td>
<td>U+Pu</td>
<td>U+Li</td>
<td></td>
</tr>
</tbody>
</table>
Circulating power = \frac{P_{in}}{P_{eb}} = \frac{1}{n_{tb}(1+Q_{p}(0.8M+0.2))}
\begin{align*}
\text{fraction} & = \frac{1}{n_{tb}n_{Q_{p}}(0.8M+0.2)} \\
\text{power to heat the} & \\
\text{plasma is retained,} & \\
\text{if } f_{n} = 0.8 & \\
\text{power to heat the} & \\
\text{plasma is not retained} & \\
\text{if } f_{n} = 0.8 & \\
\end{align*}

\eta_{L} = \text{laser system efficiency} \\
\eta_{in} = \text{injection efficiency} \\
\eta_{h} = \text{plasma (or pellet) heating efficiency} \\
P_{in} = \text{electric power input to injectors (or to laser)} \\
n_{tb} = \text{thermal to electrical power conversion of the fusion-fission breeder} \approx 0.35 \\
f_{n} = \text{fraction of energy carried by the D-T neutrons} (~ 0.8) \\
Q_{p} = \text{plasma power multiplication (or pellet gain)} \\
nQ_{p} = \text{fusion gain} = \frac{\text{fusion power}}{\text{electric input to the fusion driver}} 

\text{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 ~ 0.95) compared to the burners (C ~ 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, \(R_s\), 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.
References:


Appendix A

Review of Different Fusion-Fission System Designs and Schematic Representation of Their Blankets
<table>
<thead>
<tr>
<th>Authors or group</th>
<th>Lidsky</th>
<th>Blumkin &amp; Novikov</th>
<th>LL</th>
<th>Sa &amp; McCormick</th>
<th>Woodruft &amp; Quinby Math, Sci. Mv</th>
<th>Lidsky (MIT) Cooper (Physicint)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ref. Blanket #</td>
<td>2</td>
<td>3</td>
<td>5</td>
<td>6</td>
<td>12</td>
<td>9</td>
</tr>
<tr>
<td>Type of machine</td>
<td>Tokamak</td>
<td>Tokamak</td>
<td>Laser</td>
<td>Tokamak</td>
<td>Laser solenoid</td>
<td>Electron beam heated linear</td>
</tr>
<tr>
<td></td>
<td>R = 3.0 m, r = 1.25 m, toroidal field 25.11, 20 keV</td>
<td>R = 11.4 m, r = 5 m</td>
<td></td>
<td>20 m hybrid</td>
<td>300 m # of tube</td>
<td>solenoid 300 m length</td>
</tr>
<tr>
<td>Criteria for blanket design</td>
<td>As in Lidsky but fusion reactors breed tritium only</td>
<td>U²³³ breeding from Th²³³ metal + power</td>
<td>Power + high gases U²³³ breeding</td>
<td>Py³⁹³ + some power(high M)</td>
<td>Breed U²³³ from molten salt</td>
<td>Free stream gas blanket</td>
</tr>
<tr>
<td>Neutron spectrum in the blanket</td>
<td>Thermal</td>
<td>Thermal</td>
<td>Fast</td>
<td>Thermal + epithermal</td>
<td>Fast</td>
<td>Fast</td>
</tr>
<tr>
<td>Fuel</td>
<td>Molten salt /Li²²²F = THF₄ / 71S²²²F-27S</td>
<td>Molten salt /Li²²²F = THF₄ / 71S²²²F-27S</td>
<td>Th²³³ metal</td>
<td>Th²³³ metal</td>
<td>U²³³ depleted in the front zone multi.</td>
<td>Th²³³ in breeding zone (U²³³ in front zone)</td>
</tr>
<tr>
<td>Structure</td>
<td>Li or Nb</td>
<td>Li or Nb</td>
<td>Li or Nb</td>
<td>Li or Nb</td>
<td>Li or Nb</td>
<td>Li or Nb</td>
</tr>
<tr>
<td>Coolant</td>
<td>Li</td>
<td>LiF</td>
<td>Li (natural) + Na</td>
<td>Li in BZ</td>
<td>Na in front 2</td>
<td>Na</td>
</tr>
<tr>
<td>Mt. to breed Tritium</td>
<td>Li</td>
<td>NaF salt</td>
<td>Natural lithium</td>
<td>Li</td>
<td>Li</td>
<td>Li (nat.)</td>
</tr>
<tr>
<td>T[°K]</td>
<td>1.126</td>
<td>0.0</td>
<td>1.05</td>
<td>1.15</td>
<td>1.05</td>
<td>1.0</td>
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<tr>
<td>Rupture</td>
<td>Fr=0.225</td>
<td>f/m=1.47</td>
<td>f/m=0.17</td>
<td>2.25</td>
<td>1.1</td>
<td>0.52</td>
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<tr>
<td>Gas</td>
<td>1.5</td>
<td>1.5</td>
<td>1.77</td>
<td>2.53</td>
<td>80.9</td>
<td>7.0</td>
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<tr>
<td>p (Mₐ)</td>
<td>130</td>
<td>52</td>
<td>None</td>
<td>None</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>P (Mₐ, A)</td>
<td>295</td>
<td>298</td>
<td>2300, 0.06</td>
<td>2000, 0.06</td>
<td>10000</td>
<td>4000</td>
</tr>
<tr>
<td>Wall thick</td>
<td>1 Mₐ/m²</td>
<td>1 Mₐ/m²</td>
<td>2.38</td>
<td>2.38</td>
<td>0.5 Mₐ/m²</td>
<td>2.7</td>
</tr>
<tr>
<td>Burnup</td>
<td>0.115 yr</td>
<td>0.626 yr</td>
<td>0.115 yr</td>
<td>0.626 yr</td>
<td>1000</td>
<td>0.5</td>
</tr>
<tr>
<td>Fusion power</td>
<td>105 Mₐ, 3.5%</td>
<td>130</td>
<td>Fusion gain n²=3.0</td>
<td>laser off = 30</td>
<td>100</td>
<td>0.33</td>
</tr>
</tbody>
</table>
| Fuel power | 130 Mₐ, 3.5% | 130 | Fusion power: | Fiss. reactor power: | 130 Mₐ, 3.5% | 130 Mₐ, 3.5% | 210 Mₐ, 3.5% | 32
<table>
<thead>
<tr>
<th>Type of Blanket</th>
<th>Authors or Lab &amp; Year</th>
<th>Ref. #</th>
<th>Function</th>
<th>T</th>
<th>f/n</th>
<th>M</th>
<th>(n, fission)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non-fissioning blanket</td>
<td>Cook &amp; Maniscalco (blk. #1) (1976)</td>
<td>34</td>
<td>Th(^{232}) to breed U(^{233}) (no front zone multiplier)</td>
<td>1.07</td>
<td>0.55</td>
<td>1.6</td>
<td>0.029</td>
</tr>
<tr>
<td>(symbiotic)</td>
<td>Lidsky (blk. #2) (1969)</td>
<td>132</td>
<td>Th(^{232}) to breed U(^{233}) (no front zone multiplier)</td>
<td>1.05</td>
<td>0.33</td>
<td>1.3</td>
<td>0.015</td>
</tr>
<tr>
<td></td>
<td>Blinkin &amp; Novikov (blanket #3) (1976)</td>
<td>3</td>
<td>Th(^{232}) to breed U(^{233}) (Nb multiplier)</td>
<td>1.08</td>
<td>0.233</td>
<td>1.3</td>
<td>0.009</td>
</tr>
<tr>
<td></td>
<td>Lee</td>
<td>8</td>
<td>Th(^{232}) to breed U(^{233}) (Be multiplier)</td>
<td>1.12</td>
<td>0.14</td>
<td>1.1</td>
<td>0.004</td>
</tr>
<tr>
<td></td>
<td>Parish &amp; Draper (1973) (blk. #4)</td>
<td>11</td>
<td>Nb multiplier</td>
<td>1.13</td>
<td>0.33</td>
<td>1.5</td>
<td>&lt; 0.002</td>
</tr>
<tr>
<td></td>
<td>Maniscalco &amp; Cook (Blk. #5) (1976)</td>
<td>34</td>
<td>(\Delta ) Th(^{232})+U(^{233}) metal</td>
<td>1.05</td>
<td>0.84</td>
<td>2.3</td>
<td>0.097</td>
</tr>
<tr>
<td>still fast fission blanket without</td>
<td>Maniscalco (blk. #6) (1975)</td>
<td>35</td>
<td>(\Delta ) Th(^{232})-U(^{233}) metal</td>
<td>1.05</td>
<td>0.84</td>
<td>2.3</td>
<td>0.097</td>
</tr>
<tr>
<td>front zone multiplier</td>
<td>Maniscalco (LLL) (1978) (blk. #7)</td>
<td>13</td>
<td>(\Delta ) Th(^{232})-U(^{233}) metal</td>
<td>1.05</td>
<td>0.84</td>
<td>2.3</td>
<td>0.097</td>
</tr>
<tr>
<td>scoping design</td>
<td>Maniscalco (LLL) (1978) (blk. #8-b) final design</td>
<td>13</td>
<td>(\Delta ) Th(^{232})-U(^{233}) metal</td>
<td>1.05</td>
<td>0.84</td>
<td>2.3</td>
<td>0.097</td>
</tr>
<tr>
<td></td>
<td>Math. Sc. NW (blk. #9) (1976)</td>
<td>12</td>
<td>Th(^{233}) front zone contains doped U</td>
<td>&gt; 1.0</td>
<td>1.2 (U(^{233})+Pu(^{239}))</td>
<td>7</td>
<td></td>
</tr>
<tr>
<td>fast fission multiplier</td>
<td>Math. Sc. NW (blk. #9) (1976)</td>
<td>12</td>
<td>Th(^{233}) front zone includes 4% Pu of the fuel content</td>
<td>&gt; 1.0</td>
<td>2 (U(^{233})+Pu(^{239}))</td>
<td>23</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Maniscalco (LLL) (1978) (blk. #10) scoping design</td>
<td>13</td>
<td>Th(^{232})-U(^{233}) metal doped U front zone followed by Be zone</td>
<td>1.02</td>
<td>1.23</td>
<td>4.88</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Maniscalco (LLL) (1978) (blk. #8-b) final design</td>
<td>13</td>
<td>Th(^{232})-U(^{233}) metal doped U front zone followed by Be zone</td>
<td>1.15</td>
<td>0.67</td>
<td>2.5</td>
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<tr>
<td></td>
<td>Cook &amp; Maniscalco (1976) (blk. #11)</td>
<td>34</td>
<td>Th(^{233}) metal doped U front zone followed by Li zone</td>
<td>1.12</td>
<td>0.763</td>
<td>6.2</td>
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<tr>
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<td>Cook &amp; Maniscalco (1976) (blk. #12-a)</td>
<td>34</td>
<td>Th(^{233}) metal doped U front zone followed by Li zone</td>
<td>1.07</td>
<td>0.85</td>
<td>6.6</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cook &amp; Maniscalco (1976) (blk. #12-b)</td>
<td>34</td>
<td>Th(^{233}) metal doped U with 2% Pu in the front zone</td>
<td>1.05</td>
<td>1.05</td>
<td>10</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Su &amp; McCormick (blk. # 13)</td>
<td>6</td>
<td>Th(^{232})-U(^{233}) Pu</td>
<td>1.05</td>
<td>3.54</td>
<td>80.9</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Cook &amp; Maniscalco (1976) (blk. #14)</td>
<td>34</td>
<td>Th(^{232}) metal- (\gamma) Pu</td>
<td>1.13</td>
<td>0.71</td>
<td>2.2</td>
<td>0.076</td>
</tr>
</tbody>
</table>

\(\gamma\) Thermal blanket with U/Pu fast fissioning front zone
+ Th fast fission blanket with U fast fission front zone
\(\Delta\) Th fast fission with no front zone
* Non-fissioning Th-blanket (symbiotic)
<table>
<thead>
<tr>
<th>Author or group</th>
<th>LBL/Bechtel</th>
<th>LBL/Hunting-House</th>
<th>LBL/Bechtel</th>
<th>LLI/GA</th>
<th>LLI/GA</th>
<th>LLI/FML</th>
<th>PNLL</th>
<th>GE</th>
<th>Feeding-bright</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ref. # Blanket #</td>
<td>5, 14-16</td>
<td>16</td>
<td>5, 14-16, 17</td>
<td>18</td>
<td>19</td>
<td>20</td>
<td>21</td>
<td>22</td>
<td>36</td>
</tr>
<tr>
<td>Type of machine</td>
<td>2nd generation, laser driven (operates for 3 years) cost - $3 x LHR</td>
<td>Laser driven (operates for 2.5 years) cost - $4 x LHR</td>
<td>1st generation, laser driven (operates for 3.75 years) cost - $5 x LHR</td>
<td>Standard minimum 8-mirror (Ying- Yang) (operates for 3.8 years) cost - $6 x LHR</td>
<td>Standard minimum 8-mirror (Ying- Yang) (operates for 4 years) cost - $7 x LHR</td>
<td>Mirror (Ying- Yang) (operates for 4.3 years) cost - $8 x LHR</td>
<td>Mirror (Ying- Yang) (operates for 4.5 years) cost - $9 x LHR</td>
<td>Mirror (Ying- Yang) (operates for 4.7 years) cost - $10 x LHR</td>
<td></td>
</tr>
<tr>
<td>Criteria for blanket design</td>
<td>Laser derived power at 80% above fuel cost</td>
<td>Laser derived power at 85% above fuel cost</td>
<td>Laser derived power at 90% above fuel cost</td>
<td>Laser derived power at 95% above fuel cost</td>
<td>Laser derived power at 100% above fuel cost</td>
<td>Laser derived power at 105% above fuel cost</td>
<td>Laser derived power at 110% above fuel cost</td>
<td>Laser derived power at 115% above fuel cost</td>
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<td>Neutron spectrum in blanket</td>
<td>Fast</td>
<td>Fast</td>
<td>Fast</td>
<td>Fast</td>
<td>Thermal</td>
<td>Fast</td>
<td>Thermal</td>
<td>Fast</td>
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<td>Fuel type</td>
<td>Depleted uranium metal</td>
<td>Spent fuel from LHR's in carboneous [UO2]</td>
<td>Depleted uranium metal</td>
<td>Depleted uranium metal</td>
<td>Depleted uranium metal</td>
<td>UO2</td>
<td>UO2</td>
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<td>Coolant</td>
<td>High in fuel zone Li in top, bottom and radial silks.</td>
<td>Natural lithium</td>
<td>Na in fuel zone Li in top, bottom and radial silks.</td>
<td>Sodium</td>
<td>Helium gas</td>
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<td>Material to breed tritium</td>
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<td>Li (nat.)</td>
<td>Li (nat.)</td>
<td>Li (nat.)</td>
<td>Li (nat.)</td>
<td>Li (nat.)</td>
<td>Li (nat.)</td>
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<td>TTR/B</td>
<td>0.39+1.07</td>
<td>0.39+1.07</td>
<td>0.39+1.07</td>
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<td>Fissile production</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
<td>1.89-20.68 kW/kg-yr ex. 0.05 kW/kg-yr</td>
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<td>Energy multiplication</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
<td>6.9-3.3</td>
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<td>(MeV)</td>
<td>3, 7, 15</td>
<td>3, 7, 15</td>
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<td>3, 7, 15</td>
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<td>3, 7, 15</td>
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<tr>
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<td>4000</td>
<td>4000</td>
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<td>Wall load (MW)</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
<td>4.1, 4.6</td>
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<td>Turbine</td>
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<td>0.65-1.0 percent</td>
<td>0.65-1.0 percent</td>
<td>0.65-1.0 percent</td>
<td>0.65-1.0 percent</td>
<td>0.65-1.0 percent</td>
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<td>0.65-1.0 percent</td>
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<td>Fusion Energy (MW)</td>
<td>100, 200, 300</td>
<td>100, 200, 300</td>
<td>100, 200, 300</td>
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<td>100, 200, 300</td>
<td>100, 200, 300</td>
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<td>Power density (W/cm²)</td>
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<td>1.9, 3.9, 6.9</td>
<td>1.9, 3.9, 6.9</td>
<td>1.9, 3.9, 6.9</td>
<td>1.9, 3.9, 6.9</td>
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<td>1.9, 3.9, 6.9</td>
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Notes: 1. Table A-3: Selective Designs of Fusion-Fission Systems Which Breed Pu-239.
<table>
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<tr>
<th>Type of Blanket</th>
<th>Authors or Lab. &amp; Year</th>
<th>Ref. #</th>
<th>Function</th>
<th>T</th>
<th>f/n</th>
<th>M</th>
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<tbody>
<tr>
<td>U fast fission</td>
<td>LLL/Bechtel (blk. #15)</td>
<td>5,14-16</td>
<td>Breed Pu$^{239}$ from depleted uranium</td>
<td>1.03</td>
<td>1.61</td>
<td>7.15</td>
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<td>blanket</td>
<td>(1977)</td>
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<td></td>
<td>LLL/Westinghouse (blk. #17) (1977-1978)</td>
<td>5,14,16,17</td>
<td>Produce power + Pu$^{239}$</td>
<td>0.98</td>
<td>1.23</td>
<td>11</td>
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<td></td>
<td>LLL/Bechtel (blk. 18) (1976)</td>
<td>18,19</td>
<td>Breed Pu$^{239}$ + power</td>
<td>1.1</td>
<td>1.17</td>
<td>8.7</td>
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<td></td>
<td>LLL/GA (blk. #19) (1978)</td>
<td>32,33</td>
<td>Breed Pu$^{239}$ + power</td>
<td>1.01</td>
<td>1.71</td>
<td>10.9</td>
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<td>LLL/GA (blk. #20) (1976)</td>
<td>27-31</td>
<td>Breed Pu$^{239}$ + power</td>
<td>1.14</td>
<td>1.55</td>
<td>11.1</td>
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<td>Maniscalco (blk. # 23-a) (1975)</td>
<td>35</td>
<td>Breed Pu$^{239}$ + power</td>
<td>1.02</td>
<td>1.95</td>
<td>10</td>
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<td></td>
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<td>1.07</td>
<td>1.99</td>
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<td>1.96</td>
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<td></td>
<td></td>
<td>1.06</td>
<td>1.34</td>
<td>7.00</td>
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<td></td>
<td></td>
<td>1.16</td>
<td>4.58</td>
<td>80.2</td>
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<tr>
<td>U fast fission</td>
<td>LLL/PNL (blk. #21) (1974)</td>
<td>2,20-22</td>
<td>Produce power + Pu$^{239}$</td>
<td>1.1</td>
<td>0.33</td>
<td>39.8</td>
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<td>U thermal blanket</td>
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<td></td>
<td>PNL (blk. #22) (1972)</td>
<td>7,23-25</td>
<td>Produce power + Pu$^{239}$</td>
<td>1.06</td>
<td>2.63</td>
<td>35</td>
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<tr>
<td></td>
<td>Maniscalco (blk. # 23-b) (1975)</td>
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<td>Produce power + Pu$^{239}$</td>
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<td>1.04</td>
<td>25</td>
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<tr>
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<td></td>
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<td>1.03</td>
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<td>1.16</td>
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<td>1.00</td>
<td>1.03</td>
<td>18.7</td>
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<td>1.18</td>
<td>1.05</td>
<td>35.4</td>
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</table>
(1) MANISCALCO'S Th - NON-FISSIONING BLANKET

\[ M = 1.5 \]
\[ t/n = 0.55 \]
\[ T = 1.07 \]

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

\[ M = 1.5 \]
\[ t/n = 0.33 \]
\[ T = 1.13 \]

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

\[ P_e = 1000 \text{ MWe} \]
\[ P_t = 4450 \text{ MWe} \]
\[ C/R = 0.36 \]

COMBINED SYSTEM:

\[ P_e (\text{net}) = 1690 \text{ MWe} \]
\[ t (\text{net}) = 0.36 \]
\[ 10 \text{ MSR}, T = 10 \text{ YR.} \]

(3) BLINKIN + NOVIKOVS' TH - NON-FISSIONING BLANKET

\[ \text{CHARACTERISTIC OF THE MOLTEN SALT (MSR) REACTOR WHICH BREEDS TRITIUM AND COUPLED TO THE FUSION REACTOR:} \]
\[ P_{th} = 2250 \text{ MW (th)} \]
\[ P_e = 1000 \text{ MWe} \]
\[ C/R = 0.85 \text{ (TRITIUM)} \]
\[ COMPOSITION OF FUEL SALT: 50\% \text{ LiF} \]
\[ 50\% \text{ BeF}_2 \]
\[ 0.1\% \text{ U}^{233}_{\text{F}} \]

\[ M_{\text{MSR}} = 11 \]
\[ T = 5 \text{ YR} \]
\[ t/n = 0.0 \]
\[ M = 1.6 \]
(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

M = 6.6
T = 1.07
f/n = 0.85 (U-233)

(13) SU & MCCORMICK U/Pu FAST FISSION Th-BLANKET

M = 80.0
f/n = 3.54 (U233)
T = 1.05
(14) MANISCALCO'S Be-FRONT ZONE Th-BLANKET

\[
\begin{align*}
M &= 2.2 \\
T &= 1.13 \\
f/n &= 0.71
\end{align*}
\]

M = 1.01
\[
\begin{align*}
T &= 1.0 \\
f/n &= 0.31
\end{align*}
\]

(16) LLL/BECHTEL - 2nd DESIGN

\[
\begin{align*}
M &= 7.15 \\
f/n &= 1.61 \\
T &= 1.03
\end{align*}
\]

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

M = 35
f/n = 2.6
T = 1.06

DIFFERENT FAST FISSION BLANKETS

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<th>B</th>
<th>C</th>
<th>D</th>
<th>E</th>
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<tbody>
<tr>
<td>FUEL TYPE</td>
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<td>METAL</td>
<td>CARBIDE</td>
<td>CARBIDE</td>
<td>CARBIDE</td>
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<tr>
<td>FERTILE FUEL</td>
<td>235 U</td>
<td>235 U</td>
<td>235 U</td>
<td>235 U</td>
<td>239 U</td>
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<td>FISSILE FUEL</td>
<td>U (0.25%)</td>
<td>U (0.25%)</td>
<td>U (0.72%)</td>
<td>U (0.72%)</td>
<td>Pu (9.5%)</td>
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<td>COOLANT</td>
<td>Li</td>
<td>Li</td>
<td>He</td>
<td>Li</td>
<td>Na</td>
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<tr>
<td>THICK (cm)(d)</td>
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<td>21</td>
<td>22</td>
<td>18</td>
<td>20/40</td>
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<tr>
<td>T</td>
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<td>1.07</td>
<td>1.04</td>
<td>1.06</td>
<td>1.16</td>
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(23-a) MANISCALCO'S FAST FISSION BLANKETS
### Fast Fission Zone
- LMFBR Fuel Type
  - 63% Depleted U
  - 24% Li
  - 13% S.S.

### Thermal Fission Zone
- Fuel Pins:
  - 63% UC
  - 24% Coolant
  - 8% S.S.
  - 5% Structure

### Lithium Breeding Zone

### Different Thermal Lattices in Fission Zone

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<td>U²³⁵</td>
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<td>Li-7</td>
<td>He</td>
<td>He</td>
<td>He</td>
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<td>Moderator</td>
<td>ZrH</td>
<td>ZrH</td>
<td>Li¹H</td>
<td>ZrH</td>
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<td>S.S.</td>
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<td>T</td>
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(23)b MANISCALCO'S THERMAL FISSION BLANKETS
References for Appendix A


26. Oliver, D., Cooper, R., Lidsky, L., "Breeding of Fissile Fuel With Linear Fusion Sources", Ref. 12 (1976), 775-786.


