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Lawrence Radiation Laboratory
UNIVERSITY OF CALIFORNIA
LIVERMORE

# NEUTRONICS OF SUB-CRITICAL FAST FISSION BLANKETS FOR D-T FUSION REACTORS

J. D. Lee

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ABSTRACT

Neutronics of Sub-Critical Fast Fission Blankets for D-T Fusion Reactors*. J. D. Lee, Lawrence Livermore Laboratory, Livermore, California.

The coupling of fusion and fission has the potential of significantly reducing plasma confinement criteria below that required for a pure fusion system while also reducing fissile fuel doubling times below those possible with pure fission systems.

To give a better feeling for fusion-fission's potential, and problems, a neutronic appraisal of the sub-critical fast fission blanket concept is presented. Results indicate energy increases over non-fission blankets of a factor of 10 to 20 (250 to 500 MeV per 14 MeV fusion neutron) are possible in blankets where tritium breeding, cooling, structure, and fissile fuel breeding requirements are considered.

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WHILE FUSION AND FISSION are generally considered strong competitors, they might also be attractive partners. As presently conceived D-T fusion reactors will be power poor but neutron rich. Fission breeder reactors are just the opposite, power rich but neutron poor. Together, fusion and fission might be attractive, while separately they might not.

The idea of fusion-fission goes back to the early 50's but interest has been generally cool and what results are available are generally inconclusive.

In 1953, Powell proposed a $U^{238}$ blanket and speculated that "200 to 400 MeV of fission energy could be generated per fusion neutron" (1)*. And in 1954, Imhoff et al, reported on a preliminary calculation that estimated energy generation of approximately 160 MeV from a $U^{238}$ blanket (2).

In 1960, Weale et al, reported on experimental measurements made on a 40" uranium metal cylindrical pile with a 14 MeV D-T neutron source in the center. Measured $U^{235}$ and $U^{238}$ fission rates were 1.18 and 0.28 per 14 MeV neutron and the $U^{238}$ $(n,\gamma)$ rate was 4.08. The authors postulated from this experiment that 15 cm and 30 cm thick uranium blankets could produce 150 MeV and 220 MeV respectively, per 14 MeV source neutron (3).

The first substantive neutronic analysis of a fissile blanket was reported in a 1965 thesis by Lontai, in which, he found a "practical and feasible" blanket containing ($LiF$-$BeF_2$-$UF_4$) to give approximately twice the energy of a non-fissile blanket while producing the required tritium (4).

At the 1969 Culham Conference on fusion reactor technology, Lidsky reported on his "Fission-Fusion Symbiosis" in which excess fusion induced neutrons are thermalized and used to breed fissile fuel which in turn is used to increase the effective breeding ratio of a separate fission reactor (5). In a specific example, a D-T fusion reactor with thorium in the blanket producing 0.325 $U^{233}$ atoms per fusion supplemented the breeding in a molten salt reactor having a breeding ratio of 0.96. Specifying a desired fuel doubling time of 5.0 years Lidsky calculated the required fusion to fission ratio to be 0.585. This ratio gives 342 MeV from the fission reactor per fusion reaction, which makes the total energy generation $\sim$ 360 MeV per D-T fusion.

*Numbers in parentheses designate References at end of paper.

There are two basic approaches to fusion-fission, the fission blanket in which fissile fuel is bred and "burned", and Lidsky's "Symbiosis" in which a fertile blanket supplements the breeding of a separate fission reactor.

While Lidsky's proposal is quite attractive, the sub-critical fission blanket has not been analyzed in sufficient detail to determine if it is a credible alternative to Lidsky's "Symbiosis" and what advantages it may have, if any.

NEUTRONIC APPRAISAL

To be neutronically viable, the sub-critical fast fission blanket must generate large amounts of energy while breeding sufficient tritium and fissile fuel for fuel doubling times of less than 10 years. The blanket must also be sub-critical.

A series of neutronic calculations were run to determine in a general way the potential of sub-critical fast fission blankets. Most neutronic calculations were performed using the "SORS Monte Carlo Transport Code" (6). The code used 66 group neutron cross-sections based on the LLL Howerton evaluated library. Most problems used 5 batches of 200 source neutrons (14 MeV) each.

The following are considered:

(1) Energy generation per 14 MeV fusion neutron.

(2) Tritium Breeding.

(3) Fission products effects.

(4) The effect different types of fissile/fertile fuel materials have on results.

(5) Limits imposed by criticality safety.

(6) Fission fuel doubling times.

(7) Heat transfer and structural requirements.

The first step was to consider large (effectively infinite) assemblies of pure thorium, uranium 238 and natural uranium. The resulting energy generation and fissile fuel breeding reactions per 14 MeV neutron were:

| Blanket Material | Energy(MeV) | Breeding Reactions |
|---|---|---|
| Thorium | 64 | 2.7 [$Th^{232}$ (n,γ)] |
| Uranium 238 | 233 | 4.4 [$U^{238}$ (n,γ)] |
| Uranium (natural) | 309 | 5.0 [$U^{238}$ (n,γ)] |

It appears the potential is here as predicted. It is also interesting to compare the above calculated results for uranium (natural) with Weale's experimental measurements on the 40" uranium metal cylindrical pile (3):

| | Calculated | Measured |
|---|---|---|
| Fission reactions | 1.50 | 1.46 |
| $U^{238}$ (n,γ) reactions | 4.96 | 4.08 |

The agreement is good since the calculational model is effectively an infinite assembly.

Now let's consider what happens when tritium breeding, cooling, and structure are considered by examining a conceptual blanket design containing uranium, niobium, and lithium Blanket geometry is a two zone spherical shell with an inner radius of 200 cm and an outer radius of 300 cm. The makeup of the blanket zones are:

| | Element | Volume Fraction |
|---|---|---|
| Zone 1 (30 cm) | Li | 0.95 |
| | Nb | 0.05 |
| Zone 2 (70 cm) | U | 0.65 |
| | Li | 0.30 |
| | Nb | 0.05 |

A number of different depleted Li and U isotopic ratios (but with blanket geometry and material volume fractions held constant) were tried to maximize energy generation while still breeding the required tritium. Near optimum results were found for Li depleted to 4% $Li^6$ (7.56% is natural) and U depleted to 0.04% $U^{235}$. These results per 14 MeV neutron were:

Energy Generation - 103 MeV
Tritium Breeding - 0.986
$U^{238}$ (n,γ) Reactions - 1.68

These results are interesting, and with 1.68 $Pu^{239}$ breeding reactions per D-T fusion it is obvious $Pu^{239}$ should replace the $U^{235}$ as the fissile fuel. A series of problems were run using the same blanket makeup to determine the effect of $Pu^{239}$ concentration. Results are listed in Table 1, cases 1-4.

The results indicate a fast fission blanket could be an excellent breeder. The 4% Pu (case 4) has a Pu breeding ratio of 3.1 while generating 430 MeV. (Pu breeding ratio is the $U^{238}$ (n,γ) reactions per $Pu^{239}$(n, f) reactions.) Note that fast fission of $U^{238}$ accounts for a large percentage of the fission reactions; 41% for the 4% Pu case.

CRITICALITY - As the Pu loading is increased the blanket energy generation increases because the assembly's neutron multiplication (K) increases. Since we want the blanket assembly to be sub-critical at all times, I will specify K max to be 0.90. As listed in Table 1 energy generation and K for the 4% Pu case was 430 MeV and 0.843 respectively and for the 5% Pu case it was 930 MeV and 0.934. The K calculations were performed using the "Tart" Monte Carlo Code (7). To estimate energy generation for this blanket with a K of 0.9 I will assume:

$$\text{Energy} = C \cdot \frac{1}{1-K}$$

The 4% and 5% cases gives values for the constant C of 67.6 and 61.3 respectively. So a similar blanket with K = 0.9 should have an energy generation between

$$E_{min} = 61.3 \cdot \frac{1.0}{1.0-0.9} = 613 \text{ MeV and}$$

$$E_{max} = 67.6 \cdot \frac{1.0}{1.0-0.9} = 676 \text{ MeV}$$

Therefore an energy multiplication of ≈ 30 times over D-T fusion with non-fission blankets appears possible in sub-critical fission blankets with $K \leq 0.9$.

FISSION PRODUCTS - The cases discussed thus far have neglected fission products. To see what effect fission product buildup has on blanket performance, I reran the 4% Pu case with 8% fission products (FP). This assumes individual fuel elements are removed with 8% burnup (∿ 80,000 MWD/T). While this burnup is high by today's standards it is below the upper design target of 100,000 MWD/T for fission breeders (8). The results are listed in the last row of Table 1, (case 6). The results are encouraging; the breeding ratio (3.0) decreased only 3% and the fraction of $U^{238}$ fission (0.44) increased 5%. There was also a 29% decrease in energy generation but this can be offset by slightly increasing the Pu loading.

$PU^{239}$ DOUBLING TIME - To be attractive, blankets consuming $Pu^{239}$ must have a $Pu^{239}$ doubling time less than the electric power demand doubling time. Electrical power requirements are doubling approximately every eight years so I will assume a five year fuel doubling time is necessary to replace old plants as well as meeting the new power demand. Using the results from the 4% Pu + 8% FP case (#6) discussed above the average blanket power density ($P_{ave}$)

required to give a 5 year Pu doubling time ($t_d$) is:

$$P_{ave} = 5.08 \times 10^{-21} \cdot \frac{E \cdot N(Pu)}{t_d \cdot [U(n,\gamma) - Pu(n,f)]}$$

where: $P_{ave}$ ≡ average power density (W/cc)

$E$ ≡ energy generation $\left(\frac{MeV}{fusion}\right)$

$N(Pu)$ ≡ Pu atom density $\left(\frac{atoms}{cc}\right)$,

$t_d$ ≡ Pu doubling time (year)

$[U(n,\gamma) - Pu(n,f)]$ ≡ Pu breeding excess $\left(\frac{atoms}{fusion}\right)$

$$P_{ave} = 5.08 \times 10^{-21} \cdot \frac{306}{5} \cdot \frac{1.20 \times 10^{21}}{1.66}$$

$$= 225 \text{ W/cc}$$

The wall loading (WL), flux of 14 MeV fusion neutrons on the first wall, associated with this average power is:

$$WL = P_{ave} \cdot Tb \cdot \frac{14}{E}$$

$$= 225 \cdot 70 \cdot \frac{14}{306} = 725 \text{ W/cm}^2 = 7.25 \text{ MW/M}^2$$

Where Tb = thickness of blanket uranium zone.

Both the power density (225 W/cc) and wall loading (7.25 $MW/M^2$) needed to get a five year doubling time in this example blanket should not pose an undue heat transfer problem and are therefore encouraging. Neutron heating of the first wall is not the only heating mechanism, photon and ions must also be considered, therefore each type of plasma containment will impose somewhat different wall loading limitations.

THORIUM FUEL - Since thorium is also a fertile element, the fissile $U^{233}$ isotope is the decay product of the $Th^{232}$ $(n,\gamma)$ reaction, fission blankets fueled with $U^{233}$ enriched thorium are also a possibility.

To compare thorium to uranium a few problems were run in which the uranium 238 and plutonium were replaced by thorium and uranium 233. All other blanket parameters were unchanged including the presence of 8% fission products. Results are listed in Table 2. Throium is clearly inferior to uranium for this type of blanket. The breeding ratio is only 1.3 compared to 3.0 and a fraction of Th fission is only 0.07 compared with 0.44 for $U^{238}$. The average blanket power required for a five year doubling time is:

$$P_{ave} = 5.08 \times 10^{21} \cdot \frac{478}{5} \cdot \frac{1.49 \times 10^{21}}{.69}$$

$$= 1049 \text{ W/cc}$$

compared with 225 W/cc for the uranium case.

MIXED OXIDE FUEL - The next step taken in this neutronic appraisal of the sub-critical fast fission blanket concept was to look at the effect of changing to a mixed oxide fuel. While the metal fuel should be neutronically superior the oxide appears to be the popular choice for the liquid metal fast breeder fission reactors presently being developed. So it is of interest to see how oxide fuel performs in fission blankets.

As in the thorium case, I will start by replacing the U + Pu metal fuel by the mixed oxide. All cases have 8% fission products approximating an average burn-up of 4% of initial heavy metal loading (U + Pu). As in the previous cases the basic blanket consists of a 30 cm inner zone containing 95% by volume (v/o) lithium coolant and 5 v/o niobium structure and a 70 cm outer zone containing the mixed oxide fuel. In all cases the lithium is depleted in $Li^6$, containing 4% $Li^6$ instead of the natural 7.56%.

Seven mixed oxide cases were run to get an idea of how Pu enrichment, structural volume fraction, coolant volume fraction, and in the last case how a decrease in Zone 1 thickness from 30 to 15 cm effects the performance. Table 3 gives Zone 2's Pu enrichment and material volume fractions for each blanket case while Table 4 lists the results.

As expected, the oxide is inferior to the metal. Comparing the oxide case #2 to its metal counterpart the breeding ratio has dropped from 3.0 to 1.6 and the $U^{238}$ fission fraction has dropped from 0.44 to 0.26.

A better comparison is doubling time. The average power density required for the case #2 oxide, compared with 225 W/cc for its metal counterpart to have a five year doubling time is:

$$P_{ave} = 5.08 \times 10^{-21} \cdot \frac{483}{5} \cdot \frac{1.30 \times 10^{21}}{1.06}$$

$$= 602 \text{ W/cc}$$

This does not mean the oxide is unacceptable, it only means that from a doubling time standpoint it is not as good as a metal system.

Table 5 lists, for each oxide case, the average power density required to give a five year Pu doubling time.

The average power density for cases 5-7 are most likely too high, therefore their doubling time must increase. Doubling time and power density are inversely proportioned.

CONCLUSIONS

The coupling of fusion and fission appears to be a plausible concept. In this study an energy increase of a factor of 10 to 20 (250 to 500 MeV per 14 MeV neutron) over non-fission blankets was achieved in sub-critical fast fission blankets where tritium breeding requirements were met and which in most cases were capable of having a five year fissile fuel doubling time.

Heat transfer and structural requirements will have a profound effect on blanket neutronics. Higher power density and fissile breeding ratios can lower fissile fuel doubling times. But higher power densities will increase coolant and structural fractions which in turn lowers breeding ratio. Obviously optimization is required to minimize doubling time for each fuel type.

Metal and oxide fuels are likely candidates but carbides and nitrides should also be considered because of their higher thermal conductivity and fuel densities, compared to the oxide.

Other questions such as criticality safety under accident conditions and economics of fusion-fission systems must also be dealt with.

There appears to be no clear cut neutronic advantage of the sub-critical fast fission blanket over a fertile blanket supplementing the breeding of a separate fission reactor. While using fusion neutrons to induce fast fission in $U^{238}$ is advantageous and the Pu is bred and burned insitu, the advantages might be out-weighed by the problems of cooling and fueling a high power fission blanket inside the high B magnets required to contain the D-T plasma. Also, short duration shut down of the fusion reactor would not interrupt power production of a separate critical fission reactor.

There are two major advantages to the fusion fission hybrid. First, it could supplement the fuel production of fission breeders in the event the breeding in commercially attractive pure fission breeders is not sufficient to meet the demand for new generating capacity. And second, the greatly increased energy generation per D-T fusion of the hybrid system (10 to 20 times that of a D-T fusion only system) significantly reduces plasma containment requirements for a viable power system. This should aid fusion research by providing an intermediate goal. The ultimate goal is of course an all fusion power economy.

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Table 1 - Uranium and Plutonium Metal Blanket Results (Per 14 MeV Neutron)

| Case # | Pu | $T^6$ | $T^7$ (Zone 1) | T | U(n,γ) | Pu(f) | U(f) | Energy | K |
|---|---|---|---|---|---|---|---|---|---|
| 1 | .01 | 0.567 | 0.425 | 0.99 | 1.82 | 0.123 | 0.426 | 126 | - |
| 2 | .02 | 0.666 | 0.421 | 1.09 | 2.32 | 0.351 | 0.552 | 195 | - |
| 3 | .03 | 0.755 | 0.427 | 1.18 | 2.75 | 0.633 | 0.617 | 261 | - |
| 4 | .04 | 0.947 | 0.438 | 1.38 | 3.92 | 1.25 | 0.880 | 431 | 0.843 |
| 5 | .05 | - | - | - | - | - | - | 929 | 0.934 |
| 6 | .04 + .08 FP | 0.751 | 0.434 | 1.18 | 2.49 | 0.827 | 0.643 | 306 | - |

where:

| | | | |
|---|---|---|---|
| Pu | ≡ | $Pu^{239}$ fraction of heavy atoms ($U^{238}$ + $Pu^{239}$) | per 14 MeV Source Neutron |
| $T^6$ | ≡ | $Li^6$ (n,γ) T reaction | |
| $T^7$ | ≡ | $Li^7$ (n,n'α) T reactions in Zone 1 | |
| T | ≡ | Total T breeding reactions | |
| U(n,γ) | ≡ | $U^{238}$ (n,γ) $U^{239}$ Pu breeding reactions | |
| Pu(f) | ≡ | $Pu^{239}$ (n, fission) reactions | |
| U(f) | ≡ | $U^{238}$ (n, fission) reactions | |
| Energy | ≡ | Blanket energy generation (MeV) | |
| K | ≡ | Neutron multiplication of water reflected blanket | |
| FP | ≡ | Fission Products | |

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Table 2 - Thorium Blanket Results

| $U^{233}$ | $T^6$ | $T^7$ (Zone 1) | T(total) | Th(n,γ) | U(f) | Th(f) | Energy |
|---|---|---|---|---|---|---|---|
| .05 | 0.515 | 0.417 | 0.93 | 1.61 | 0.719 | 0.106 | 172 |
| .08 | 0.868 | 0.422 | 1.29 | 3.00 | 2.31 | 0.18 | 478 |

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Table 3 - Mixed Oxide Blanket (Zone 2) Pu Enrichment and Material Volume Fractions

| Mixed Oxide Case # | Pu | Fuel | Li Coolant | Nb Structure |
|---|---|---|---|---|
| 1 | .08 | .65 | .30 | .05 |
| 2 | .09 | .65 | .30 | .05 |
| 3 | .09 | .60 | .30 | .10 |
| 4 | .09 | .50 | .40 | .10 |
| 5 | .11 | .50 | .40 | .10 |
| 6 | .11 | .50 | .30 | .20 |
| 7* | .11 | .50 | .30 | .20 |

* Zone 1 thickness reduced from 30 to 15 cm.

Table 4 - Mixed Oxide Blanket Results

| Case # | $T^6$ | $T^7$ (Zone 1) | T(total) | U(n,γ) | Pu(f) | U(f) | Energy | K |
|---|---|---|---|---|---|---|---|---|
| 1 | 0.825 | 0.435 | 1.26 | 2.02 | 1.09 | 0.44 | 316 | - |
| 2 | 1.062 | 0.436 | 1.50 | 2.82 | 1.76 | 0.62 | 483 | 0.835 |
| 3 | 0.828 | 0.436 | 1.26 | 1.80 | 1.13 | 0.38 | 313 | - |
| 4 | 0.828 | 0.428 | 1.26 | 1.41 | 0.843 | 0.31 | 243 | - |
| 5 | 1.13 | 0.441 | 1.57 | 1.72 | 1.54 | 0.42 | 400 | - |
| 6 | 0.851 | 0.421 | 1.27 | 1.50 | 1.21 | 0.30 | 313 | - |
| 7 | 0.891 | 0.252 | 1.14 | 2.63 | 2.06 | 0.50 | 521 | - |

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Tables 3 & 4 should be inserted in text following each other as shown.

Table 5 - Average Power Densities Required for a Pu Doubling Time of Five Years in Mixed Oxide Blankets

| Case # | $P_{ave}$ (W/cc) |
|---|---|
| 1 | 400 |
| 2 | 602 |
| 3 | 570 |
| 4 | 433 |
| 5 | 2777 |
| 6 | 1349 |
| 7 | 1142 |

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