Summary of the Fusion-Fission Hybrid Fuel Cycle Analysis (Tritiumless Hybrids?)

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FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

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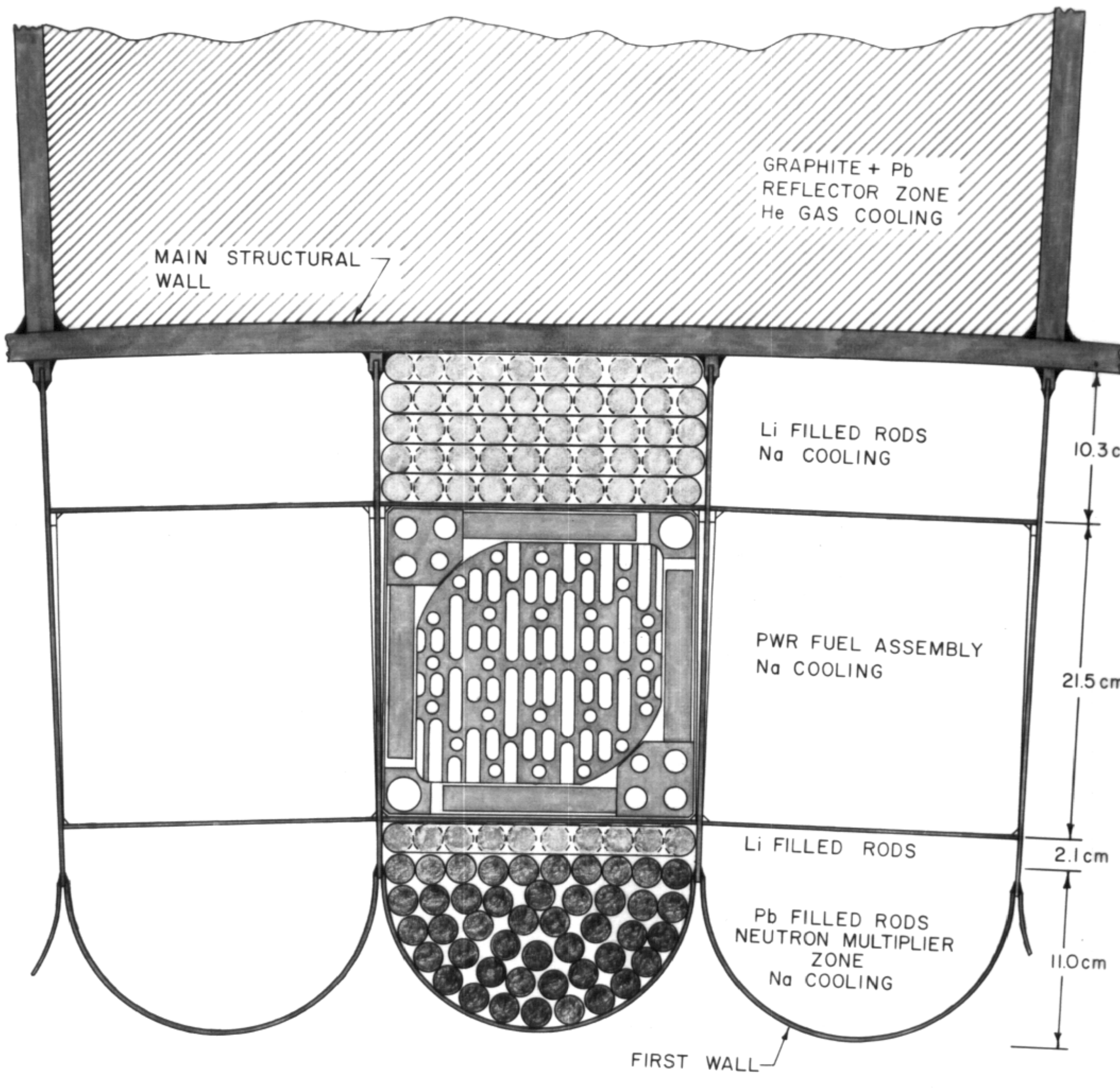
Abstract

Fusion-fission hybrid reactors offer the opportunity for fusion to impact the energy production scenario at an earlier date and in a more substantial fashion than simple fusion electricity reactors. This hypothesis is predicated on the belief that the technological problems associated with fusion-fission hybrids can be solved in a timely manner. Elimination or deferral of such systems could help to ease the introduction of fusion-fission reactors. In this summary we look at the elimination of the tritium breeding function in the fusion blanket and its effect on the early introduction of hybrid reactors.

I. Introduction

The fusion-fission hybrid reactor concept utilizes 14.1 MeV DT fusion neutrons to breed fissile fuel (^{239}Pu or ^{233}U) in the reactor blanket. The breeding is usually in addition to the breeding of tritium to complete the DT fusion fuel cycle. This bred fissile fuel can then be periodically removed from the hybrid blanket and burned in conventional light water fission reactors. Figure I-1 shows the blanket design of the SOLASE-H laser fusion hybrid where fertile fuel bearing LWR assemblies are directly enriched in the hybrid blanket to the required 4% fissile content and are then removed for use in LWRs without the intermediate reprocessing step. Most other fusion-fission hybrid designs require that the fuel removed from the blanket be reprocessed before introduction into the fission reactor.

In either of these cases the hybrid offers the opportunity to amplify the effect of fusion on the future energy scenario. This results from two things. (1) The ultimate energy release per fusion event is multiplied by ten (1 fusion $\simeq$ 20 MeV + 1 fission $\approx$ 200 MeV) and because of this, (2) the fusion performance requirements are reduced while still having an economic system. A large support ratio of LWR burner reactors to fuel producing hybrids could allow the hybrid cost to be large without greatly affecting the total energy production system cost. The blanket energy multiplication in the hybrid could improve the power balance in the fusion reactor itself, thus making poorer fusion performance economically acceptable. This could bring the date that fusion impacts the energy situation closer to the present.



CROSS SECTION OF SOLASE-H BLANKET

Fig. I-1

Many hybrid studies have quantified the amount of fissile fuel produced in a hybrid (support ratio) and the amount of power multiplication in the hybrid. These have led to a determination of acceptable fusion performance. However, this potential for earlier introduction of the hybrid posed other problems related to the fusion technology of hybrids. For instance, can the introduction of the hybrid be accelerated through elimination or alteration of ancillary systems such as tritium breeding and recovery? Elimination of this system could close the gap between the last fusion "experiment" and the first useful application of fusion. This of course raises the question: where will the tritium come from if it is not produced in the hybrid? The effect of eliminating the tritium breeding function from the hybrid and an analysis of the other possible sources of tritium and how they couple to the hybrid are the subject of this report.

II. Outline of Analysis

The analysis of the tritiumless hybrid is divided into several different parts shown in Fig. II-1. A survey of the fissile and tritium breeding performance in 18 hybrid reactor designs reported in the literature has been compiled. In addition to this survey, benchmark neutronics calculations have been done to compare blanket designs with and without tritium production. These are compared to the trends found in the survey.

Simple relations have been developed to model the mass flow of fissile fuel and tritium between the hybrid, burner LWRs and possible dedicated tritium producing fission reactors (such as those used at Savannah River). These indicate the advantage or disadvantage of separating the tritium producing function from the hybrid.

The economic benefit or penalty has been assessed for tritiumless hybrid systems as compared to the conventional hybrid. Figures of merit for ^{239}Pu and ^{233}U producing systems are computed.

Removal of the tritium breeding function from the hybrid has been analyzed in detail. A survey of the tritium systems in 15 conceptual reactor designs has been compiled. The importance of removing each system has been rated with a simple rating system.

A survey of possible tritium sources including fission reactors, dedicated production reactors, LMFBRs, fuel reprocessing plants, etc. has been made along with estimates of the tritium production capabilities of each.

TRITIUMLESS HYBRID ANALYSIS

. BLANKET PERFORMANCE

- SURVEY OF THE LITERATURE
- BENCHMARK CALCULATIONS

. MASS FLOW BALANCES OF TRITIUM AND FISSILE FUEL

- T-LESS HYBRID + FISSILE BURNERS
- T-LESS HYBRID + FISSILE BURNERS
  + DEDICATED T PRODUCER

. ECONOMIC FIGURE OF MERIT

. TRITIUM SYSTEMS IN THE HYBRID

. TRITIUM SOURCES - SURVEY

Fig. II-1

III. Blanket Performance

A survey of many hybrid blanket designs reported in the literature is reported in UWFDM-308. Tables III-1 and III-2 show the data compiled for ^{233}U and ^{239}Pu breeding blankets respectively. Fig. III-1 shows the relation between the number of breeding neutron captures, both fissile and tritium,as a function of the blanket multiplication for a subset of the blankets (those producing ^{239}Pu in a hard spectrum). These design points include blankets that are self-sufficient in tritium and those that are not. To supplement these specific design points, parametric neutronics calculations have also been done to systematically determine the effect of removing the tritium breeding function from the hybrid blanket. These are reported in UWFDM-334. A schematic of the blanket model for these calculations is shown in Fig. III-2 and the results are given in Table III-3. Some of these points are also plotted on Fig. III-1. From this figure it is clear that the total number of breeding captures increases with the blanket multiplication. The difference in total breeding captures between blankets that are self-sufficient in tritium production and those that produce no tritium is one. This might be expected since there is now one additional neutron that can be used to produce fissile material. This data can be adequately represented by the relation

$$\# \text{ of breeding captures} = 2.18 + 0.063\ M \ .$$

By itself, this data does not support or refute the idea of removing the tritium breeding function from the hybrid. To do this, these results must be joined into an analysis of the total system, including the external sources of tritium. This is done in the next section.

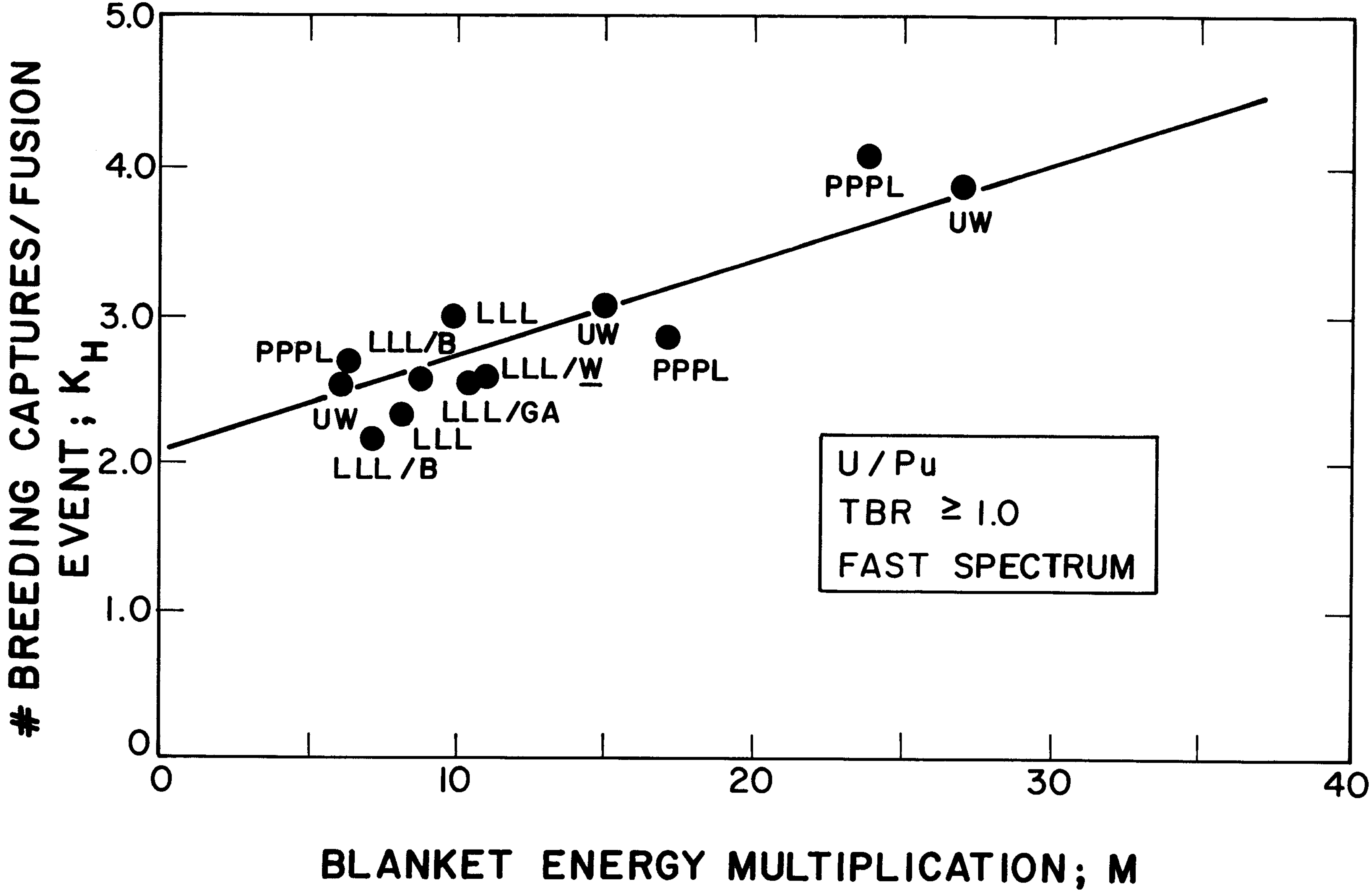
Table III-1

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 plasm. I/P 114	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 $\frac{MW}{m^2}$	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 $\frac{kW}{\ell}$				

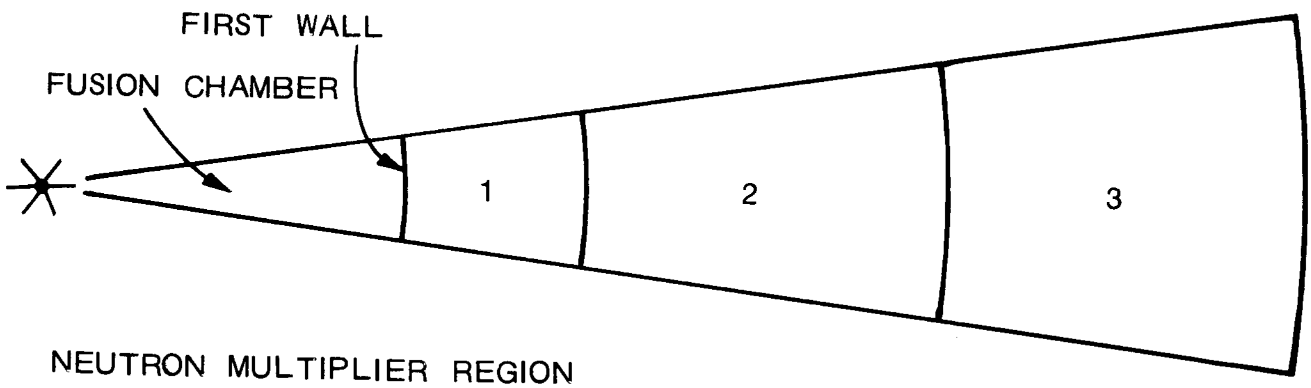
Table III-2 Selective Designs of Fusion-Fission Systems Which Breed Pu-239

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 $n\tau = 3.5 \times 10^{13}$/ 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 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 η_t=39% η_{net}=32%	Gross=400 MWt η_{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 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	



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

Figure III-1



① NEUTRON MULTIPLIER REGION

② FUEL REGION

③ TRITIUM BREEDING REGION OR VACUUM DEPENDING ON TRITIUM BREEDING SYSTEM OR TRITIUMLESS SYSTEM

FUELS

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

COOLANT

- LIQUID METAL Na

NEUTRON MULTIPLIER

- Pb

COMPOSITION OF FUEL ZONE (VOL. %)

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

COMPOSITION OF MULTIPLIER ZONE

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

Fig. III-2

Table III-3

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

* UO_2/PuO_2

+ UC/PuC

IV. Fissile and Tritium Material Flow

In this section we investigate the potential of the tritiumless hybrid by computing the total fissile material produced and the total amount of tritium produced in a system where the tritium is manufactured external to the hybrid. This is then compared to a system where the tritium is bred in the hybrid. Three different scenarios are shown in Fig. IV-1. In Fig. IV-1a the hybrid produces all of its own tritium and the fissile material goes to the burner reactors. In Fig. IV-1b the hybrid is fueled by a dedicated tritium source (DTS) such as a Savannah River type production reactor. In Fig. IV-1c the hybrid tritium is supplied by modified burner reactors. Fig. IV-1d indicates that a large number of combinations of these simple scenarios can be chosen.

Simple balance relations can be obtained for the steady state tritium and fissile production in any such system. These are:

1. Tritium Balance

$$(1 - TBR_H) =$$

$$\left(\frac{E_{fus}}{E_{fis}}\right) \{[P(1+\alpha)(TBR)]_{FR} + [P(1+\alpha)(TBR)]_{DTS}\}$$

2. Fissile Balance

$$(FBR)_H =$$

$$\left(\frac{E_{fus}}{E_{fis}}\right) \{[P(1+\alpha)(1-CR)]_{FR} + [P(1+\alpha)(1-CR)]_{DTS}\}$$

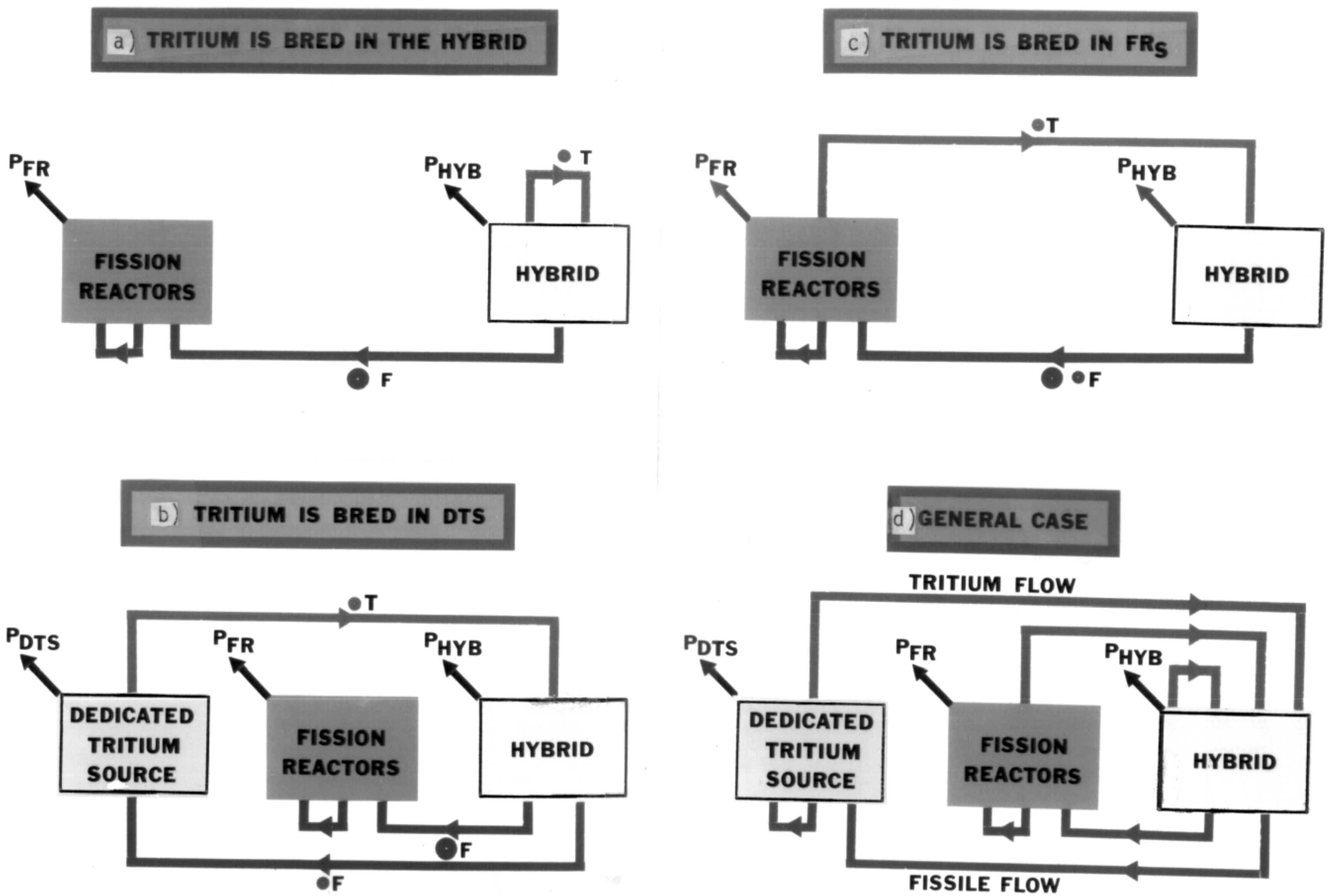

a) TRITIUM IS BRED IN THE HYBRID
P_{FR}
FISSION REACTORS
P_{HYB}
HYBRID
T
F
c) TRITIUM IS BRED IN FR_S
P_{FR}
FISSION REACTORS
P_{HYB}
HYBRID
T
F
b) TRITIUM IS BRED IN DTS
P_{DTS}
DEDICATED TRITIUM SOURCE
P_{FR}
FISSION REACTORS
P_{HYB}
HYBRID
T
F
F
d) GENERAL CASE
TRITIUM FLOW
P_{DTS}
DEDICATED TRITIUM SOURCE
P_{FR}
FISSION REACTORS
P_{HYB}
HYBRID
FISSILE FLOW

Fig. IV-1

where

TBR_H -- tritium breeding ratio in the hybrid

FBR_H -- fissile breeding ratio in the hybrid

E_{fus} -- energy released per fusion event

E_{fis} -- energy released per fission event

P -- thermal power normalized to the fusion power

α -- capture to absorption ratio

CR -- fissile conversion ratio

$[\ ]_{FR}$ -- all quantities in the brackets evaluated for a burner fission reactor

$[\ ]_{DTS}$ -- all quantities in the brackets evaluated for a dedicated tritium producer.

The thermal power of the hybrid, normalized to the fusion power is

$$P_H = \frac{1}{\eta_D G} + 1 + f_n(M-1)$$

where $\eta_D G$ -- fusion energy gain

f_n -- fraction of energy in neutrons

M -- blanket multiplication.

Another important relationship is that between the blanket multiplication and the total (T+fissile) breeding ratio in the hybrid. For the U/Pu fuel cycle this is given by

$$ToTBR_H = 2.18 + 0.063\ M.$$

This was determined by the survey and calculations reported in part III of this report.

Finally a model of the performance of the dedicated tritium source is needed. Information obtained from Savannah River Laboratory (see UWFDM-317) indicates that the best tritium breeding ratio that can be obtained in their production reactors is about 0.85-0.9. Hence we can immediately see that the extra fissile atom produced in the hybrid because the tritium breeding function was removed will be required to fuel the dedicated tritium source to produce the tritium there. This is clearly demonstrated in Table IV-1. The normalized power of the burner fission reactors is the same in each of the two limiting cases. Hence the combination of a hybrid and a dedicated tritium production fission reactor yields the same net amount of fissile fuel for the burner reactors as a hybrid that produces its own tritium. If the tritium is bred in the burner fission reactors themselves, the same result is found, Table IV-2.

Thus the mass flow in the three possible scenarios given in Table IV-1 results in a zero sum game. No more net fissile material can be produced by a hybrid that does not breed its own tritium because the additional fissile material that is produced must be sacrificed somewhere else to produce the tritium. To a good approximation, these effects cancel each other. However, the thermal power produced in the three systems is distributed differently among the various reactors. Hence one system may be distinguished economically over the others if the cost per thermal megawatt is different for the different reactors. This problem is addressed in the next section.

Table IV-1

	Th/U Cycle Conversion Ratio in FR			U/Pu Cycle Conversion Ratio in FR		
	0.65	0.8	0.95	0.65	0.8	0.95
	All Tritium Bred in the Hybrid*					
P_{DTS}	0	0	0	0	0	0
P_H	2.25	2.25	2.25	4.5	4.5	4.5
P_{FR}	17	30	120	42	75	300
	All Tritium Bred in the DTS†					
P_{DTS}	10	10	10	10	10	10
P_H	2.25	2.25	2.25	4.5	4.5	4.5
P_{FR}	17	30	120	42	75	300

*TBR = 1.0
FBR(Th/U) = 0.6
FBR(U/Pu) = 1.5
M(Th/U) = 2
M(U/Pu) = 5

†TBR = 0.0
FBR(Th/U) = 1.6
FBR(U/Pu) = 2.5
M(Th/U) = 2
M(U/Pu) = 5

Table IV-2

All Tritium is Bred in the Fission Reactors

	Th/U Cycle* Conversion Ratio**			U/Pu Cycle+ Conversion Ratio**		
	0.65	0.80	0.95	0.65	0.80	0.95
P_{DTS}		Zero			Zero	
P_{HYB}	2.25	2.25	2.25	4.5	4.5	4.5
P_{FR}	17.0	29.8	119	42.0	74.5	298
T_{FR}	0.585	0.333	0.084	0.233	0.133	0.033
(CR)	(0.065)	(0.467)	(0.866)	(0.417)	(0.667)	(0.917)

* TBR = 0.0, FBR = 1.6, M = 2

\+ TBR = 0.0, FBR = 2.5, M = 5

** Number of tritium and fissile atoms produced per fissile fuel absorption event in the fission reactor.

V. Figures of Merit for Tritiumless Hybrid Systems

Since the mass flows in the different hybrid systems are a zero sum game there is no justification for a tritiumless hybrid on the grounds that it produces more fuel or power. However, the power is distributed differently among the different reactors and these reactors may not cost the same amount. Hence the best figure of merit for the system will be the total cost of electricity. Thus a figure of merit might be defined as

$$\text{FoM} = \frac{C_{FR}P_{FR} + C_H P_H + (C_{DTS} + \alpha\delta)P_{DTS}}{P_{FR} + P_H + \delta P_{DTS}}$$

where

C_{FR} -- Cost/MW_t of the fission reactor

C_H -- Cost/MW_t of the hybrid

C_{DTS} -- Cost/MW_t of the dedicated tritium source

δ -- = 0 dedicated tritium source does not produce electricity
= 1 dedicated tritium source does produce electricity

α -- a coefficient to account for the additional cost of the DTS if it produces electricity.

It is desired to minimize this figure of merit. This can be rewritten by normalizing to the cost of a fission reactor.

$$\text{FoM} = \frac{P_{FR} + (C_H/C_{FR}) + (C_{DTS} + \alpha\delta)/C_{FR}\; P_{DTS}}{P_{FR} + P_H + \delta P_{DTS}} .$$

This figure of merit is given in Table V-1 for a specific set of assumptions. From this we see that a severe penalty is taken if the tritium is produced in a Savannah River-like production reactor that does not produce electricity.

Table V-1

Figures of Merit

Case 1. Th/U; all tritium is bred in the hybrid.

$$\text{FoM} = \frac{17+2.5*2.25}{17+2.25} = 1.17$$

Case 2. Th/U; all tritium is bred in the DTS, no electricity is produced in the DTS.

$$\text{FoM} = \frac{17+2.5*2.25+10}{17+2.25} = 1.69$$

Case 3. Th/U; all tritium is bred in DTS, electricity is produced in DTS.

$$\frac{17+2.5*2.25+10*1.5}{17+2.25+10} = 1.29$$

Case 4. U/Pu; all tritium is bred in the hybrid.

$$\text{FoM} = \frac{42+1.75*4.5}{42+4.5} = 1.07$$

Case 5. U/Pu; all tritium is bred in the DTS, no electricity is produced in the DTS.

$$\text{FoM} = \frac{42+1.75*4.5+10}{42+4.5} = 1.29$$

Case 6. U/Pu; all tritium is bred in the DTS; electricity is produced in the DTS.

$$\text{FoM} = \frac{42+1.75*4.5+10*1.5}{42+4.5+10} = 1.15$$

However, if the production reactor does produce electricity at an efficiency comparable to a fission power reactor then a negligible cost penalty is paid for the overall system capital cost. This of course may be sensitive to the specific assumptions made about the relative capital costs of hybrids, fission reactors, and tritium production facilities. For instance it does not take into account the fact that the hybrid should be less expensive in case 2 than in case 1 because the tritium breeding system has been removed. If the relative cost of the hybrid in case 2 drops to 1.5 times the cost of a fission reactor from the assumed 2.5, then the first three cases are changed to the values shown in Table V-2. Here we see that case 3, where tritium is produced in a DTS that also produces electricity has a figure of merit comparable to case 1 where all tritium is produced in the hybrid. Hence, if the elimination of the tritium breeding function from the hybrid can significantly reduce its cost, then the idea continues to have merit.

In the next section we review problems associated with the tritium breeding system in a hybrid reactor.

Table V-2

Figures of Merit

Case 1. Th/U; all tritium is bred in the hybrid.

$$FoM = \frac{17+2.5*2.25}{17+2.25} = 1.17$$

Case 2. Th/U; all tritium is bred in the DTS, no electricity is produced in the DTS.

$$FoM = \frac{17+1.5*2.25+10}{17+2.25} = 1.58$$

Case 3. Th/U; all tritium is bred in the DTS, electricity is produced in the DTS.

$$FoM = \frac{17+1.5*2.25+10*1.5}{17+2.25+10} = 1.2$$

VI. Tritium Breeding in the Hybrid

An exhaustive survey of the literature of fusion conceptual reactor designs has resulted in a compilation of tritium system data shown in Table VI-1. Details of this survey and analysis are given in UWFDM-321. Although a large number of details are included in the table, the most important fact is displayed in Fig. VI-1. In any DT burning fusion reactor, only a fraction of the tritium entering the reaction chamber (plasma) is burned before it is pumped away. This is denoted as the fractional burnup. Table VI-1 shows that for most fusion reactors this fractional burnup is quite small, ~10%. Hence the vast majority of the tritium mass flow in the reactor is handled in the exhaust system rather than in the blanket recovery system. This exhaust system will be required even if the hybrid reactor does not breed its own tritium supply. The exception to this is inertial fusion where the fractional burnup can be as high as 60%. Table VI-2 shows normalized tritium inventories in 6 different tokamak conceptual designs. Note that in the most recent designs (NUWMAK for instance) there is little tritium inventory associated with the blanket. The major components of the tritium inventory are in the storage and fueling systems, which would be required even if there were no tritium breeding. The same conclusion can be made about inertial fusion reactors where it is found that the majority of the tritium is in the pellet manufacturing and storage systems. Hence the removal of the tritium breeding function does not significantly reduce the tritium inventory in the reactor and therefore any systems related to the tritium inventory such as emergency cleanup systems will be of the same magnitude as in a tritium producing reactor.

Table VI-1 Exhaust Characteristics

	TOKAMAKS						PINCHES		MIRRORS		INERTIAL				HYBRIDS		
REACTOR	UWMAK I	PRD	UWMAK II	UWMAK III	G.A. NONCIRC.	NUWMAK	RTPR	REVERSED FIELD	STANDARD	TMR	SOLASE (GLASS)	SOLASE (PVA)	LASER LIQ Li	FAST LINER	PPPL TOK.H.	STAND. MIRROR H.	SOLASE H
Date	3/74	8/74	10/75	7/76	11/76	6/79	3/74	77	1/78	7/77	12/77		77	2/79	11/78	5/78	6/79
Designed for (MW_e)	1474	2030	1716	1985	611	660	4100	600	447	1000	1000	1000	380	129	2419	603	720-870 (av. 800)
Normalized to	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000
Breeding (kg T/d)	0.70	0.28	0.38	0.34	0.50	0.68	0.35	0.66	0.51	0.43	0.63	0.63	0.69	0.44	0.045	0.14	0.25
Burnup (%)	7.2	8.7	4.85	0.83	1.35	1.5	0.97[a]	3.8[a]	3.0	18	41	41	40	11	0.92	1.6	27
Fueling and Exhaust (kg/d)																	
T fed	5.70	2.99	7.49	37.94	30.29	30.10	34.76	14.95	14.12	2.12[b]	1.14	1.14	1.01	3.36	4.07	6.33	0.89
T pumped	5.22	2.73	7.12	37.63	29.89	29.66	34.42	14.39	13.69	1.73[b]	0.37[c]	0.42[c]	0.60	2.99	4.03	6.22	0.64
D fed	3.80	1.99	4.99	25.30	20.14	20.07	23.17	9.97	21.09	2.29[b]	0.76	0.76	0.67	2.24	2.72	8.14	2.92
D pumped	3.48	1.82	4.75	25.09	19.87	19.77	22.95	9.59	20.81	2.03[b]	0.25[c]	0.28[c]	0.40	1.99	2.69	8.07	2.76
H entering plasma		0.002		1.0	0.080	0.20					0.48	0.63					
H produced		0.001		0.001	0.002										$3.x10^{-5}$		
H pumped		0.003		1.0	0.082	0.200	<0.23				0.26[c]	0.39[c]				0.01	
Inert gas entering plasma		3.8(Ar)(d)	1.2(He)(e)	4.6(He)(e)			0.48(He)[e]				?(Xe)[f]	?(Xe)[f]					?(Xe)[f]
He produced	0.61	0.35	0.48	0.42	0.54	0.58	0.46	0.75	0.58	0.52	0.62	0.62	0.54		0.055	0.14	0.32
Inert gas pumped	0.61	4.1	1.7	5.0	0.54	0.58	0.94	0.75	0.58	0.52	>0.62	>0.62	0.54		0.055	0.14	>0.32
Gas Blanket (material)	None	None	None	None	DT	DT	DT	None	None	None	Ne	He	Liq.Li	Liq.Li		D_2	Xe
(kg/d)											4440	5810	$1.2x10^{10}$	$1.17x10^9$		42.3	377000
$C_2(H,D,T)_2$ pumped					2.2						5.8	5.6					
CO pumped											4.3	1.1					0.84
Nonvolatiles (material)	316SS	PE-16	316SS+C	TZM+C	Si		Al_2O_3		C		Oxides, Carbides&C	C	High-Z Material	Cu			C
(source)	1st Wall	1st Wall	1st Wall & Curtain	Collector + Curtain & ISSEC	1st Wall Coating		1st Wall Coating		Direct Converter		Pellet	Pellet	Pellet	Liner			Pellet
(kg/d)	9.3[h]	2.4[h]	0.24 + 0.66[h,i]	18+1.5[j]	4.7[k]		0.44[ℓ]		1.6[m]		6.3	3.8	~100	$3.3x10^7$			2.5
Chamber Pres. (torr)	10^{-5}	$4.5x10^{-4}$	$8x10^{-5}$	$3x10^{-4}$	10^{-3}	$9x10^{-6}$	$4.6x10^{-2}$		$3-4x10^{-6}$		0.5	0.5	0.1	1		$10^{-6}-10^{-4}$	2.8
Temp. (K)	773	600	600	588	1700	573	810		300-1000		2023	2023	773	773			
Pump Type	Li divert. + Hg. Diff. & Cryo.	Diff. or Turbomolecular	Li Divert. + Cryo.	Cryo.	Cryo.	Cryo.	Roots Blowers		Cryo.	Cryo.	Roots Blowers	Roots Blowers	Li Waterfall	Roots + Li Spray	Cryo.	Cryo.	Roots Blowers
Pumping Speed (torr ·ℓ/s molecules at 300 K)	410	230	600	3,100	2,200	2,100	2,600	1,100	1,700	200	48,000	63,000	73		290	3,000	620,000
Hydrogen Isotopes (% of exhaust)	91.9	83.2	84.6	91.3	97.9	98.5	98.0	96.2	98.1	86.0	0.12	0.12			99.0	99.7	0.028
Neutral Beam Injectors (kg/d)				None		None	None	None				None	None	None			None
T Recycled	0.0004[n]	Small	0.003[n]		None				102	11.7[o]					4.4	40.2	
D Recycled	0.0002[n]	Small	0.002[n]		0.2[n]				152	12.7[o]					2.9	52.8	

Footnotes on following page

Footnotes for Table VI-1

(a) The burnups are 4.8 and 30% for the RTPR and Reversed Field, respectively, at the end of the "quench" stage but only ~ 1 and 4% when the DT gas blanket used to dilute the ash and impurities is considered.

(b) Only the total number of D and T ions are given in this report. Table I values are calculated by assuming the central cell is fueled with an equimolar D:T mixture. The larger number of moles of D reflects its presence in the end plugs as well as the central cell.

(c) H, D and T alsoleave the chamber as $C_2(H,D,T)_2$.

(d) Ar is added to the feed to prevent excessive escalation of electron temperatures in the plasma.

(e) He is an impurity in the fuel.

(f) Xe is the high-Z material in the fuel pellet.

(g) The DT blanket acts as a fueling mechanism and accounts for 67% of the fuel fed to the reactor in the case of G.A. Noncirc. and 100% in the case of NUWMAK, RTPR and Reversed Field.

(h) This erosion rate is due to consideration of charged particle blistering and sputtering and neutron sputtering.

(i) These values represent geometric means between optimistic (0.019 + 0.096) and pessimistic (2.9 + 4.4) predictions of the erosion rate of the 1st wall + curtain.

(j) TZM erosion rate is due to consideration of charged particle sputtering; C erosion rate is due to consideration of vaporization and charged particle and neutron sputtering.

(k) SiC erosion is due to α sputtering and chemical reaction with atomic D and T.

(l) This erosion rate considers neutron sputtering only and represents the geometric mean between optimistic (0.14) and pessimistic (1.4) predictions.

(m) This erosion rate is due to consideration of charged particle sputtering.

(n) This value is not given in the report. It is calculated assuming the quantity of gas recycled is 7 times that injected.

(o) Only the percent of D(85%) recycled in the high energy neutral beam lines driving the end plugs is given in this report. To arrive at the total recycle values, this recirculating fraction (0.85) was assumed to hold for the low energy neutral beam lines fueling the central cell as well (see also footnote b).

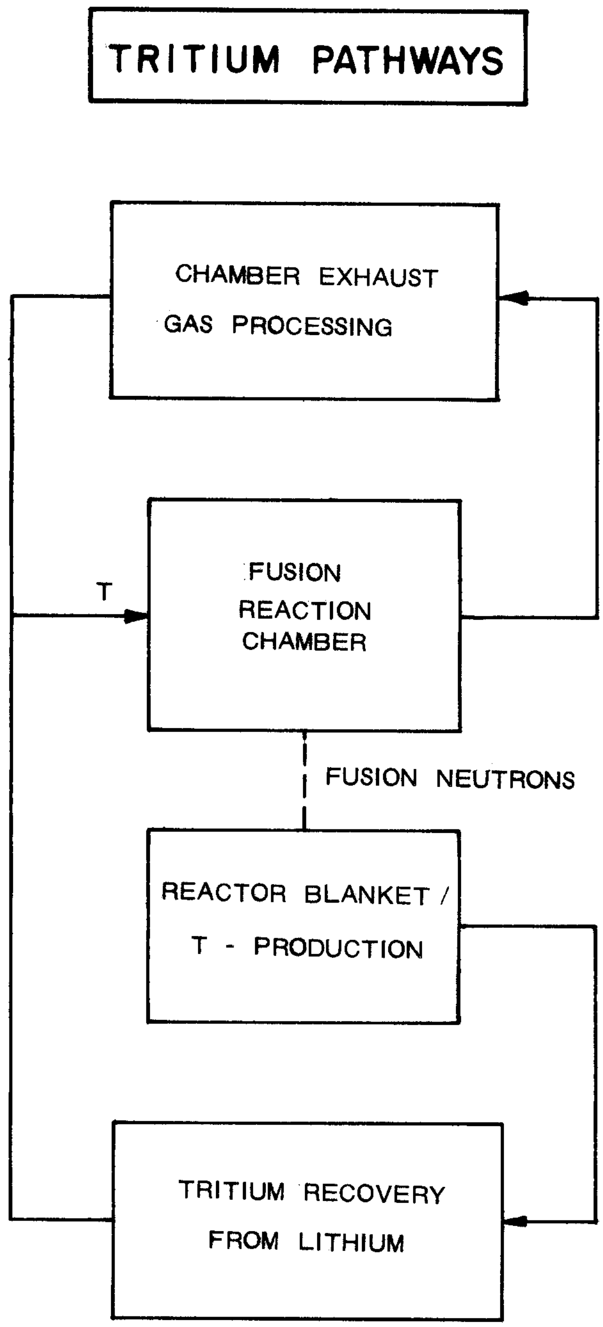

TRITIUM PATHWAYS
CHAMBER EXHAUST
GAS PROCESSING
T
FUSION
REACTION
CHAMBER
FUSION NEUTRONS
REACTOR BLANKET /
T - PRODUCTION
TRITIUM RECOVERY
FROM LITHIUM

Fig. VI-1

Table VI-2

Normalized Tritium Inventories of Tokamaks

Reactor	UWMAK I	PRD	UWMAK II	UWMAK III	G.A. Noncirc. Tok. Demo.	NUWMAK
Fueling System (kg)						
Pumps	0.033		0.050	3.1	1.54	1.60
H Isotope Extraction	1.85		3.06			-
Liquefaction and Isotope Fractionation	-	0.51	-	3.1	0.62	0.3
Fueling Mechanism	-		-		1.00	-
Storage (kg)	4.20	3.03	6.43	37.7	9.25	9.70
Blanket System (kg)						
Breeding Material	8.70	0.10	0.73	1.00	0.12	0.088
Breeder Reprocessing	0.25	-	0.21	0.17	0.75	0.010
Total T Inventory (kg)	15.03	3.64	10.47	45.02	13.28	11.74
Total T Inventory (kg) Per 1000 MW_e	10.2	1.80	6.10	22.7	21.7	17.8
Burnup (%)	7.2	8.7	4.85	0.83	1.35	1.5

Although there are some detailed differences between the various fusion systems, the basic conclusion of this analysis is that removal of the tritium breeding function from the hybrid does not substantially alter its technical complexity or cost. With the possible exception of a detailed problem that could not be detected by this level of analysis there is no reason to exclude the tritium breeding function from early hybrid development on the presumption that it greatly simplifies the technology.

VII. Sources of Tritium

A survey of tritium sources has been made and reported in UWFDM-317. This survey includes descriptions of the Savannah River production reactors with estimates of their tritium breeding capability and the Hanford N-Reactor, the only power producing reactor to make substantial quantities of tritium. Detailed two dimensional neutronics calculations of LMFBR cores with Li bearing blankets have been performed at Kernforschungszentrum Karlsruhe in West Germany at the request of the UW Fusion Program to assess the potential of tritium breeding in LMFBRs. Finally the numerous results of Rhinehammer and Wittenberg have been reproduced here for comparison with our other studies. A summary of these results is given in Table VII-1. The production reactors clearly consume the greatest amount of fissile fuel themselves (the tritium breeding ratio is always gained at the expense of the conversion ratio). Liquid metal fast breeder reactors have the potential of producing large quantities of tritium while maintaining their own self-sufficiency in fissile fuel. However the number of LMFBRs of equivalent power to the hybrid that are required to feed it tritium are so great that this scenario makes little sense.

Table VII-1 Possible Tritium Producing Sources

Source	Production kg/(MWe·year)	Remarks
Existing Light Water Reactors (LWRs)	7.50×10^{-8}	From shim control boron
Heavy Water Reactors	1.90×10^{-4}	Activation of D_2O
Fuel Reprocessing	1.18×10^{-6}	Ternary fission tritium
Liquid Metal Fast Breeder	5.1×10^{-3}	Li_2O used in the axial and radial blankets
Savannah River Reactors	$\sim 1.66 \times 10^{-2}$	Maximum, No power production
Hanford N-Reactor	4.34×10^{-3}	Minimum + Pu + power
	1.14×10^{-2}	Maximum, No Pu, + power

Fuel Consumption in Fusion Reactors: 0.11 - 0.17 kg/(MWe·year)

VIII. Conclusions

This multi-faceted analysis of the fusion-fission hybrid fuel cycle has yielded several significant conclusions.

(1) Production of tritium external to the hybrid reactor is not an attractive alternative for technical and economic reasons. It does not significantly ease the technology development associated with hybrid introduction.

(2) Steady-state mass flow balances show that tritium production in a dedicated tritium source compared to a tritium producing hybrid is a zero sum game. Hybrid blanket design studies show that removal of tritium breeding increases the fissile breeding ratio by one atom/fission event at constant blanket multiplication. Analysis of dedicated tritium production reactors (Savannah River) shows that breeding ratios of about one tritium/fissile atom consumed are achievable. Hence the two effects cancel each other.

(3) A figure of merit that measures the total hybrid plus fission reactor system capital cost shows that a dedicated tritium production facility that does not also produce electricity puts a severe penalty on the total system capital cost. If this production reactor produces electricity at nearly the same efficiency as normal fission reactors then the figure of merit is nearly the same as for a tritium producing hybrid. If the removal of the tritium breeding function from the hybrid significantly lowers its cost, then a hybrid coupled to a dedicated tritium producer that also produces electricity can have a better figure of merit than the tritium producing hybrid system.

(4) Analysis of the tritium subsystem of the hybrid indicates that there is no substantial technology simplification or cost saving associated with removal of the tritium breeding function from the blanket.