FISSILE AND FUSILE BREEDING IN THE THORIUM FUSION FISSION HYBRID

Magdi Ragheb and Ayman Nour Eldin

Department of Nuclear, Plasma and Radiological Engineering
University of Illinois at Urbana-Champaign,
216 Talbot Laboratory,
104 South Wright Street,
Urbana, Illinois 61801, USA.
mragheb@illinois.edu, anourel2@illinois.edu

ABSTRACT

The thorium fuel cycle fissile and fusile breeding in a molten salt and a solid fuel blanket for a system consisting of fusion fuel factories coupled to fission satellites burners, is considered. The use of the thorium cycle in a fusion fission hybrid could bypass the stage of fourth generation fission breeder reactors in that the energy multiplication in the fission island allows the satisfaction of energy breakeven and the Lawson condition in magnetic and inertial fusion reactor experiments, hence an early introduction of fusion energy.

The nuclear performance of a fusion-fission hybrid reactor having a molten salt composed of Na-Th-F-Be as the blanket fertile material and operating with a catalyzed or a semi-catalyzed Deuterium-Deuterium (DD) plasma is compared to a system with a Li-Th-F-Be salt operating with a Deuterium-Tritium (DT) plasma. In a fusion reactor with a 42-cm thick salt blanket followed by a 40-cm thick graphite reflector, the catalyzed DD system exhibits a fissile nuclide production rate of 0.88 Th(n, γ) reactions per fusion source neutron. The DT system, in addition to breeding tritium from lithium for the DT reaction yields 0.74 Th(n, γ) breeding reactions per fusion source neutron. Even though both approaches provide substantial energy amplification through the fusion-fission coupling process, the DT system possesses marginal tritium breeding in the fusion island of 0.467 triton / source neutron and would need supplemental breeding in the fission satellites to reach a value of unity. Neutron multiplication and flux trap strategies are needed to maximize the fusile breeding aspects.

In a solid enrichment factory approach using a flux trap concept and lead as a neutron multiplier to maximize breeding, a tritium yield per source neutron above unity of 1.08 and a Th (n, γ) reaction yield of 0.43 can be obtained for the DT fusion plasma in a concept where ThO_2 Zircaloy-clad fuel assemblies for Light Water Reactors (LWRs) are enriched in the U^{233} isotope. This corresponds to 0.77kg/[MW(th).year] of fissile fuel production, and 1.94 years of irradiation in the fusion reactor to attain an average 3 w/o fissile enrichment in the fuel assemblies.

1. INTRODUCTION

With the present day availability of fissile U^{235} and Pu^{239}, and available fusion and accelerator neutron sources, a fresh look at the Thorium-U^{233} fuel cycle is warranted. The use of the thorium cycle in a fusion fission hybrid could bypass the stage of fourth generation breeder reactors in that the energy multiplication in the fission part allows the satisfaction of energy breakeven and the Lawson condition in magnetic and inertial fusion reactor designs. This allows for the incremental development of the technology for the eventual introduction of a pure fusion technology [1-10].

2. ADVANTAGES OF THE THORIUM FUEL CYCLE

The following advantages of the thorium fuel cycle over the U^{235}-Pu^{239} fuel cycle can be enumerated [11-17]:
1. Breeding is possible in both the thermal and fast parts of the neutron spectrum with a regeneration factor of $\eta > 2$ (Fig. 1).

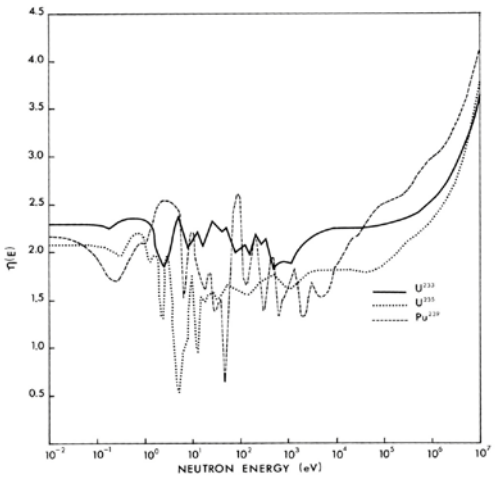

Figure 1: Regeneration factor as a function of neutron energy for the different fissile isotopes.

2. Expanded nuclear fuel resources due to the higher abundance of the fertile Th^{232} than U^{238}. The USA proven resources in the state of Idaho amount to 600,000 tons of 30 percent of Th oxides. The probable reserves amount to 1.5 million tons. There exists about 3,000 tons of already milled thorium in a USA strategic stockpile stored in Nevada.
3. Lower nuclear proliferation concerns due to the reduced limited needs for enrichment of the U^{235} isotope that is needed for starting up the fission cycle and can then be later replaced by the bred U^{233}. The fusion fission fusion hybrid totally eliminates that need.
4. A superior system of handling fission products wastes than other nuclear technologies and a much lower production of the long lived transuranic elements as waste. One ton of natural Th^{232}, not requiring enrichment, is needed to power a 1,000 MWe reactor per year compared with about 33 tons of uranium solid fuel to produce the same amount of power. The thorium just needs to be purified then converted into a fluoride. The same initial fuel loading of one ton per year is discharged primarily as fission products to be disposed of for the fission thorium cycle.
5. Ease of separation of the lower volume and short lived fission products for eventual disposal.
6. Higher fuel burnup and fuel utilization than the U^{235}-Pu^{239} cycle.
7. Enhanced nuclear safety associated with better temperature and void reactivity coefficients and lower excess reactivity in the core.
8. With a tailored breeding ratio of unity, a fission thorium fuelled reactor can generate its own fuel, after a small amount of fissile fuel is used as an initial loading.
9. The operation at high temperature implies higher thermal efficiency with a Brayton gas turbine cycle instead of a Joule or Rankine steam cycle, and lower waste heat that can be used for desalination or space heating. An open air cooled cycle can be contemplated eliminating the need for cooling water and the associated heat exchange equipment in arid areas of the world.
10. A thorium cycle for base-load electrical operation would provide a perfect match to peak-load cycle wind turbines generation. The produced wind energy can be stored as compressed air which can be used to cool a thorium open cycle reactor, substantially increasing its thermal efficiency, yet not requiring a water supply for cooling.
11. The unit powers are scalable over a wide range for different applications such as process heat or electrical production. Units of 100 MWe each can be designed, built and combined for larger power needs.
12. Operation at atmospheric pressure without pressurization implies the use of standard equipment with a lower cost than the equipment operated at high pressure in the LWRs cycle.

3. FUSION ISLAND CONSIDERATIONS

For an immediate application of the fusion hybrid using the Th cycle, the DT fusion fuel cycle can be used:

$$ {}_1D^2 + {}_1T^3 \rightarrow {}_2He^4(3.52\ MeV) + {}_0n^1(14.06\ MeV) + 17.58\ MeV \quad (1) $$

The tritium would have to be bred from the abundant supplies of lithium using the reactions with its two isotopes:

$$ {}_3Li^6 + {}_0n^1(thermal) \rightarrow {}_2He^4(2.05\ MeV) + {}_1T^3(2.73\ MeV) + 4.78\ MeV \quad (2) $$
$$ {}_3Li^7 + {}_0n^1(fast) \rightarrow {}_2He^4 + {}_0n^1 + {}_1T^3 - 2.47\ MeV $$

In this case a molten salt containing Li for tritium breeding as well as Th for U^{233} breeding can be envisioned:

$$ LiF.BeF_2.ThF_4 $$

For a practically unlimited supply of deuterium from water at a deuterium to hydrogen ratio of D/H = 150 ppm in the world oceans, one can envision the use of the catalyzed DD reaction in the fusion island:

$$ {}_1D^2 + {}_1D^2 \rightarrow {}_1T^3(1.01) + {}_1H^1(3.03) + 4.04\ MeV $$
$$ {}_1D^2 + {}_1D^2 \rightarrow {}_2He^3(0.82) + {}_0n^1(2.45) + 3.27\ MeV $$
$$ {}_1D^2 + {}_1T^3 \rightarrow {}_2He^4(3.52) + {}_0n^1(14.06) + 17.58\ MeV $$
$$ {}_1D^2 + {}_2He^3 \rightarrow {}_2He^4(3.67) + {}_1H^1(14.67) + 18.34\ MeV $$

$$ 6\,{}_1D^2 \rightarrow 2\,{}_1H^1 + 2\,{}_2He^4 + 2\,{}_0n^1 + 43.23 MeV \quad (3) $$

with each of the six deuterons contributing an energy release of 43.23 / 6 = 7.205 MeV.

For plasma kinetic reactions temperatures below 50 keV, the DHe^3 reaction is not significant and the energy release would be 43.23 - 18.34 = 24.89 with each of the five deuterons contributing an energy release of 24.89/5 = 4.978 MeV.

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$$ 5\,{}_1D^2 \rightarrow {}_1H^1 + {}_2He^3 + {}_2He^4 + 2\,{}_0n^1 + 24.89 MeV $$

In this case, there would be no need to breed tritium, and the lithium can be replaced by Na in a molten salt with the following composition:

$$ NaF.BeF_2.ThF_4 $$

With a density and percentage molecular composition of:

$$\rho = 4.52 \frac{gm}{cm^3}, (71 - 2 - 27\, mol\,\%)$$

4. FUSION FISSION HYBRID LIQUID METAL THORIUM BREEDER

A system consisting of fusion fuel factories using DT or Catalyzed DD fusion and fission satellites receiving the bred fissile fuel for burning is shown in Fig. 2.

A one dimensional calculational model considers a plasma cavity with a 150 cm radius. The plasma neutron source is uniformly distributed in the central 100 cm radial zone and is isolated from the first structural wall by a 50 cm vacuum zone (Table 1).

The blanket module consists of a 1 cm thick Type 316 stanless steel first structural wall that is cooled by a 0.5 cm thick water channel, a 42 cm thick molten salt filled energy absorbing and breeding compartment, and a 40 cm thick graphite neutron reflector (Table 2).

The molten salt and graphite are contained within 1 cm thick Type 316 stainless steel structural shells

5. BREEDING IN MOLTEN SALT

Computations were conducted using the one dimensional Discrete Ordinates transport ANISN code with a P_3 Legendre expansion and an S_{12} angular quadrature.

The catalyzed DD system exhibits a fissile nuclide production rate of 0.880 Th(n, γ) reactions per fusion source neutron. The DT system, in addition to breeding tritium from lithium for the DT reaction yields 0.737 Th(n, γ) breeding reactions per fusion source neutron.

Even though both approaches provide substantial energy amplification through the fusion-fission coupling process, the DT system possesses marginal tritium breeding in the fusion island of 0.467 triton per source neutron and would need supplemental breeding in the fission satellites to reach a value of unity (Table 3).

The largest Th(n,γ) reaction rate (0.966) occurs when the sodium salt is used in conjunction with the DT reaction. For this case, however, the tritium required to fuel the plasma must be supplied to the system, since that produced in the blanket would be negligible (3.18×10^{-3}). A system of such kind has been proposed and studied by Blinken and Novikov [13].

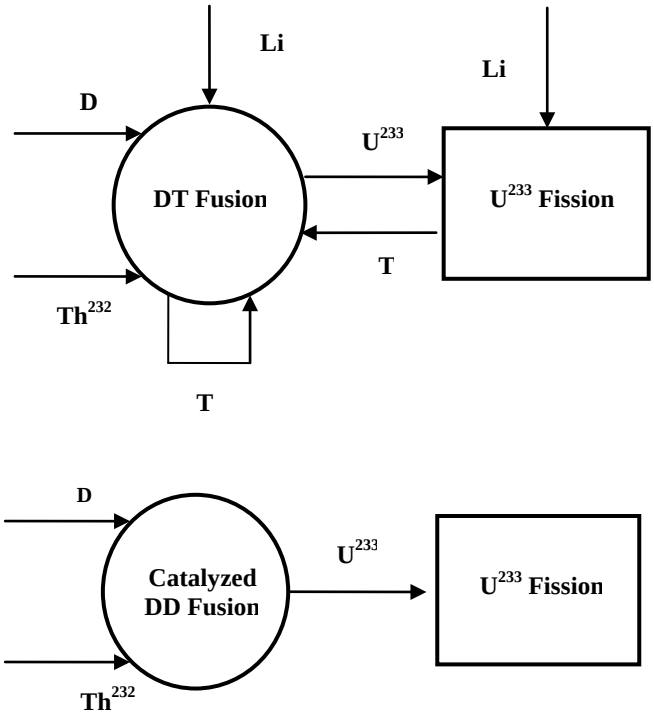

Figure 2. Material flows in the DT (top) and Catalyzed DD fusion-fission hybrid (bottom) alternatives with U^{233} breeding from Th^{232}. The Catalyzed DD approach does not contain the Li and T paths.

Table 1: Fusion-fission reactor geometrical model.

Material	**Zone**	**Outer Radius (cm)**	**Thickness (cm**	**Remarks**
Plasma	1	100.0	100.0	DT(14.06 MeV) or, Catalyzed DD (50 % 2.45 MeV + 50 % 14.06 MeV)
Void	2	150.0	50.0	Vacuum zone
First wall	3	151.0	1.0	Type 316 stainless steel
Water coolant	4	151.5	0.5	H_2O cooling channel
Structure	5	152.5	1.0	
Molten salt	6	194.5	42.0	$NaF.BeF_2.ThF_4$ or: $LiF.BeF_2.ThF_4$ ρ =4.52 gm/cm^3 (71-2-27 mol %)
Structure	7	195.5	1.0	Type 316 stainless steel
Neutron reflector	8	235.5	40.0	Graphite as C^{12}
Structure	9	236.5	1.0	Type 316 stainless steel
Albedo	10	-	-	20 percent albedo surface to simulate neutron and gamma ray reflection

Table 2: Fusion-fission material compositions.

Material	**Composition**	**Nuclide Density [nuclei/(b.cm)]**
1. $LiF.BeF_2.ThF_4$ salt ρ = 4.52 gm/cm^3 71-2-27 mol %	$_3Li^6$ $_3Li^7$ $_4Be^9$ $_{90}Th^{230}$ $_9F^{19}$	1.414×10^{-3} 1.744×10^{-2} 5.310×10^{-4} 7.169×10^{-3} 4.859×10^{-2}
2. $NaF.BeF_2.ThF_4$ salt ρ =4.52 gm/cm^3 71-2-27 mol %	$_{11}Na^{23}$ $_4Be^9$ $_{90}Th^{230}$ $_9F^{19}$	1.697×10^{-2} 4.799×10^{-4} 6.452×10^{-3} 4.373×10^{-2}
3. Type 316 stainless steel 63.6 wt% Fe, 18 wt% Cr, 13 wt% Ni, 2.6 wt% Mo, 1.9 wt% Mn, 0.9 wt% (Si+Ti+C) ρ = 7.98 gm/cm^3	C Si Ti Cr Mn Fe Ni Mo	1.990×10^{-4} 1.360×10^{-3} 4.980×10^{-5} 1.150×10^{-2} 1.650×10^{-3} 5.430×10^{-2} 1.060×10^{-2} 1.290×10^{-3}
4. Graphite ρ =2.25 gm/cm^3	C	1.128×10^{-1}
5. H_2O ρ = 1.0 gm/cm^3	H O	6.687×10^{-2} 3.343×10^{-2}

Table 3: Fissile and fusile breeding for sodium and lithium salts in DT and DD symbiotic fusion-fission fuel factories. Blanket thickness = 42 cm, reflector thickness =40 cm; no structure in the salt region.

Source	**Li-Be-Th-F Salt**						**Na-Be-Th-F Salt**			
	$Li^6(n,\alpha)T$	**$Li^7(n,n'\alpha)T$**	**$Be^9(n,T)$**	**F(n,T)**	**Total T**	**Th(n,γ)**	**$Be^9(n,T)$**	**F(n,T)**	**Total T**	**Th(n,γ)**
	(Nuclei / fusion source neutron)									
DD 100% 2.45 MeV	0.311	0.001	4.03×10^{-10}	1.01×10^{-7}	0.312	0.579	4.18×10^{-10}	1.04×10^{-7}	1.04×10^{-7}	0.794
DT 100% 14.06 MeV	0.391	0.073	1.08×10^{-4}	3.33×10^{-3}	0.467	0.737	1.04×10^{-4}	3.08×10^{-3}	3.18×10^{-3}	0.966
Catalyzed DD 50% 2.45 MeV 50% 14.06 MeV	0.351	0.037	5.40×10^{-5}	1.67×10^{-3}	0.390	0.658	5.20×10^{-5}	1.54×10^{-3}	1.59×10^{-3}	0.880

For the catalyzed DD plasma, the Th(n,γ) reaction rate is 0.880 in the sodium salt compared with 0.737 obtained in the DT system using the lithium salt. For the DT system the tritium production rate is 0.467 tritium nuclei per source neutron. The tritium production rate is too low to sustain the DT plasma, and an extra supply of the isotope is required to supplement the plasma. This could be internal by allowing some fissions to occur in the blanket supplying extra neutrons, or this could be external in the satellite fission reactors producing the needed difference as a byproduct and feeding it back to the DT plasma.

The fissile fuel production rate is higher in the DD system because of the absence of lithium in the salt that would introduce a competition between the Th(n,γ) and the $Li^6(n,T)$ reactions.

6. USE OF NEUTRON MULTIPLIERS

An approach to the fusion-fission hybrid is the fuel factory, where fissile fuel for fission reactors is bred in fusion reactors.

In this approach, there is a need to maximize the fissile breeding with the constraint of maintaining self sufficiency in tritium production and realistically accounting for the modeling for structural and coolant compositions and configurations imposed by the thermal hydraulic and mechanical designs.

For the DT fusion reaction cycle there appears a need for neutron multiplication using threshold neutron multiplication reactions such as from Pb (Fig. 3), Be (Fig. 4), and Bismuth (Fig. 5). Thorium itself in also provides neutron multiplication (Fig. 6).

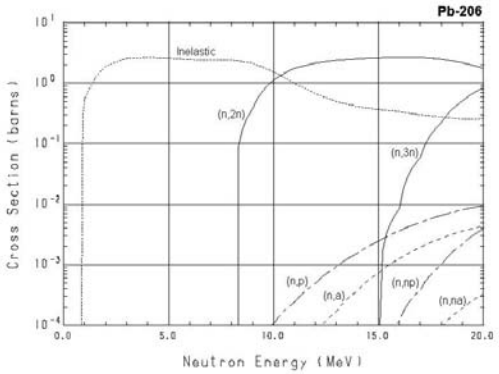

Figure 3. Neutron multiplication threshold reactions in Pb^{206}. Source: JENDL.

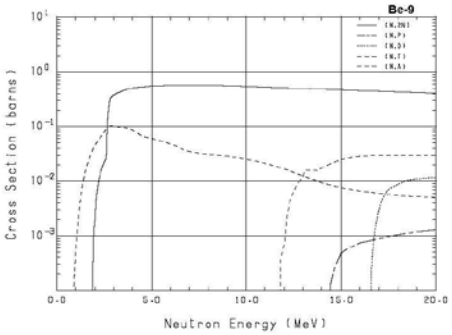

Figure 4. Neutron multiplication threshold reactions in Be^9. Source: JENDL.

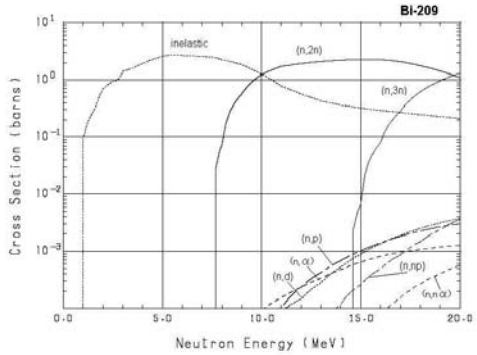

Figure 5. Neutron multiplication threshold reactions in Bi^{209}. Source: JENDL.

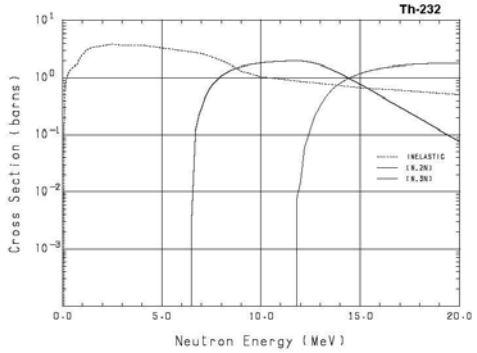

Figure 6: The cross section distribution for the (n, 2n) and n(3n) neutron multiplication reactions in Th^{232} showing energy thresholds at 6.465 and 11.61 MeV. Source: JENDL.

7. THE ENRICHMENT FACTORY APPROACH

The direct enrichment approach considers the use of fabricated ThO_2 Light Water Reactor (LWR) fuel assemblies and enriching them in a DT fusion neutron spectrum.

Figures 7 and 8 show the reactor configuration using a spherical DT fusion neutron source driven by an inertial confinement system such as a laser, electron or heavy ion beam. The reactor is composed of a right circular cylinder of 6.1 m radius referred to as the radial blanket and two end caps referred to as the axial blanket. The radial blanket is used for both fissile and fusile breeding, whereas the axial blanket is used solely for fusile breeding. The full 4π solid angle is used for neutron multiplication with lead as a neutron multiplier. A 0.25 m lead thickness precedes the axial blanket and reflects a fraction of the neutrons falling on it to the radial blanket. A graphite reflector follows both the axial and radial blankets. Zircaloy-4 is used as a structural material with the composition given in Table 4.

Tritium breeding occurs in natural lithium contained in Zircaloy-4 tubes and cooled by sodium. In the axial blanket, the main fusile breeding zone is emplaced between the reflector and the lead multiplying-reflecting zone. In the radial blanket, it precedes the reflector.

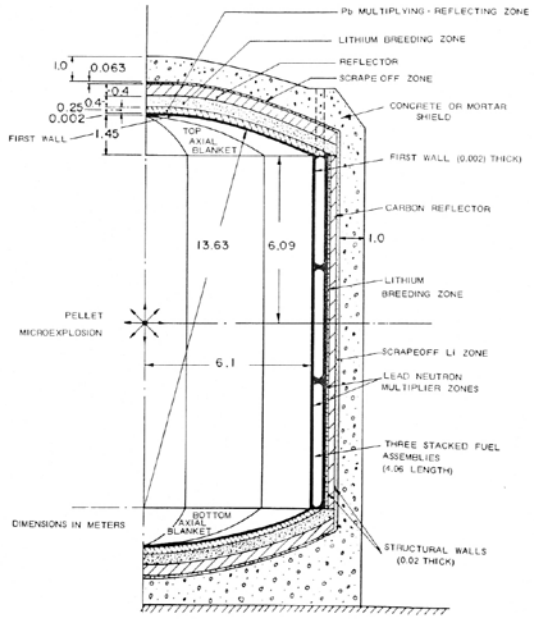

Figure 7. Laser fusion fissile generator plant with U^{233} breeding.

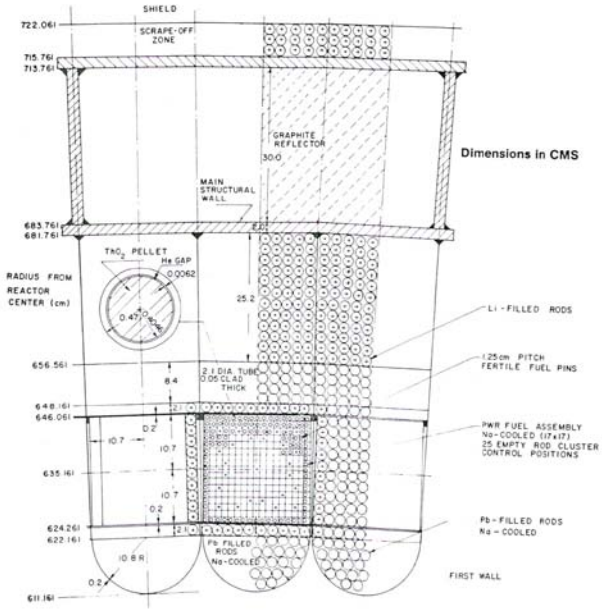

Figure 8. ThO_2 Pressurized Water Reactor fuel elements within the neutron multiplication zone, followed by a tritium breeding zone and a graphite reflector.

8. LEAD FAST FLUX TRAP

A lead fast-flux trap is used for fissile breeding, in which LWR reactor ThO_2 fuel assemblies are surrounded by lead from all sides. Between the lead and the assemblies a row of natural lithium tubing is emplaced to achieve a filtering action by absorbing slow neutrons. Maintaining a hard spectrum in the fuel assemblies will avoid the excessive fissioning of the bred U^{233}. In the lead flux trap, three LWR reactor assemblies are superimposed atop each other around the reactor circumference with a total number of 270.

The justification for the use of the flux trap is that due to radiation damage limitations to the first wall, a large cavity of 6.1 m radius is required. If the fuel assembles were positioned side by side around the reactor cavity, this results in a large fertile inventory and a subsequent long time to attain a given average enrichment. However, since the number of neutrons in the system is constant, the use of the flux trap was shown to reduce the fuel inventory and the time to attain a given enrichment at the expense of a slight reduction in the overall fissile fuel production.

In a blanket composition not using the flux trap, 2.87 metric tons of U^{233} would be produced per year, and the time to attain an average enrichment of 4 percent is 3.47 years. When the flux trap is used, 2.18 metric tons can be produced yearly, and the time to attain an average 4 percent enrichment is a shorter 2.27 years, together with a reduction in the fuel inventory of 50 percent.

9. CALCULATIONAL AND GEOMETRICAL MODELS

The unit cell shown in Figs. 24 and 25 was developed for the three-dimensional analysis using Monte Carlo particle transport calculations. An albedo surface surrounded the cell, and the source neutrons were sampled isotropically at the lower left edge of the cell as shown in Fig. 24. The material compositions were homogenized according to the volume fractions and compositions of Table 4.

The optimization of the fusile and fissile breeding by altering the material thicknesses of the radial and axial blanket regions is shown in Fig. 11.

10. BREEDING IN SOLID FUEL

A tritium yield per source neutron of 1.08 and a Th (n, γ) reaction yield of 0.43 can be obtained in the concept where ThO_2 Zircaloy-clad fuel assemblies for Light Water Reactors (LWRs) are enriched in the U^{233} isotope by irradiating them in a Pb flux trap.

This corresponds to 0.77kg/[MW(th).year] of fissile fuel production, and 1.94 years of irradiation in the fusion reactor to attain an average 3 w/o fissile enrichment in the fuel assemblies (Fig. 11).

For a once through LWR cycle, a support ratio of 2-3 is estimated. However, with fuel recycling, more attractive support ratios of 4-6 may be attainable for a conversion ratio of 0.55, and 5-8 for a conversion ratio of 0.70.

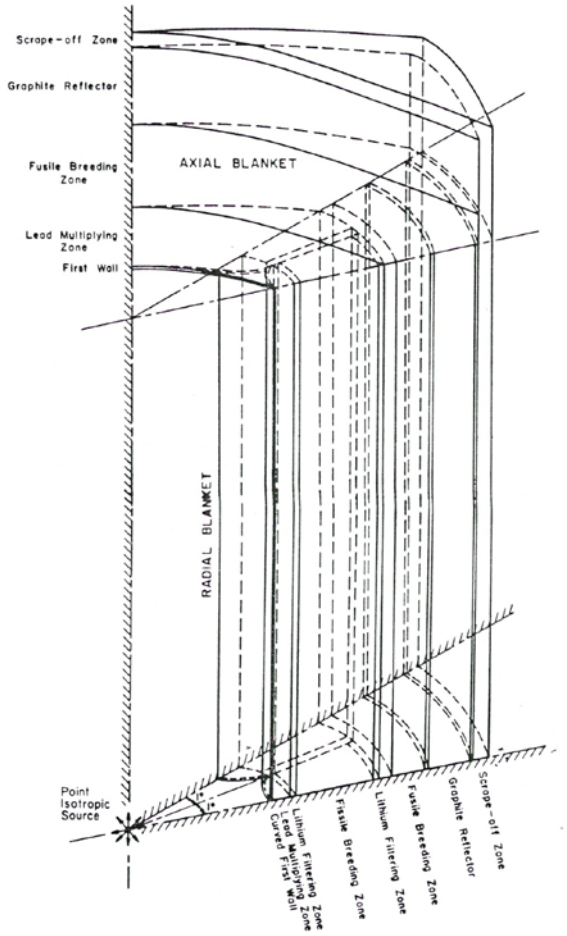

Figure 9. Three dimensional unit cell of computational model of laser fusion hybrid plant.

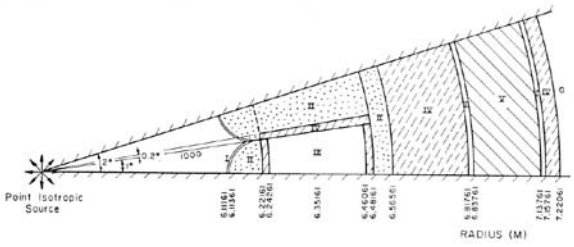

Figure 10. Horizontal cut through unit cell of three dimensional computational model.

Table 4: Material compositions and number densities.

Material composition	Element	Atomic densities [atoms/(barn·m)]
1. First wall and structural wall		
100 v/o Zircaloy-4	Zr	4.374+0
98.24 w/o Zr+1.5 w/o Sn	Sn	4.962−2
+0.21 w/o Fe	Cr	7.812−3
+0.10 w/o Cr	Fe	1.527−2
ρ(Zircaloy-4)=6.745×10^3 kg/m^3		
2. Reflector	^{12}C	8.373+0
100% Reactor-grade graphite		
ρ(graphite)=1.67×10^3 kg/m^3		
3. Neutron multiplication zones	Pb	2.145+0
65.03 v/o Pb+844 v/o	Zr	3.692−1
Zircaloy-4+26.53 v/o	Sn	4.188−3
Na coolant.	Cr	6.593−4
ρ(Pb)=11.35×10^3 kg/m^3	Fe	1.289−3
ρ(Na)=9.71×10^2 kg/m^3	Na	6.748−1
4. Fusile breeding zones	^{6}Li	2.364−1
68.78 v/o natural lithium	^{7}Li	2.950+0
+7.97 v/o Zircaloy-4	Zr	3.486−1
+23.25 v/o Na Coolant	Sn	3.955−3
ρ(Li)=0.534×10^3 kg/m^3,	Cr	6.226−4
7.22 a/o ^{6}Li+92.58 a/o	Fe	1.217−3
^{7}Li	Na	5.914−1
5. Fissile breeding zone	Th	6.415−1
28.10 v/o ThO_2+10.47 v/o	^{16}O	1.283+0
Zircaloy-4	Zr	4.580−1
+60.98 v/o Na Coolant	Sn	5.195−3
+1.15 v/o He Fill Gas	Cr	8.179−4
ρ(ThO_2)=10.01×10^3 kg/m^3	Fe	1.599−3
	Na	1.533+0

9. DISCUSSION

The thorium fusion fission hybrid is discussed as a sustainable longer term larger resource base to the fast breeder fission reactor concept. In addition, it offers a manageable waste disposal process, burning of the produced actinides and serious nonproliferation characteristics.

With the present day availability of fissile U^{235} and Pu^{239}, and available fusion and accelerator neutron sources, a new look at the thorium-U^{233} fuel cycle is warranted. Since no more than 7 percent of the ThO_2 as a breeding seed fuel can be added to a Heavy Water Reactor, HWR before criticality would not be achievable; this suggests that fusion and accelerator sources are the appropriate alternative for the implementation of the Thorium fuel cycle.

The use of the thorium cycle in a fusion fission hybrid could bypass the stage of fourth generation breeder reactors in that the energy multiplication in the fission part allows the satisfaction of energy breakeven and the Lawson condition in magnetic and inertial fusion reactor designs [19-24]. This allows for the incremental development of the technology for the eventual introduction of a pure fusion technology.

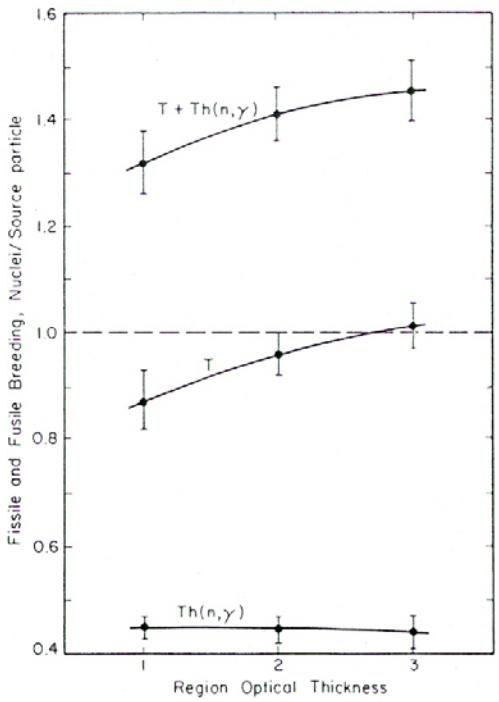

Figure 11. Optimization of fissile U^{233} and fusile tritium (T) breeding.

A catalyzed or semi-catalyzed DD fusion cycle will not need tritium breeding. Both the DT and Catalyzed DD approaches provide substantial energy amplification through the fusion-fission coupling process. However, the DT system possesses marginal tritium breeding in the fusion island and would need supplemental breeding in the fission satellites to reach a value of the tritium breeding ratio unity [1-4].

As a first generation cycle, a DT fusion cycle needs serious consideration of its tritium breeding potential. Enhancement of the fusile breeding would require the use of neutron multipliers and optimized flux trap geometrical configurations.

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