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The Fusion-Fission Thorium Hybrid

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ABSTRACT

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

Historically, the thorium fission fuel cycle was investigated over the period 1950-1976 in the Molten Salt Breeder Reactor (MSBR) at the Oak Ridge National Laboratory (ORNL) as well as in the pilot Shippingport fission reactor plant. It has also been used in the High Temperature Gas Cooled Reactor (HTGR) in a pebble bed and a prismatic moderator and fuel configurations. The General Atomics (GA) Company built two thorium reactors over the 1960-1970 period. The first was a 40 MWe prototype at Peach Bottom, Pennsylvania operated by Philadelphia Electric. The second was the 330 MWe Fort St. Vrain reactor for Public service of Colorado which operated between 1971 and 1975.

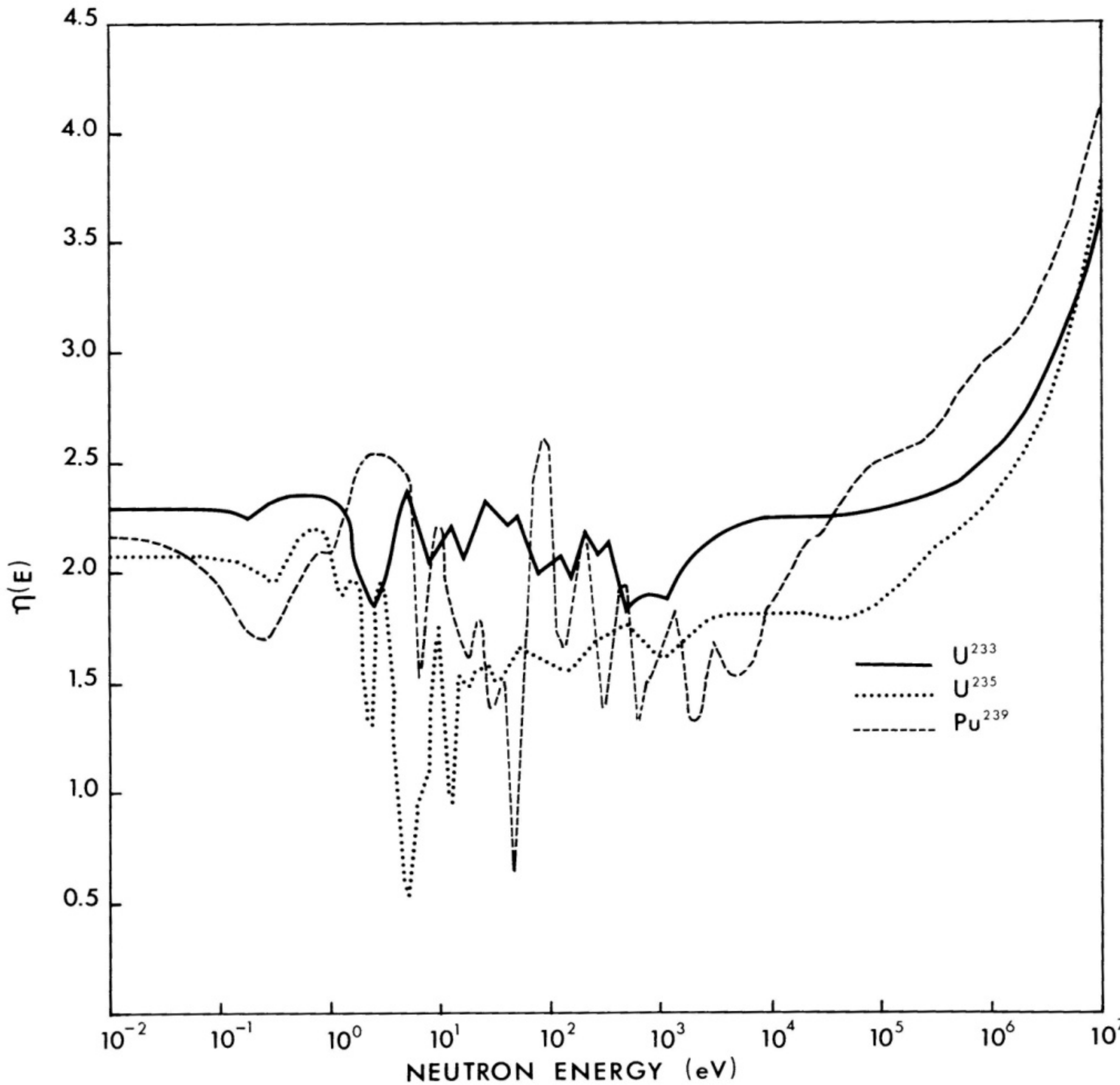
With the present day availability of fissile U^{235} and Pu^{239}, and available fusion and accelerator neutron sources, a fresh look at the thorium cycle is ongoing. Whereas the U^{233}-Th^{232} fuel cycle is undergoing a revival as a replacement of the existing Light Water Reactors (LWRs) system, a highly promising approach is its use in fusion-fission hybrid reactors as an eventual bridge and technology development for future pure fusion reactors, bypassing the intermediate stage of the fast fission breeder reactors. We discuss the possibility of taking advantage of the Th cycle benefits in the form of an optimized fission-fusion thorium hybrid.

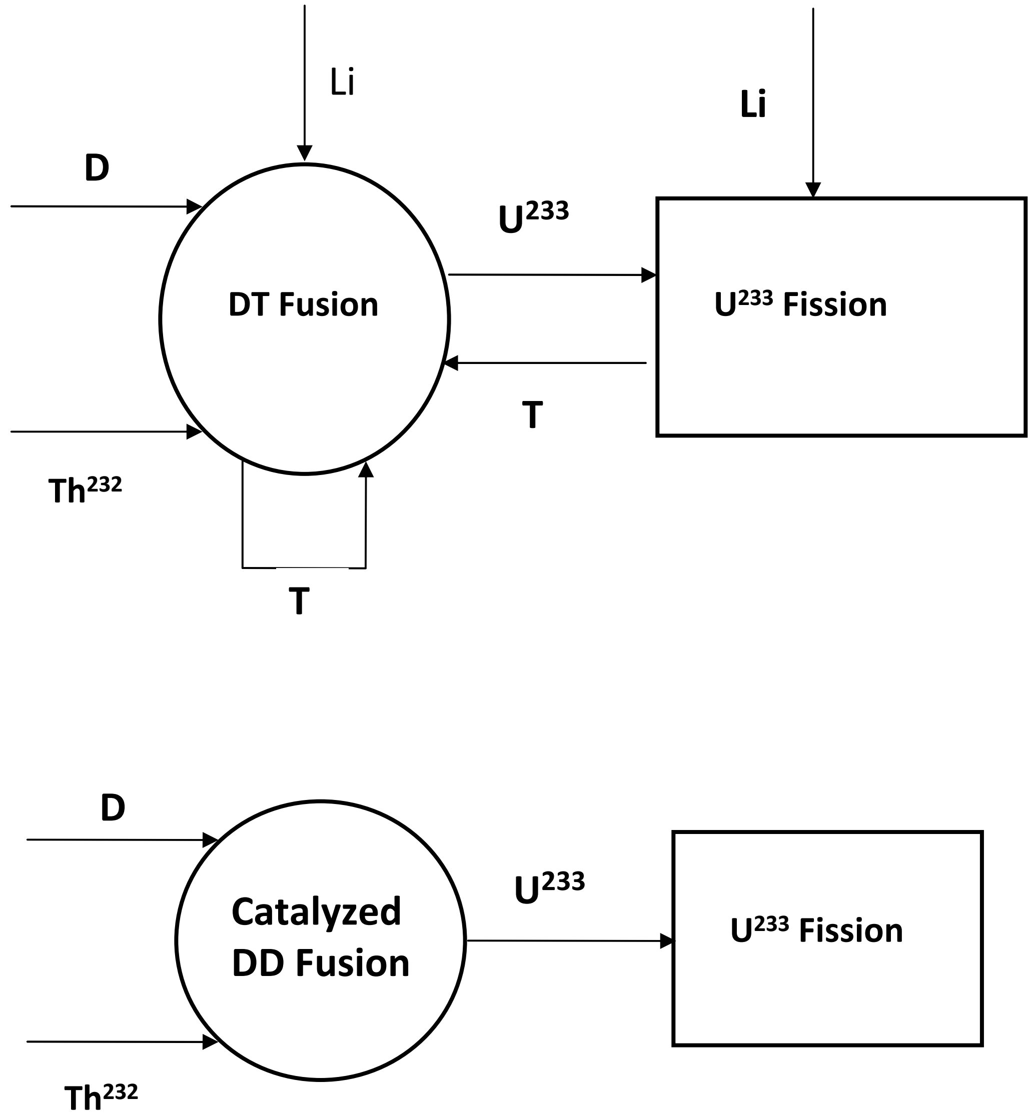
The nuclear performance of a fusion-fission hybrid reactor having a molten salt composed of Na-Th-F-Be as the blanket fertile material and operating with a catalyzed Deuterium-Deuterium (DD) plasma is compared to a system with a Li-Th-F-Be salt operating with a Deuterium-Tritium (DT) plasma. In a reactor 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. Both approaches provide substantial energy amplification through the fusion-fission coupling process.

In a fuel factory concept using a DT fusion source, a tritium yield per source neutron of 1.08 and a Th (n, γ) reaction yield of 0.43 can be obtained whereas 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. 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.

Such an alternative sustainable paradigm would provide the possibility of an optimized fusion-fission thorium hybrid for long term fuel availability with the added advantages of higher temperatures thermal efficiency for process heat production, proliferation resistance and minimized waste disposal characteristics.

Regeneration factor as a function of neutron energy for the different fissile isotopes. Breeding in the Thorium-U^{233} fuel cycle can be achieved with thermal or fast neutrons.





Material flows in the DT (top) and Catalyzed DD fusion-fission hybrid (bottom) Fuel Factory alternatives with U^{233} breeding from Th^{232}.

For a first generation application of the fusion hybrid using the Th cycle, the DT fusion fuel cycle can be used

$$ {}_{1}D^{2} + {}_{1}T^{3} \rightarrow {}_{2}He^{4}(3.52\ \text{MeV}) + {}_{0}n^{1}(14.06\ \text{MeV}) + 17.58\ \text{MeV} $$

Deuterium can be obtained from heavy water D_2O separated from ordinary water H_2O.
Tritium (T) must be bred from abundant supplies of lithium as feed to the DT fusion reaction.

$$ {}_{3}Li^{6} + {}_{0}n^{1}(\text{thermal}) \rightarrow {}_{2}He^{4}(2.05\ \text{MeV}) + {}_{1}T^{3}(2.73\ \text{MeV}) + 4.78\ \text{MeV} $$

$$ {}_{3}Li^{7} + {}_{0}n^{1}(\text{fast}) \rightarrow {}_{2}He^{4} + {}_{0}n^{1} + {}_{1}T^{3} - 2.47\ \text{MeV} $$

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

:

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

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

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

Fusion-fission material compositions

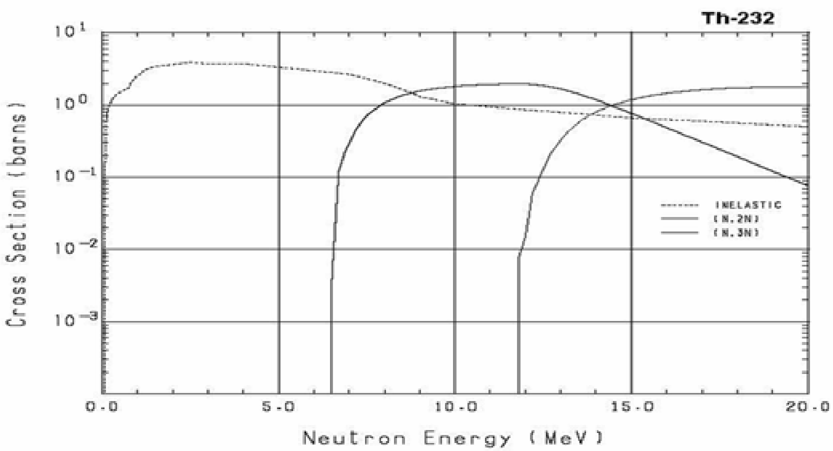
Material	Composition	Nuclide Density [nuclei/(b.cm)]
1. $LiF.BeF_2.ThF_4$ salt ρ = 4.52 gm/cm^3 71-2-27 mol %	$_3Li^6$ $_3Li^7$ $_4Be^9$ $_{90}Th^{230}$ $_9F^{19}$	1.414×10^{-3} 1.744×10^{-2} 5.310×10^{-4} 7.169×10^{-3} 4.859×10^{-2}
2. $NaF.BeF_2.ThF_4$ salt ρ =4.52 gm/cm^3 71-2-27 mol %	$_{11}Na^{23}$ $_4Be^9$ $_{90}Th^{230}$ $_9F^{19}$	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}

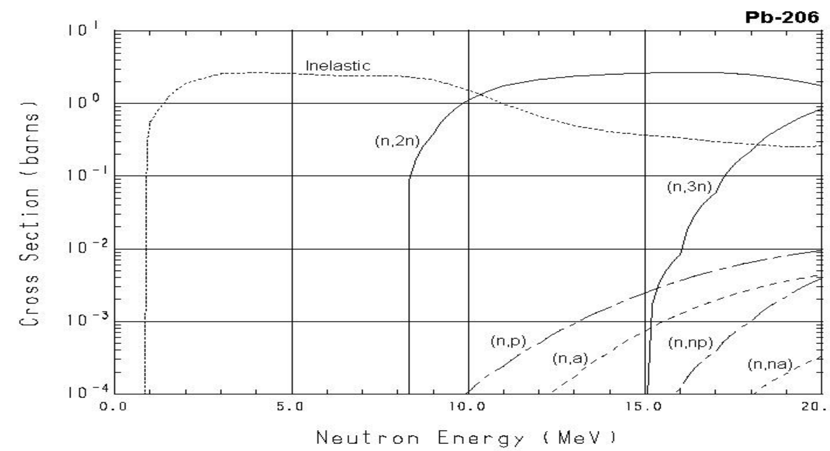
Fissile and fusile breeding for sodium and lithium salts in DT and DD symbiotic fusion-fission fuel factories. Blanket thickness = 42 cm, reflector thickness =40 cm; no structure in the salt region.

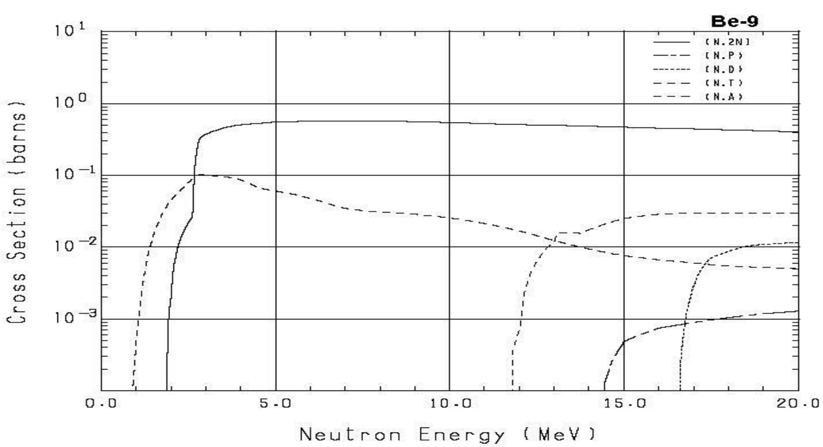
Source	Li-Be-Th-F Salt						Na-Be-Th-F Salt			
	$Li^6(n,\alpha)T$	$Li^7(n,n'\alpha)T$	$Be^9(n,T)$	F(n,T)	Total T	$Th(n,\gamma)$	$Be^9(n,T)$	F(n,T)	Total T	$Th(n,\gamma)$
	(Nuclei / fusion source neutron)									
DD 100% 2.45 MeV	0.311	0.001	4.03×10^{-10}	1.01×10^{-7}	0.312	0.579	4.18×10^{-10}	1.04×10^{-7}	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

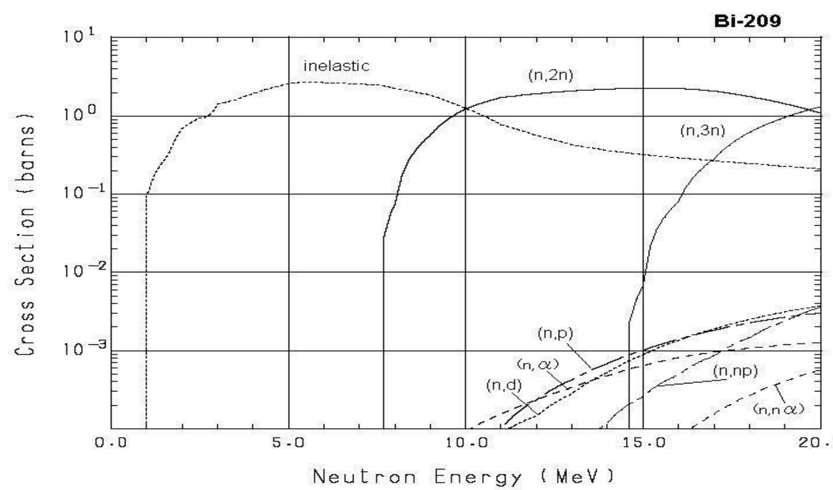
Neutron Multiplication

The cross section distribution for the (n, 2n) and n(3n) neutron multiplication reactions in Th^{232} shows energy thresholds at 6.465 and 11.61 MeV. Other candidate neutron multipliers are Pb, Be, Bi and U.

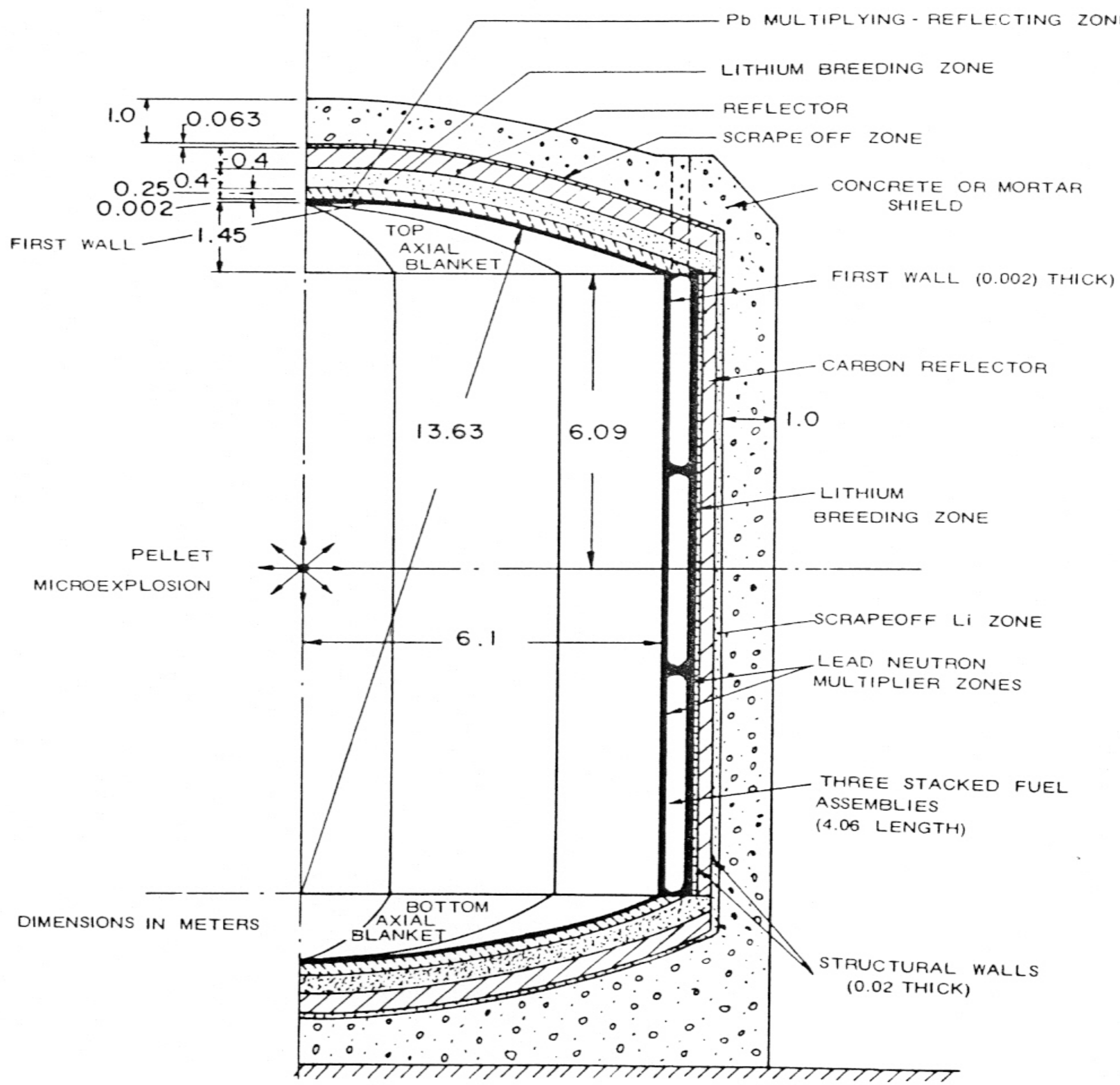




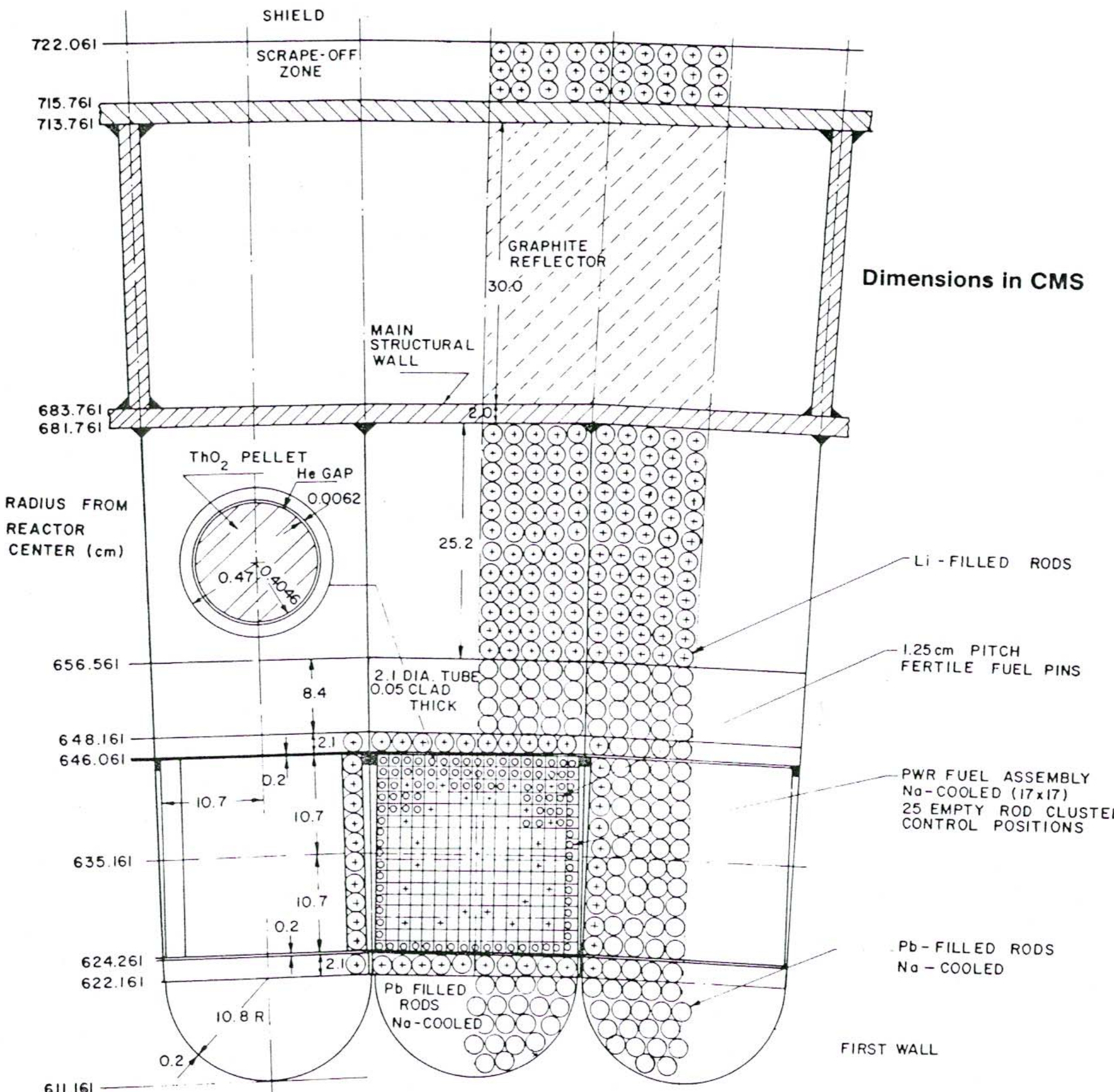




Laser fusion fissile generator plant with U^{233} breeding.



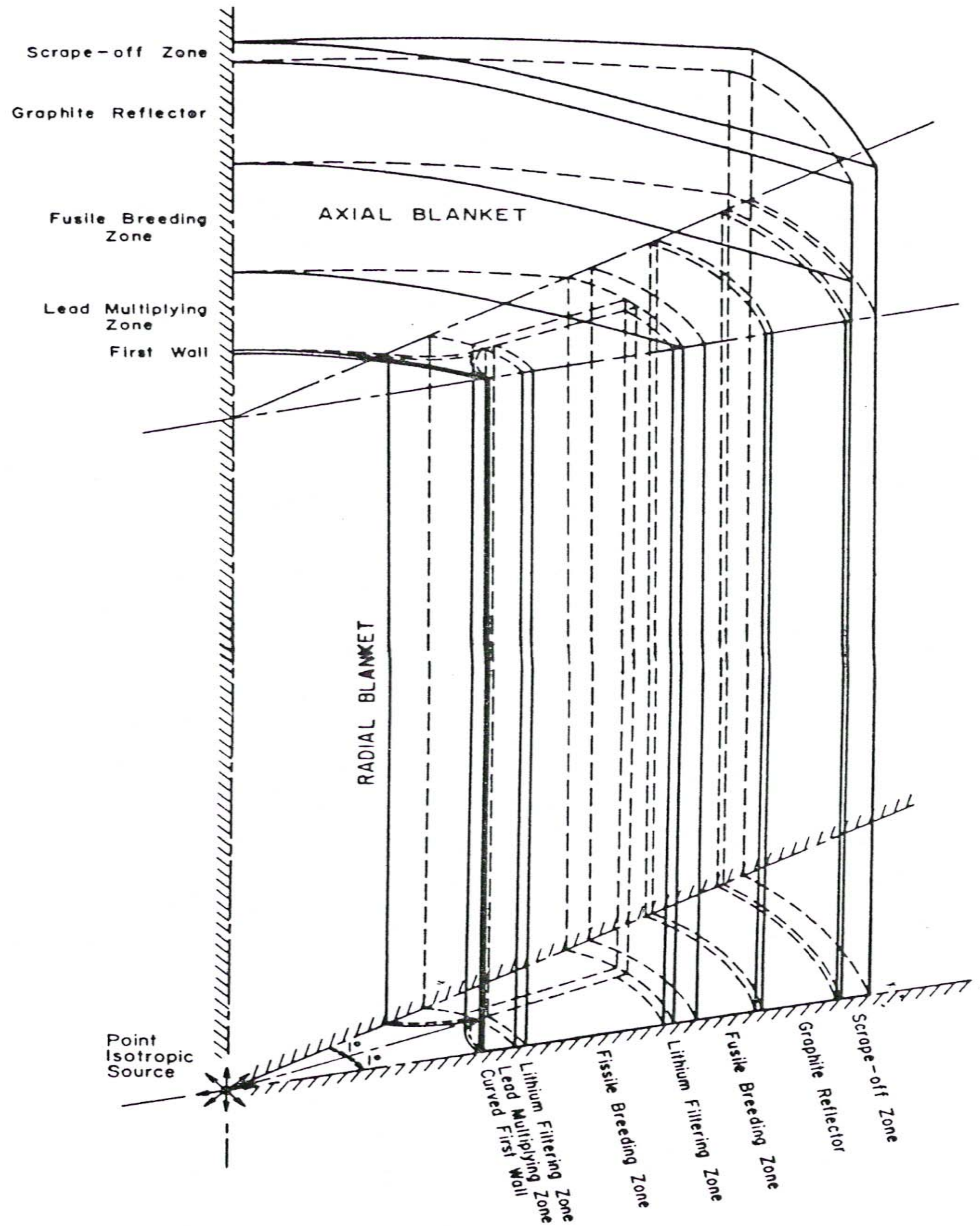
ThO_2 Pressurized Water Reactor fuel elements within a flux trap neutron multiplication zone, followed by a tritium breeding zone and a graphite reflector.



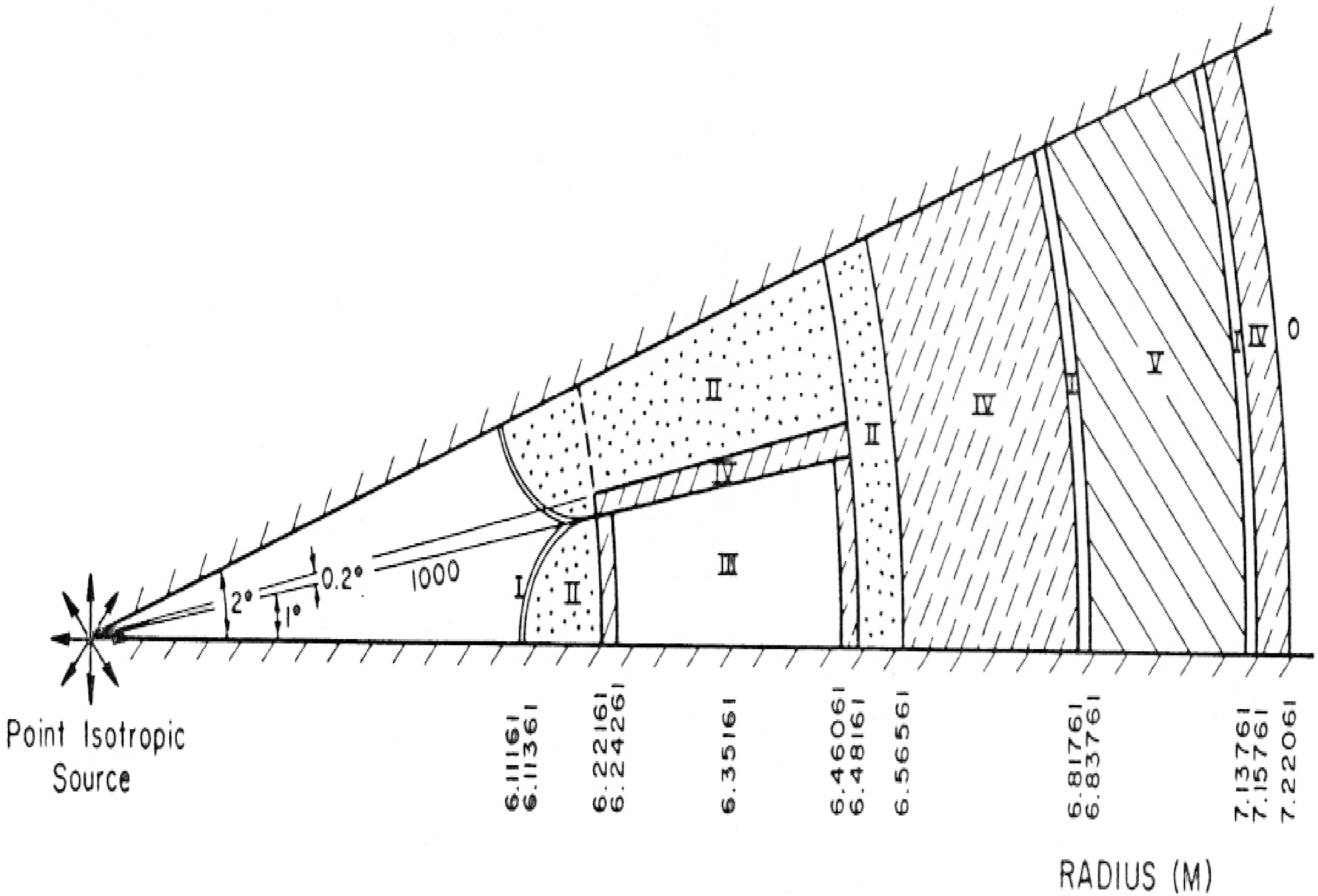
Fusion fissile generator plant with U^{233} breeding. Material compositions.

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	^6Li	2.364−1
68.78 v/o natural lithium	^7Li	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 ^6Li+92.58 a/o	Fe	1.217−3
^7Li	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
$\rho(ThO_2)$=10.01×10^3 kg/m^3	Fe	1.599−3
	Na	1.533+0

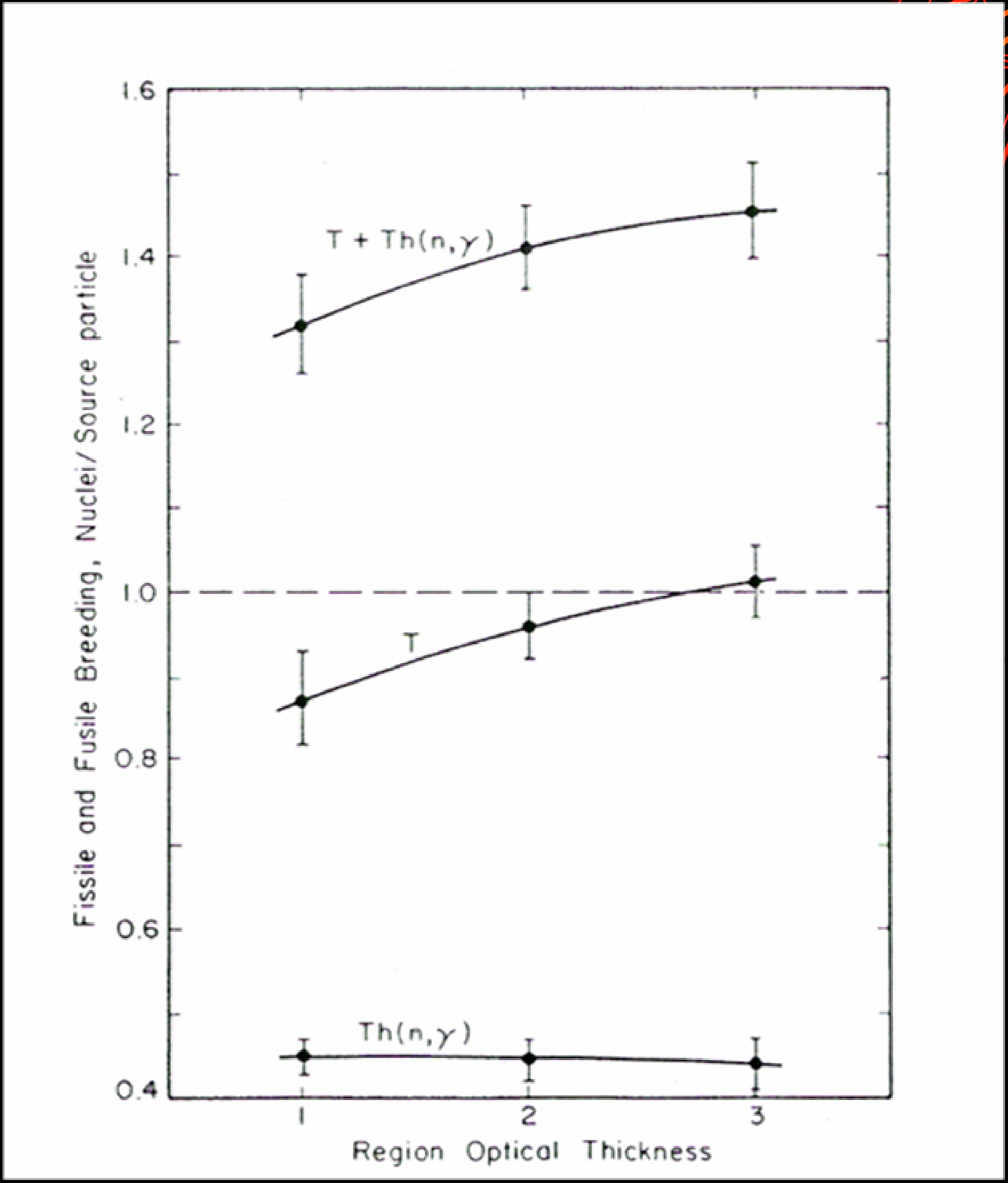
Laser fusion fissile generator plant with U^{233} breeding.



Horizontal cut through unit cell of three dimensional lead flux trap computational model.



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



DISCUSSION

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 system.

As a proof of principle, a compact experimental device can be built using inertial electrostatic confinement with DD or DT fusion with a cusped configuration or a grid diode, and coupled to a molten salt breeding blanket at a university or a national laboratory site.

Such an alternative sustainable paradigm or architecture would provide the possibility of a well optimized fusion-fission thorium hybrid for sustainable long term fuel availability with the added advantages of higher temperatures thermal efficiency for process heat production, proliferation resistance and minimized waste disposal characteristics.

SUMMARY

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

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 cycle is ongoing. Whereas the U^{233}-Th^{232} fuel cycle is undergoing a revival as a replacement of the existing Light Water Reactors (LWRs) system, a highly promising approach is its use in fusion-fission hybrid reactors as an eventual bridge and technology development for future pure fusion reactors, bypassing the intermediate stage of the fast fission breeder reactors. We discuss the possibility of taking advantage of the Th cycle benefits in the form of an optimized fission-fusion thorium hybrid.

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 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 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. Both approaches provide substantial energy amplification through the fusion-fission coupling process.

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