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^{233}U BREEDING IN A MODIFIED PROMETHEUS-L IFE REACTOR

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Abstract- In this study, neutronic analysis of a hybrid version of the PROMETHEUS-L IFE reactor is investigated by using thorium fuels namely, ThN, ThC_2 and ThF_4. Calculations of neutronic data per DT fusion neutron are performed by using SCALE 4.3 Code. Despite a partial replacement of the tritium breeding zone by the fissile fuel breeding zone, tritium breeding remains still > 1.05, which will be required for a self-sustaining fusion driver. Energy multiplication 1.44 to 1.50 depending on the fuel types in the blanket. Therefore, the investigated reactor can produce substantial electricity *in situ*. Substantial amount of fissile material will be produced at start-up conditions with a fissile fuel breeding rate of ^{233}U = 0.154, 0.126 and 0.077 for ThN, ThC_2 and ThF_4 per incident fusion neutron, which correspond to 1858, 1521, 934 kg ^{233}U/year, respectively, by a full fusion power of 2807 MW.

Keywords- Hybrid reactor, fissile fuel breeding, thorium.

1. INTRODUCTION

Two systems, main rich neutron sources as fissile fuel producers, have been investigated to produce nuclear fuel required for production of energy in the conventional nuclear reactors and transmutation of nuclear wastes. These are:

1) Fusion reactors based on (D,T) or (D,D) reactions,
2) Electro-nuclear breeders based on principle of spallation of heavy nucleus under bombardment of relatively high energetic protons (~1 GeV)

The idea of production of abundant fissile fuel by using fusion breeders or electro-nuclear breeders is quite old [1-7]. The studies show that a fusion breeder can produce up to 30 times more fissile fuel than in a FB (Fast Breeder) per unit of energy [1]. Hybrid reactor is a combination of fusion and fission processes that surrounds the fusion plasma with a blanket containing fertile materials (^{232}Th and/or ^{238}U) and lithium for producing fissile fuel and tritium, respectively. Moreover, the fertile materials may also undergo a significant amount of fission and waste actinides can be burnt effectively under highly energetic fusion neutrons. Therefore, fusion energy breeder reactors are alternative by using thorium reserves, natural uranium and spent fuel containing a substantial amount of ^{238}U for producing fissile fuel and electricity *in situ*, and burn all nuclear waste actinides, such as, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Am, ^{243}Am and ^{244}Cm. Many research and development studies have been done on inertial confinement fusion power reactors. These systems can use fertile fuels and spent fuels without fissile fuel requirement. The blanket concept should be adaptable to most nuclear applications, including the use of natural uranium, LWR (Light Water Reactor) or CANDU (Canada Deuterium-Uranium Reactor) spent fuel or thorium as fertile materials. The capability of the production of fissile fuel in a fusion energy breeder reactor has been studied in earlier works [1,3-6,8-43].

The PROMETHEUS-L (Laser) and PROMETHEUS-H (Heavy Ion) IFE (Inertial Fusion Energy) power plant designs were completed by Department of Energy in USA in early 1992. The PROMETHEUS-L design would use a direct-drive target, driven by a KrF laser, whereas the PROMETHEUS-H design would use an indirect-drive target, driven by 4 GeV lead ions. These two designs are nearly identical from the consideration of target chambers [44]. The PROMETHEUS-L/-H IFE design was strongly based on considerations of safety, reliability, simplicity, and flexibility. Safety considerations led to the choice of a helium-cooled, solid breeder blanket with low activation materials-SiC structure and neutron reflector and Li_2O breeder [45-50]. The blanket also uses SiC structure as reflector, with low-activation Li_2O breeder and He coolant. The tritium inventory in the breeder was minimized. PROMETHEUS uses an innovative, highly-effective shield consisting of Al structure, water coolant, and B_4C, Pb, and SiC absorbers as the shield material instead of concrete for reducing activation [51,52].

2. BLANKET GEOMETRY

The PROMETHEUS IFE uses a thin liquid Pb film supplied from Pb coolant tubes through a porous structure of SiC composite material to protect the first wall. Thickness of film nominally is 0.5 mm that is allowed to form on the surface facing the pellet explosions. The first wall system consists of a series of panels 2-m wide which are lowered into the cavity vertically and locked into a support system attached to the blanket [52]. The blanket of which material compositions and dimensions are given in table 1 contains several rings through the cylindrical and hemispherical sections.

Blanket modules are pre-assembled into the rings, which stack vertically on top of one another. SiC is used to make blanket modules and contains a number of U-bend woven SiC tube sheets inside which the pressurized He coolant flows. The Li_2O that is placed in packed bed form between the tube sheets and is purged by He flowing along the axis of the module, in conjunction with the first wall Pb coolant supplies the potential for adequate tritium breeding without the need for Be as a multiplier. SiC has been selected as the structural material in the first wall and blanket/reflector/plena regions due to the high neutron reflectivity property of carbon. SiC has lower absorption cross-section for neutrons than stainless steels [44].

Table 1. Material composition and dimension of the zones of the blanket

Zone	Material	Dimension [cm]	Fraction [%]	Material	Density [g/cm^3]
Cavity	Vacuum	450.00		He	0.00715
Pb Film	Pb	0.05	100	Li_2O	2.013
First Wall	SiC	0.50	90	SiC	3.20
	Pb		10	Pb	11.34
	SiC	5.00	10	ThN	9.86
	Pb		90	ThC_2	9.60
				ThF_4	6.20
	SiC	0.50	100		
Gap	Vacuum	3.95			
Blanket Wall	SiC	2.50	84		
	He		16		
Tritium Breeding	Li_2O	15.00	44[a]		
	SiC		22		
	He		34		
Fissile Fuel Breeding	Fuel[b]	12.00	44[c]		
	SiC		22		
	He		34		
Tritium Breeding	Li_2O	33.00	44[a]		
	SiC		22		
	He		34		
Reflector	SiC	20.00	90		
	He		10		
Plena	SiC	17.50	10		
	He		90		
Blanket Wall	SiC	4.00	100		

[a]80 % Theoretical density, [b]ThN, ThC_2, ThF_4
[c]76 % Fuel, 3 % Clad (SiC) and 21 % Coolant (He)

In this study, the tritium breeding zone of PROMETHEUS-H IFE reactor having thickness of 60 cm, is divided into three parts as 15 cm, 12 cm and 33 cm. Then, in order to breed fissile fuel from fertile fuel, FFB (Fissile Fuel Breeding) zone containing fuel spheres filled with compounds of thorium and cladded with SiC, is located instead of the second part of the tritium breeding zone having a thickness of 12 cm (see Figure 1). Inner and outer diameters of the fuel spheres are 1.1 and 1.12 cm, respectively and they are arranged as hexagonal in ten rows having pitch length=1.2 cm in radial direction. Different type of fuels namely, ThN, ThC_2, ThF_4, are considered in the FFB to investigate the neutronic performance of the blanket. Helium is used as a coolant and an energy carrier in the FFB.

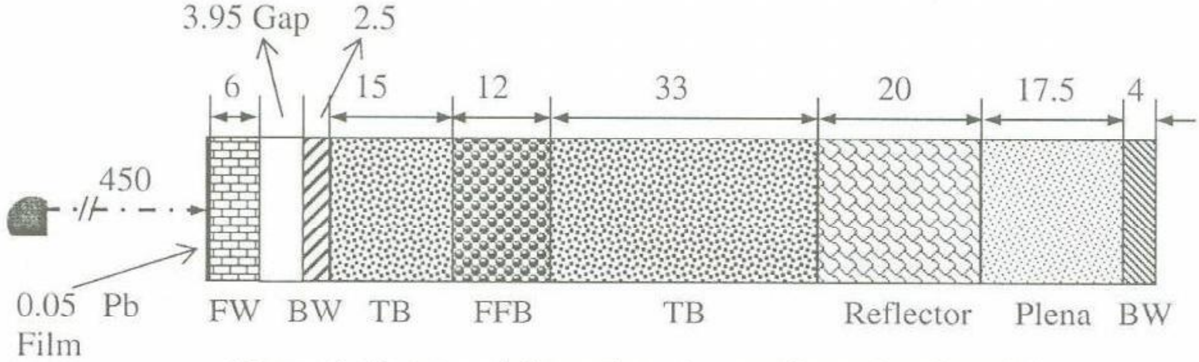

Figure 1. Zones and dimensions in one-dimensional model (FW=First Wall, BW=Blanket Wall, TB=Tritium Breeding, FFB=Fissile Fuel Breeding), (dimensions are given in cm).

3. NUMERICAL RESULTS

3.1. Calculation method

Determination of neutron flux, ϕ, is very important for neutronic calculations. Using Boltzmann transport equation is a common way to calculate neutron flux distribution in a reactor which is given in equation 1.

$$\frac{1}{v}\frac{\partial}{\partial t}\phi(r,E,\Omega,t) = S(r,E,\Omega,t) + \int_{E'}\int_{\Omega'}\sum\nolimits_s(r,E' \to E,\Omega' \to \Omega)\phi(r,E,\Omega',t)dE'd\Omega'$$
$$-\Omega\nabla\phi(r,E,\Omega,t) - \sum\nolimits_T(r,E)\phi(r,E,\Omega,t) \tag{1}$$

Terms in the first equation for dE and $d\Omega$ around E and Ω, respectively can be defined as follows:

$\frac{1}{v}\frac{\partial}{\partial t}\phi(r,E,\Omega,t)dEd\Omega =$ Change of neutron flux in unit time

$S(r,E,\Omega,t) =$ Contribution of source and fission neutrons on neutron flux

$\left[\int_{E'}\int_{\Omega'}\sum\nolimits_s(r,E' \to E,\Omega' \to \Omega)\phi(r,E',\Omega',t)d\Omega'dE'\right]dEd\Omega =$ Contribution of neutrons on neutron flux due to scattering

$\Omega\nabla\phi(r,E,\Omega,t)dEd\Omega =$ Neutron loss because of convection

$\sum\nolimits_T(r,E)\phi(r,E,\Omega,t)dEd\Omega =$ Neutron loss due to nuclear reactions between environmental atoms

Neutron transport calculations are conducted with the help of SCALE4.3 System by solving the Boltzmann transport equation with code XSDRNPM [53] in S_8-P_3 approximation with Gaussian quadratures [54] using the 238 groups library, derived from ENDF/B-V [55]. The resonance calculations in the fissionable fuel element cell are performed with

- BONAMI [56] for unresolved resonances and
- NITAWL-II [57] for resolved resonances.

CSAS control module [58] is used to produce the resonance self-shielded weighted cross-sections for XSDRNPM. The numerical output of XSDRNPM is processed with XSCALC [59] to evaluate neutronic performance of the blanket.

3.2. Neutronic performance of the overall blanket

Integral neutronic data per DT fusion neutron in the overall investigated blanket for different fuel types are given in table 2. The energy multiplication factor, M, is defined as the ratio of the total energy release in the blanket to the incident fusion neutron energy. Total energy release in blanket can be calculated as

$$\text{Total energy release in blanket} = 200 \cdot \Sigma_f + 4.784 \cdot T_6 - 2.467 \cdot T_7 \qquad (2)$$

Table 2. Neutronic data per DT fusion neutron in the overall blanket

	ThN	ThC_2	ThF_4
M	1.50	1.48	1.44
$^{232}Th_\gamma$	0.154	0.1261	0.0774
$^{232}Th_f$	0.0089	0.0072	0.0038
$\nu\Sigma_f$	0.0288	0.0232	0.01241
T_6	1.12	1.15	1.17
T_7	0.06	0.06	0.06
TBR	1.18	1.21	1.23
Γ	1.50	1.46	1.48
L	0.0214	0.0217	0.0199

M = Energy Multiplication Ratio [Total Energy Release (MeV)/14.1 MeV+1],
$^{232}Th_\gamma = {}^{233}U$ Breeding Ratio,
$^{232}Th_f = {}^{232}Th(n,f)$,
$\nu\Sigma_f$ = Fission Neutron Breeding Ratio,
$T_6 = {}^6Li(n,\alpha)T$,
$T_7 = {}^7Li(n,n'\alpha)T$,
TBR = Tritium Breeding Ratio,
Γ = Peak-to-Average Fission Power Density Ratio,
L = Neutron Leakage

M values are found as 1.50, 1.48, and 1.44 for ThN, ThC_2 and ThF_4, respectively due to the fact that low fission values are reached in the blanket. Figure 2 depicts fission rate per cm^3 per (D,T) fusion neutron in the FFB zone for these fuels. The highest fission rate is reached in the blanket fueled with ThN whereas the lowest one is obtained in the blanket fueled with ThF_4. One can see from this figure that fission rates decline sharply in radial direction towards to outer end of the blanket since the fusion neutron energy decreases by deeper penetration in the blanket. The fast neutron fluxes decrease in the radial direction while the lower energy group fluxes increase because they are generated mainly beyond the fuel zone and reflected back. In other words, the neutron flux curves show a variation towards the outer boundary from the harder neutron spectrum shapes to the softer ones.

FFBR (Fissile Fuel Breeding Ratio) per cm^3 per DT fusion neutron in the FFB zone of the blanket for three different fuels is shown in figure 3. The ^{233}U fissile fuel can be produced from $^{232}Th(n,\gamma)$. ^{233}U breeding capability of ThN fuel is higher than that of either ThC_2 or ThF_4 in the blanket and FFBR values for these fuels decrease towards the outer end of the blanket. Calculations give a fissile fuel breeding rate of 0.154, 0.126 and 0.077 for ThN, ThC_2 and ThF_4, per incident fusion neutron at start-up conditions, which correspond to 1858, 1521, and 934 kg ^{233}U/year, respectively, by a full fusion power of 2807 MW. However, previous long-term plant operation calculations on hybrid blankets indicate that this high fissile fuel production rate would decrease rapidly, due to the burn-up of the new fissile fuel *in situ* [29-31]. Therefore, the reduction in fissile fuel breeding can be more than by a factor of 2 after a plant operation of 1 year, respectively [33].

TBR values in the blanket for all type of fuels which are greater than 1.05 means that tritium self-sufficiency is maintained for DT fusion driver in all cases. TBR value changes from ~1.18 to ~1.23 depending on the fuel types. Almost 95 % of tritium is produced from ^6Li while only 5 % of it is gained from ^7Li. The maximum reached TBR value is ~1.23 in the blanket fueled with ThF_4.

Peak-to-average fission power density ratio, Γ is a measure of spatial non-uniformly fission energy density that must be reduced to ~1.0 for obtaining a flat fission power density. Γ values are in the ranges of 1.46-1.50 depending on the type of fuels. SiC reflector reduces the neutron leakage out of the blanket drastically due to high reflectivity of carbon so that neutron leakage out of the blanket, L, is quite low in all cases. This value changes from 0.0199 to 0.0217 according to the fuel type.

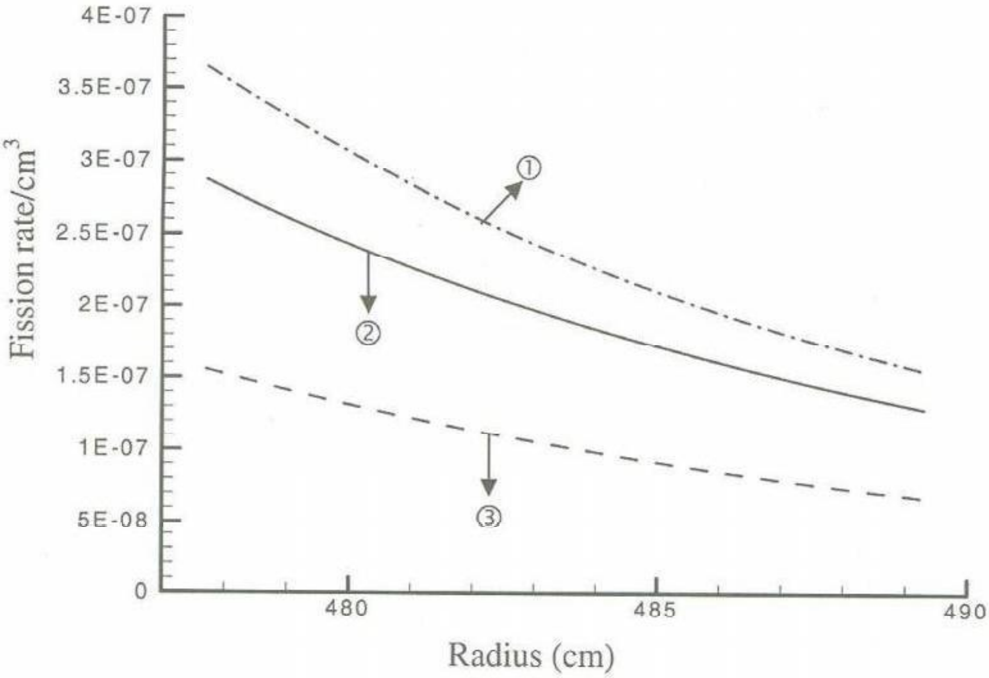

Figure 2. Fission rate per cm^3 per (D,T) fusion neutron in the FFB zone for different fuels ① ThN, ② ThC_2, and ③ ThF_4

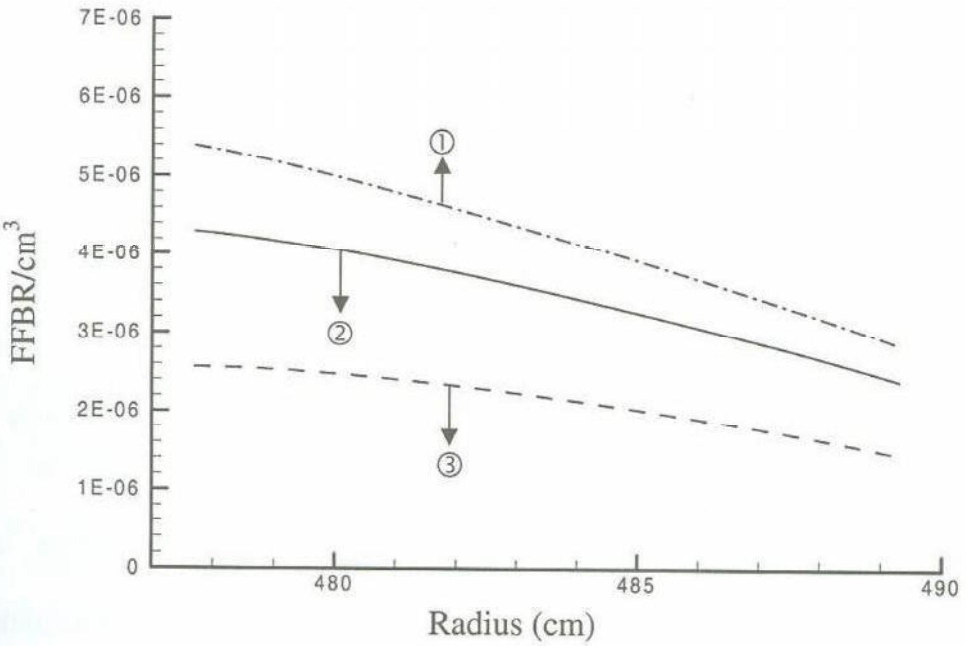

Figure 3. FFBR per cm^3 per (D,T) fusion neutron in the FFB zone for fuels, ① ThN, ② ThC_2, and ③ ThF_4

4. CONCLUSIONS AND RECOMMENDATIONS

In this study, the neutronic performance of the PROMETHEUS Fusion Reactor by using different type of fuels that are ThN, ThC_2, and ThF_4 has been investigated. The main conclusions for this study can be cited as follows:

- Substantial amount of ^{233}U fissile fuel production is possible by using FFB zone in PROMETHEUS IFE reactor.
- Higher energy multiplication values are reached with respect to pure PROMETHEUS IFE reactor.
- TBR values are greater than 1.05 for these fuels. Therefore tritium production in the blanket is self-sufficient for DT fusion driver.
- Γ values change in the ranges of 1.46 and 1.50 depending on the fuel type due to nonuniform fission density in the FFB zone.

In conclusion, it is recommended that by using different type of thorium fuels, the fissile fuel breeding in PROMETHEUS IFE reactor is possible. For further studies, use of spent fuel and power flattening can be considered.

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