Erc. Ünv. Fen Bil . Ens. Derg. 12, 1-2 (1996) 75-92

NEUTRONIC ANALYSIS OF A PURE (D,T) FUSION-DRIVEN HYBRID BLANKET FOR THE REGENERATION OF LWR SPENT FUEL

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ABSTRACT

The potential of a pure (D,T) fusion-driven hybrid blanket is investigated for the regeneration of light water reactor spent fuel. Total enrichment grade is 2.172% at beginning of regeneration. The hybrid blanket has excellent neutronic performance and is investigated to achieve different cumulative fission fuel enrichment (CFFE) grades. A regeneration period of up to 36 months is investigated by a plant factor (PF) of 75% under a first wall pure (D,T) fusion neutron current load of 10^{14}-14.1 MeV.n/cm^2.s. This corresponds to a first wall load of 2.25 MW/m^2. Regeneration periods of 20 and 32 months are considered, resulting in final enrichment grades of 3.0 and 3.5%, respectively.

The blanket energy multiplication M is quite high and increases only by ~16% in 36 months. The electricity production remains fairly constant during this period. Consequently, this power exploits the non-nuclear island very well. At the same time, the peak-to-average fission power density ratio Γ decreases by ~6%. On the other hand, it is seen that the blanket is a self-sufficient blanket for the tritium breeding density. Also, the neutron spectrum in the fuel zone is very stable and steady during a very long operation period.

KULLANILMIŞ LWR YAKITININ YENİLENMESİ İÇİN SAF (D,T) FÜZYON SÜRÜCÜLÜ BİR HİBRİD BLANKETİN NÖTRONİK ANALİZİ

ÖZET

Hafif su reaktörü kullanılmış yakıtı yenilemesi için saf (D,T) füzyon sürücülü bir hibrid rektörün potansiyeli incelendi. Yenilemenin başlangıcında toplam zenginleştirme yüzdesi %2.172 dir. Ele alınan hibrid blanket mükemmel neotronik perfonmans gösterdi ve değişik kümülatif fisyon yakıtı zenginleştirme oranları elde etmek için araştırma yapıldı. 10^{14}-14.1 MeV.n/cm^2.s lik bir cidar saf (D,T) fisyon nötron akı yüküne maruz blanket, 36 aya kadarlık yakıt yenileme peryodu %75 lik bir tesis faktörü ile araştırıldı. Bu, 2.25 MeV/m^2. lik bir cidar yüküne tekabül etmektedir. 20 ve 32 ay yenileme peryotları için sırayla 3.0 ve 3.5 zenginleştirme yüzdeleri elde edildi.

M blanket enerji çoğalım katsayısı oldukça yüksek olup ve 36 ayda sadece %16 oranında artmıştır. Dolayısıyla elektrik üretimi bu peryot süresince sabit kalmaktadır. Aynı zamanda Γ maksimum güç yoğunluğun ortalamasına oranı yaklaşık %6 mertebesinde azalmaktadır. Diğer taraftan trityum üretimi açısından blanketin kendi kendine yeterli olduğu görüldü. Ayrıca yakıt bölgesindeki nötron spekturumuda oldukça kararlı ve düzgündür.

1.INTRODUCTION

The potential of regenerating spent nuclear fuel in a hybrid blanket is an attractive area of application for early generation fusion reactors.

The lifetime of fuel bundles in a light water reactor (LWR) power plant is limited mainly because of nuclear fuel burn-up and consequent criticality and less because of material damage.

The conservative burn-up values in a HWR, LWR and FBR are in the order of 10.000, 30.000 and 100.000 MWd/t, respectively. It is reported that for burn-ups high enough to fully exploit the economic potential of the oxide fuel system, apparently peak discharge exposure must be in excess of 200.000 MWd/t [1]. Although burn-ups have not yet achieved such levels in conventional critical reactors, there is no reason to expect such performances can not be attained [2].

In recent work, the potential of a fusion driver in a hybrid blanket containing CANDU spent fuel and LWR spent fuel has been investigated, in detail [3-7], where the economic advantages of spent fuel regeneration in a hybrid reactor have been already pointed out. Regeneration to about 3% enrichment would already make re-utilization in a LWR possible.

2. MATERIAL AND METHOD

2.1.Blanket Geometry

In the present work, the investigations are extended to LWR spent fuel regeneration with pure (D,T) fusion-driven hybrid reactors.

In order to allow one a comparison of the evaluated neutronic parameters with previous work, the similar experimental and cylindrical blanket fast hybrid geometry has been chosen, as in refs. [3-7]. Figure 1 shows the hybrid blanket geometry adopted in this study.

The fissile zone is made of typical LWR fuel rods with Zircolay cladding and contains spent fuel having an isotopic composition following a burn-up 33.000 MWd/t, as published in ref.[8].

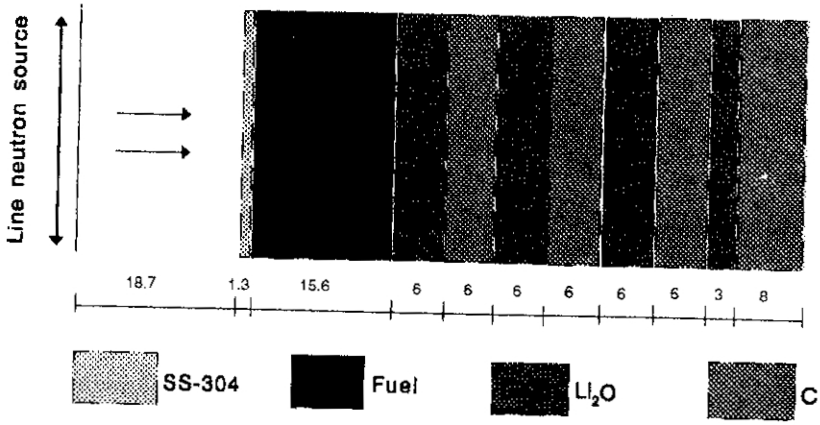

Figure 1. Cross-Sectional View of the Investigated Blanket

Figure 2 depicts the hexagonal arrangement of the LWR spent-fuel rods in an air cooled hybrid blanket. In this arrangement, the volume fractions of fuel, cladding, and air 45.5, 9 and 45.5%, respectively (V_A/V_F=1).

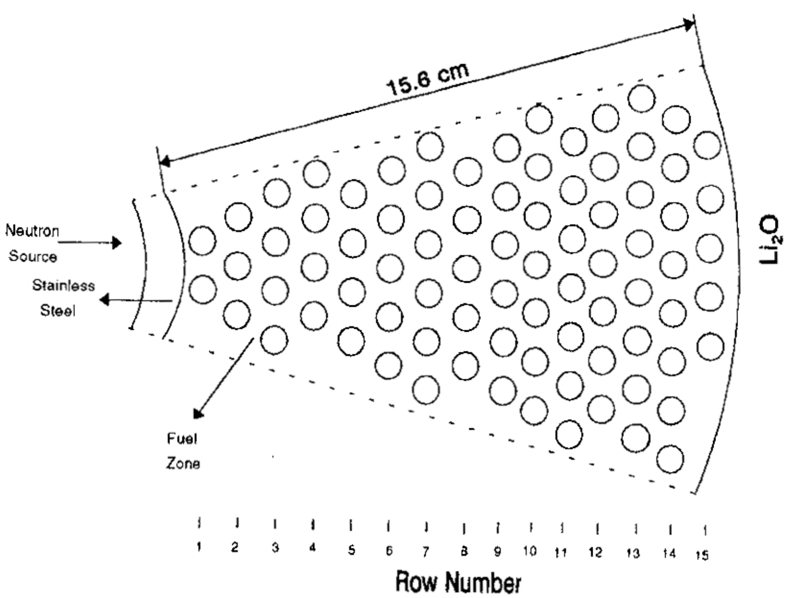

Figure 2. Hexagonal Arragement of the Blanket with Fifteen of Rows Air-Cooled Fuel Rods in the Radial Directon.

The fuel rods are arranged in the fissile zone in radial direction in fifteen rows. The fusion chamber represents an external fusion neutron source for the fissile zone. Therefore, the conversion of fissile nuclides is space dependent and changes from one row to the next row.

Figure 3 reflects the hexagonal cell dimensions of the fuel rods in the hybrid blanket configuration. Table I shows the material composition and the dimensions of the investigated hybrid blanket zones.

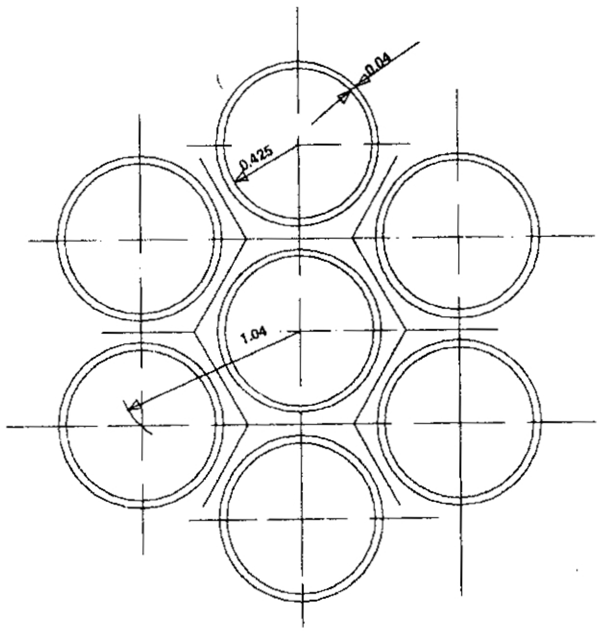

Figure 3: Cell Dimension in the Hexagonal Structure of the Fuel Rods in the Investigated Blanket.

Table 1. Material Composition of the Hybrid Blanket Zones in the Radial Direction

ZONE	MATERIAL	DIMENSION (cm)
Cavity	Air	0.0 to 18.7
First wall	Type 304 stainless steel	18.7 to 20.0
Fuel[a]	LWR spent fuel + Air	20.0 to 35.6
Tritium Breeding	Li_2O	35.6 to 41.6
Reflector	Graphite	41.6 to 47.6
Tritium Breeding	Li_2O	47.6 to 53.6
Reflector	Graphite	53.6 to 59.6
Tritium Breeding	Li_2O	59.6 to 65.6
Reflector	Graphite	65.6 to 71.6
Tritium Breeding	Li_2O	71.6 to 74.6
Reflector	Graphite	74.6 to 82.6

[a]The thickness of the fuel zone in most hybrid blanket concept studies varies typically from 10 to 20 cm.

The radial reflector contains Li_2O to produce tritium for pure (D,T) fusion reactors. The absence of Li_2O in the radial blanket would increase the fissile fuel breeding slightly. However, the sandwich structure of tritium breeding zone and the graphite reflector reduces the radial neutron leakage drastically [9-11]. Length of fuel zone of the blanket is longer then the fuel zone in refs. [3-7]. There are also differences between the sandwich structure of tritium zone and the graphite reflector of them. This option is used for a better neutron economy and regenerating more LWR spent fuel during same operation period.

Table II indicates the material composition and the homogenized atomic densities in the investigated hybrid blanket at start-up.

Table 2. Material Composition of the Hybrid Blanket Zones at Start-up

ZONE	MATERIAL	NUCLIDE	NUCLEI DENSITY ($10^{30}/m^3$)
First Wall	Type 304 Stainless Steel	Carbon	7.87287E-4[a]
		Silicon	6.73380E-4
		Chrominium	1.72770E-2
		Iron	5.92626E-2
		Nickel	8.05460E-3
Fuel	LWR spent fuel + Air	Oxygen	2.00580E-2
		Zirconium	3.85190E-3
		^{234}U	3.46000E-7
		^{235}U	3.20928E-5
		^{236}U	7.05039E-6
		^{238}U	9.30691E-3
		^{237}Np	1.57455E-6
		^{238}Pu	3.18922E-6
		^{239}Pu	1.15334E-4
		^{240}Pu	9.98888E-5
		^{241}Pu	6.29821E-5
		^{242}Pu	3.58035E-5
		^{241}Am	4.66349E-6
		^{243}Am	9.94877E-6
		^{244}Cm	4.41276E-6
Tritium Breeding	Li_2O	^{6}Li	4.63794E-3
		^{7}Li	5.70367E-2
		Oxygen	3.08374E-2
		Aluminium	3.01356E-3
	Graphite	Carbon	1.12840E-1

[a]Read as 7.87287×10^{-4}.

2.2 Numerical Calculations

The neutronic analysis has been performed in S_8-P_3 approximation with the help of the neutron transport code ANISN [12] using the neutron group data of CLAW-IV Library [13]. The neutron cross-sections in this library are averaged over 30 energy groups containing 12, 9 and 9 neutron groups in the MeV, keV and eV, respectively (see Table III). The numerical output of the ANISN calculations were further processed with the help of the auxiliary code ERDEMLI [14] to evaluate specific information for this work.

Pure (D,T) fusion driver:

$$D + T \longrightarrow {}^4He + n(14.1\ MeV) + Q(17.6\ MeV) \qquad (1)$$

Tritium breeding reactions:

$${}^6Li + n \longrightarrow {}^4He + T + Q(4.784) \qquad (2)$$

$${}^7Li + n \longrightarrow {}^4He + T + n + Q(-2.467\ MeV) \qquad (3)$$

A series of calculations has been performed to evaluate the neutronic performance and the regenerating behavior of the hybrid blanket with LWR spent fuel. In pure (D,T) fusion driver, it is assumed that the fuel zone is irradiated with a first wall neutron flux of 10^{14}-14.1 MeV.n/cm^2.s. This corresponds to a first wall neutron load of 2.25 MW/m^2. The plant factor (PF) is taken as 75%. For neutronic calculations, the fuel zone is divided into 15 equi-distant subzones, corresponding to the 15 fuel rod rows in the fissile zone. This allows one of to follow the space and time dependent nuclide density variations in a precise manner by considering the breeding and depletion reactions in the fissile and fissionable isotopes, in each fuel row individually.

Figure 4 depicts main nuclear conversion reactions which have been considered in course of regeneration process. The potential variations in fuel density ΔN are calculated separately for each isotope as follows.

For the breeding reactions:

$$+\Delta N_2 = PF.\Delta t.N_1 \int_E \sigma_{b1}(E)\phi(E)dE + \Delta t\, \lambda_{b1}\, N_1 \qquad (4)$$

Indices 1 and 2 denote mother and daughter nuclides, respectively.

For the depletion reactions:

$$+\Delta N = PF.\Delta t.N \int_E \sigma_{dep}(E)\phi(E)dE + .\Delta t.\lambda.N \qquad (5)$$

The temporal change of the fuel composition during hybrid reactor plant operation are evaluated for discrete time intervals Δt=6 months.

Table 3. Energy group structure of data used in the numerical calculation (E_{min}= 1.39E-4 eV)

	E^a [eV]	$E_m{}^b$	ΔU^c
1	1.70000E+07	1.59791E+07	0.125
2	1.50000E+07	1.42368E+07	0.105
3	1.35000E+07	1.27353E+07	0.118
4	1.20000E+07	1.09696E+07	0.182
5	1.00000E+07	8.84905E+06	0.250
6	7.79000E+06	6.89428E+06	0.249
7	6.07000E+06	4.77574E+06	0.500
8	3.68000E+06	3.25552E+06	0.250
9	2.86500E+06	2.53534E+06	0.250
10	2.23200E+06	1.97471E+06	0.250
11	1.73800E+06	1.53747E+06	0.250
12	1.35300E+06	1.06613E+06	0.497
13	8.23000E+05	6.48141E+05	0.498
14	5.00000E+05	3.93311E+05	0.501
15	3.03000E+05	2.38574E+05	0.499
16	1.84000E+05	1.16246E+05	1.001
17	6.76000E+04	4.26820E+04	1.003
18	2.48000E+04	1.56741E+04	1.000
19	9.12000E+03	5.76130E+03	1.002
20	3.35000E+03	2.11947E+03	0.998
21	1.23500E+03	7.80431E+02	1.001
22	4.54000E+02	2.86970E+02	1.000
23	1.67000E+02	1.05538E+02	1.001
24	6.14000E+01	3.88210E+01	0.999
25	2.26000E+01	1.42902E+01	0.999
26	8.32000E+00	5.25870E+00	1.000
27	3.06000E+00	1.93737E+00	0.996
28	1.13000E+00	7.13071E-01	1.004
29	4.14000E-01	2.61481E-01	1.002
30	1.52000E-01	2.17032E-02	6.997

[a]Higher energy boundary, $E_m=(E_i-E_{i+1})/\ln(E_i/E_{i+1})$, $\Delta U=\ln(E_i/E_{i+1})$, i=1,2,..30

Table IV shows the temporal variations of the most important integral neutronic data in the hybrid reactor for a plant operation period of up to 36 months. One can observe a general improvement of the overall blanket neutronic performance. The blanket energy multiplication M is quite high, corresponding to a substantial electricity production of this type of hybrid power plants. The relative increase ΔM is rather modest for a hybrid reactor. Also, the variation of tritium breeding density in this self-sufficient blanket is seen. At the same time, it is shown that the radial neutron leakage is quite low during operation period (~6.5% per fusion cycle).

The regeneration of the spent fuel is measured on the so called cumulative fission fuel enrichment (CFFE) grade which is defined as the accumulated isotopic percentages of the main fissile nuclides ^{235}U, ^{239}Pu, ^{241}Pu, ^{242m}Am and ^{245}Cm in the fuel.

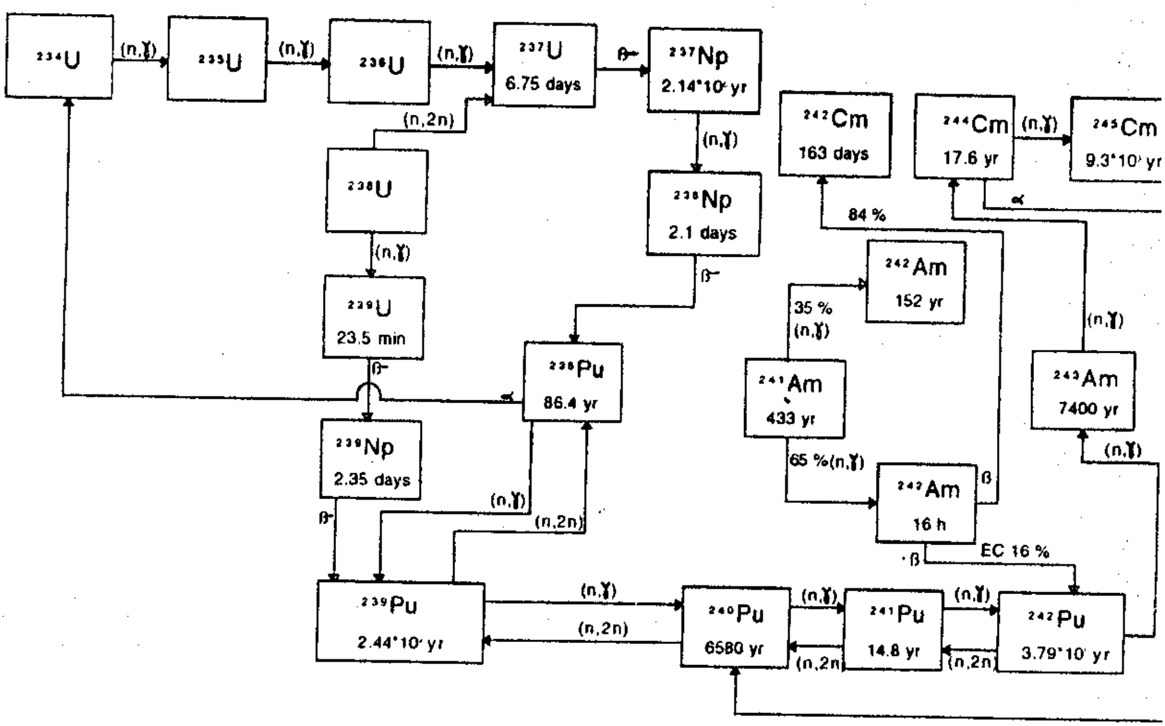

Figure 4. Main Conversion Reactions of Fissile Nuclides.

Table 4. Variation Of Integral Neutronic Data During Plant Operation per Pure (D,T) Fusion Cycle in the Fast Fissioning Blanket (Incident Neutron with 14.1 MeV)

	Plant Operation Periods (Months)						
	0	6	12	18	24	30	36
T_6	1.24640	1.26080	1.27522	1.28946	1.30405	1.31846	1.33284
T_7	0.08127	0.08151	0.08174	0.08200	0.08223	0.08248	0.08273
T	1.32767	1.34231	1.35696	1.37163	1.38629	1.40094	1.41558
$\nu\Sigma_f$	1.00388	1.03683	1.06975	1.10263	1.13544	1.16817	1.20079
Σ_f	0.27990	0.29125	0.30260	0.31393	0.32525	0.33654	0.34779
M	5.51067	5.65855	5.80636	5.95403	6.10146	6.24856	6.39523
ΔM	0.00000	0.02683	0.05366	0.08045	0.10721	0.13390	0.16052
Γ	1.96812	1.94641	1.92552	1.90538	1.88596	1.86720	1.84906
L	0.06301	0.06362	0.06424	0.06486	0.06547	0.06609	0.06671

Legend:

T6 : Integral tritium production through ^{6}Li(n, a)T reaction

T7 : Integral tritium production through ^{7}Li(n, a, n')T reaction

T : $T_6 + T_7$:Total Tritium production

$\nu\Sigma_f$: Integral fission neutron

Σf : Integral fission rate

M : (Blanket Energy Release in MeV)/14.1 + 1

Γ : Peak-to-average fission power density ratio in the fuel zone

L : Radial neutron leakage fraction per fusion cycle

Table V and Figure 5 show the variation CFFE as a function of plant operation time for discrete intervals. The start up value of the spent fuel is CFFE=2.172% for all rows.

The pure (D,T) fusion driver produces 1 neutron per fusion cycle, equation (1), and is therefore poorer in neutron than the catalyzed (D,D) fusion cycle, per unit neutron energy. One can assume that CFFE=3.5% would be sufficient for a re-utilization of the spent fuel in a LWR. Then, pure (D,T) fusion-fission hybrid reactor would regenerate the spent fuel within about 32 months for the next utilization in a conventional LWR.

In figure 6 and 7, one can observe the neutron spectrum in the fuel zone at start-up and after 36 months, respectively. It is apparent that the neutron spectra in the fuel zone do not practically change, except for very low neutron energies, where the neutron flux is very depressed. This explains the very stable and steady behavior of all neutronically related plant parameters during a very long operation period.

Table 5. Cumulative Fission Fuel Enrichment Grades Throughout the Fissile Core of the Hybrid Reactor as a Function of Plant Operation Period

*Row #	Plant Operation Period (Months)						
	0	6	12	18	24	30	36
1	2.172[a]	2.44157	2.70397	2.96003	3.20989	3.45365	3.69144
2	2.172	2.43278	2.68697	2.93538	3.17810	3.41523	3.64686
3	2.172	2.42703	2.67590	2.91939	3.15757	3.39053	3.61834
4	2.172	2.42248	2.66716	2.90680	3.14147	3.37122	3.59613
5	2.172	2.41868	2.65986	2.89633	3.12810	3.35525	3.57783
6	2.172	2.41548	2.65377	2.88760	3.11702	3.34207	3.56279
7	2.172	2.41292	2.64890	2.88068	3.10829	3.33177	3.55113
8	2.172	2.41110	2.64550	2.87592	3.10239	3.32492	3.54355
9	2.172	2.41026	2.64402	2.87399	3.10020	3.32265	3.54136
10	2.172	2.41079	2.64524	2.87607	3.10327	3.32686	3.54685
11	2.172	2.41343	2.65059	2.88421	3.11431	3.34089	3.56397
12	2.172	2.41938	2.66250	2.90211	3.13823	3.37085	3.59997
13	2.172	2.43135	2.68633	2.93770	3.18544	3.42958	3.67010
14	2.172	2.45567	2.73457	3.00946	3.28034	3.54721	3.81006
15	2.172	2.51175	2.84552	3.17409	3.49747	3.81565	4.12867

*Spent fuel row number in radial core direction, [a]CFFE(%)

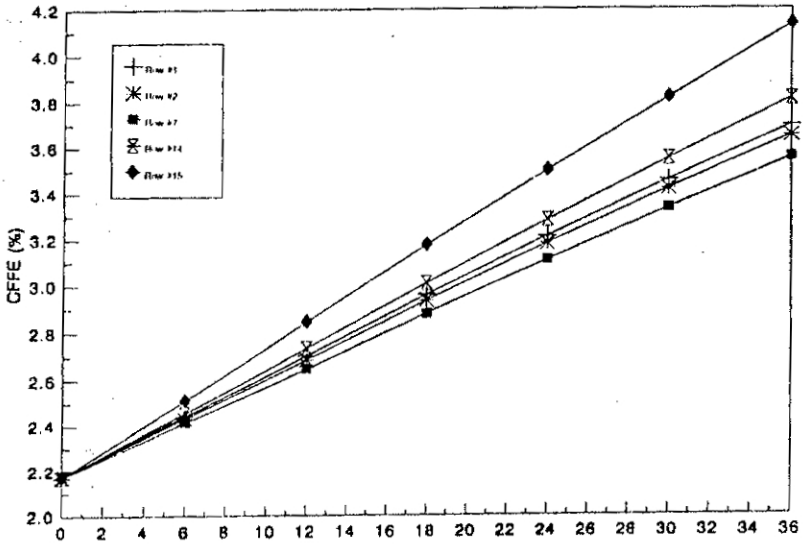

Figure 5. Accumulation of the Fissile Isotopes in Different Fuel Zones of the Pure (D,T) Fusion-Driven Hybrid Blanket During Plant Operation Period.

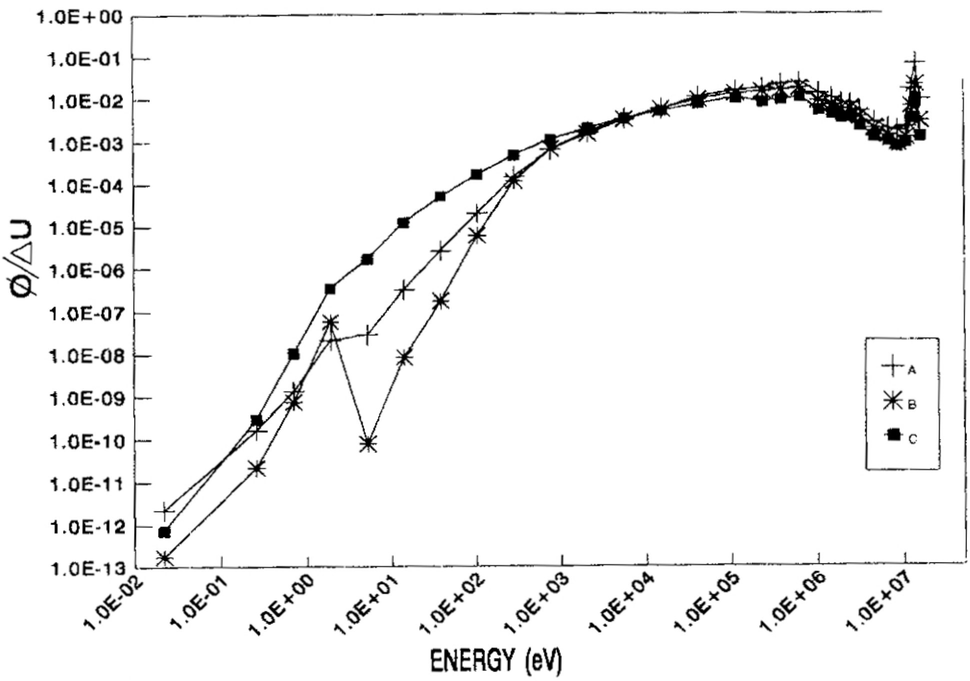

Figure 6. Neutron Spectrum in the Fuel Zone at Start-up

A, Adjacent to the First Wall;

B, In the Middle of the Fuel Zones;

C, Adjacent to the First Li2O

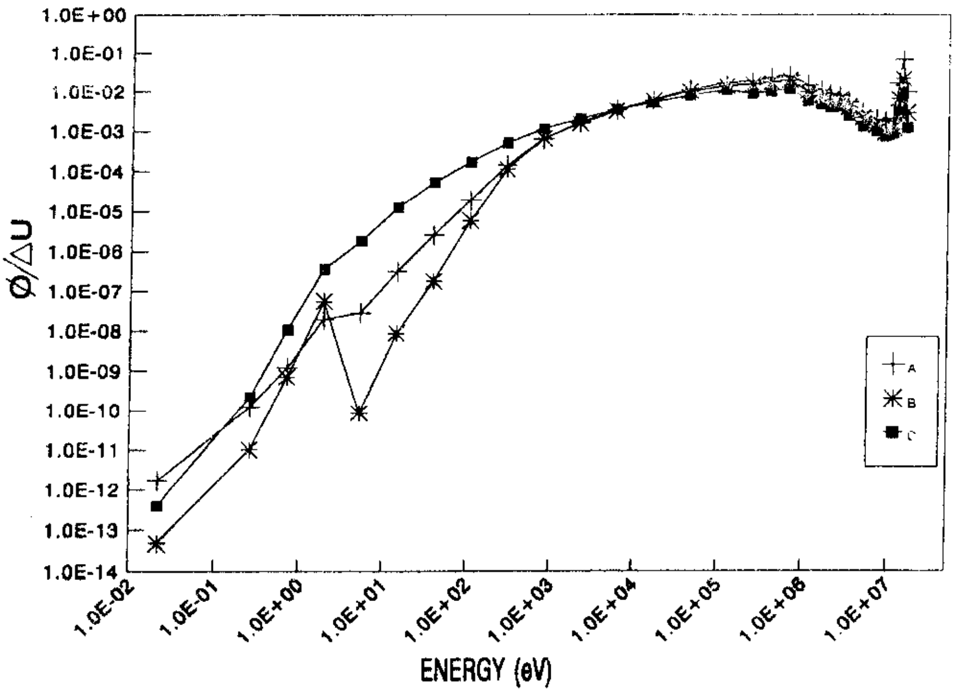

Figure 7. Neutron Spectrum in the Fuel Zone After 36 Months

A, Adjacent to the First Wall;

B, In the Middle of the Fuel Zones;

C, Adjacent to the First Li2O

3.CONCLUSIONS

The present study has analyzed the possibility of regeneration of LWR spent fuel in a pure (D,T) fusion-driven hybrid reactor. The main conclusions can be cited as follows:

a) A hybrid reactor driven by pure (D,T) fusion neutron has a high neutronic performance. The reactor can produce electricity in situ and regenerate the LWR spent fuel for a re-utilization.

b) The regeneration period of spent fuel for the next utilization in a conventional LWR is short with pure (D,T) fusion-driven hybrid reactor (36 months).

c) Fuel rods are regenerated more than the study in refs. [3-7] during the same regeneration period (1.5 times fuel rods).

d) The blanket is self-sufficient for producing tritium required to obtain pure (D,T) fusion reaction during operation period (1.3 ~ 1.4 tritium per fusion cycle.)

e) It is seen that the sandwich structure of tritium breeding zone and the graphite reflector used in this study reduces the radial neutron leakage more than the reflector in ref. [7] during the same operation period (~6.5% per fusion cycle).

f) The presence of a substantial amount of different fissile isotopes in the initial charge of the hybrid blanket containing LWR spent fuels leads to a relatively slow increase of the blanket energy multiplication, because the burn-up and breeding rates of these isotopes remain approximately at the same level with the exception of ^{239}Pu isotope. The latter causes the slow temporal increase in M which is essential for a better exploitation of the non-nuclear island over a long plant operation period. A hybrid blanket having an initial charge with natural uranium would have much higher power swings [15].

In summary, a pure (D,T) fusion driven hybrid blanket is suitable for LWR spent fuel regeneration under all investigated aspects of neutronic behavior .

NOMENCLATURE

E = neutron energy

L = radial neutron leakage fraction per fusion cycle

M = blanket energy multiplication factor

N = isotope atomic density

PF = plant factor

t = plant operation time

V = volume

Greek

G = peak-to-average ratio of fission power density

n = neutron production per fission

l = radioactive decay rate

s = microscopic cross section

S = macroscopic cross section

f = neutron flux

Subscript

A = air

b = breeding

dep = depletion

F = fuel

f = fission

REFERENCES

1) R. D Leggert, R. P. Omberg, "Mixed Oxide Fuel Development, "Proceedings of the International Conference on Fast Breeder Systems, Pasco, Washington (1987).

2) A. E. Walter, L. W. Deitrich, "Status of Research on Key LWR Safety Issues," Nuclear Safety, Vol.29, No.2, pp.125-148(1988).

3) S. Şahin, H. Yapıcı, "Rejuvenation of CANDU Spent Fuel in a Hybrid Blanket," Transactions of The American Nuclear Society, Vol.59, pp.105-106, Annual Meeting, Atlanta (June 4-8, 1989).

4) S. Şahin, H. Yapıcı, "Investigation of the Neutronic Potential of Moderated and Fast (D,T) Hybrid Blankets for Rejuvenation CANDU Spent Fuel," Fusion Technology, Vol.16, pp.331-345 (Nov.1989).

5) S. Şahin, H. Yapıcı, E. Baltacıoğlu, "Heavy Water Reactor Spent-Fuel Regeneration with Fusion Neutrons," Transactions of The American nuclear society, vol.60, pp.176-178, Nov.26-30, 1989.

6) S. Şahin, E. Baltacıoğlu, H. Yapıcı, "Potential of a Catalyzed Fusion-Driven Hybrid Reactor for the Regeneration of CANDU Spent Fuel," Fusion Technology, vol.20, no.1, pp.26-39, August 1991.

7) S. Şahin, H. Yapıcı, E. Baltacıoğlu, "Rejuvenation of LWR Spent Fuel in a Catalyzed Fusion Hybrid Blanket," Kerntechnik, vol.59, no.6, pp.270-277, Nov. 1994.

8) D. H. Berwald and J. J. Duderstatdt, "Preliminary Design and Neutronic Analysis of a Laser Fusion Driven Actinide Waste Burning Hybrid Reactor," Nucl.Technol.,vol.42, pp.34-50(1979).

9) A. Kumar, S. Şahin, "(D,D) Driven Experimental Hybrid Blankets and their Neutronic Analysis," Fusion Technology, vol.6, no.2, pp.225-239 (1984).

10) S. Şahin, T. A. Ai-Kusayer, M. Abdul raoof, "Neutronic Performance of (D,D) Driven Experimental Thorium Hybrid Blankets," Transaction of the American Nuclear Society. vol.49, pp.102-104 (June 1985)

11)S. Şahin, T. A. Ai-Kusayer, M. Abdul raoof,, "Preliminary Design Studies of a Cylindrical Experimental Hybrid Blanket with Deterium-Tritium Driver," Fusion Technology, vol.10, p.84 (1986).

12) W. W Engle, Jr., "ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K1693, OAK Ridge Nationaly Laboratory (1970).

13) T. A. Ai-Kusayer, S. Şahin, A. Drira, "CLAW-IV Coupled 30 Neutrons 12 Gamma-Ray Group Section with Retrieval Programs for Radiation Transport Calculations," Radiation Shielding Information Center OAK Ridge National Laboraty, RSIC-Newletter, pp.4 (May 1988).

14) S. Şahin, H. Yapıcı, S. Ünalan, "ERDEMLY, A Computer Program to Process ANISN Output Data," Gazi University, Ankara, (1991).

15) F.R. Mynatt, "Analysis of Acceleration Breeder Concepts with LMFBR, GCFF and Molten Salt Type Blankets," Proc, Information Mtg. Accelerator Breeding, Upton, New York, January 18-19, 1977, CONF.770107