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# POTENTIAL OF A CATALYZED (D,D) FUSION-DRIVEN HYBRID REACTOR FOR THE REGENERATION OF LWR SPENT FUEL

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## ABSTRACT

The potential of a catalyzed (D, D) 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 of 75% under a first wall catalyzed (D, D) fusion neutron current load of $10^{14}$. Neutron (14.1 MeV)/cm$^2$. s and $10^{14}$ neuton (2.45 MeV)/cm$^2$. s. This corresponds to a first wall load of 2.64 MW/m$^2$. Regeneration periods of 12, 20, 28 and 36 months are considered, resulting in final enrichment grades of 3.0, 3.5, 4.0, and 4.5 %, respectively. The blanket energy multiplication M is quite high and increases only by ~ 30 % 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 ~10 %.

**Key Words :** Regeneration, Spent fuel, LWR

## LWR'DE KULLANILMIŞ YAKITIN YENİLENMESİ İÇİN KATALİZE (D, D) FÜZYON SÜRÜCÜLÜ BİR HİBRİD REAKTÖRÜN POTANSİYELİ

## ÖZET

Hafif su reaktörü kullanılmış yakıtı yenilemesi için katalize (D, D) füzyon sürücülü bir hibrid reaktörünün potansiyeli incelendi. Yenilemenin başlangıcında toplam zenginleştirme yüzdesi % 2.172 olarak ele alındı. Ele alınan hibrid blanket mükemmel nötronik perfonmans gösterdi ve değişik kümülatif fisyon yakıtı zenginleştirme oranları elde etmek için araştırma yapıldı. $10^{14}$ nötron (14.1 MeV)/cm$^2$ s ve $10^{14}$ nötron (2.45 MeV)/cm$^2$ s.'lik bir cidar katalize (D, D) füzyon nötron akı yüküne maruz blanket, 36 aya kadarlık yakıt yenileme peryodu % 75'lik bir tesis faktörü ile araştırıldı. Bu 2.64 MW/m$^2$'lik bir cidar yüküne tekabül etmektedir. 12, 20, 28 ve 36 ay yenileme periyotları için sırayla 3.0, 3.5, 4.0 ve % 4.5 zenginleştirme yüzdeleri elde edildi. M blanket enerji çoğalım katsayısı oldukça yüksek olup ve 36 ayda sadece % 30 oranında artmıştır. Dolayısıyla elektrik üretimi bu periyot süresince oldukça sabit kalır. Aynı zamanda Γ maksimum güç yoğunluğun ortalamasına oranı yaklaşık % 10 mertebesinde azalmaktadır.

**Anahtar Kelimeler :** Yenileme, Kullanılmış yakıt, Hafif su reaktörü

## 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 (Leggert and Omberg, 1987). 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 (Waltar and Deitrich, 1988).

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 (Şahin and Yapıcı 1989a; Şahin and Yapıcı, 1989b; Şahin, Yapıcı and Baltacıoğlu, 1989; Şahin, Baltacıoğlu and Yapıcı, 1991; Şahin, Yapıcı and Baltacıoğlu, 1994), where the economic advantages of spent fuel regeneration in a hybrid reactor have been already pointed out.

## 2. MATERIAL AND METHOD

### 2. 1. Blanket Geometry

In the present work, the investigations are extended to LWR spent fuel regeneration with catalyzed (D, D) 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. (Şahin and Yapıcı 1989a; Şahin and Yapıcı, 1989b; Şahin, Yapıcı and Baltacioğlu, 1989; Şahin, Baltacıoğlu and Yapıcı, 1991; Şahin, Yapıcı and Baltacıoğlu, 1994).

Figure 1 shows the hybrid blanket geometry adopted in this study.

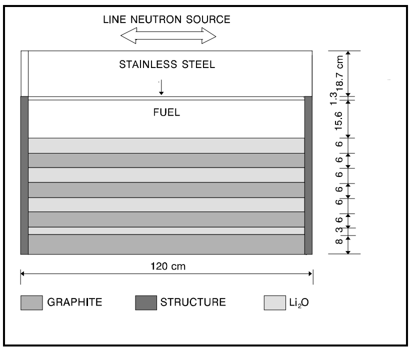


Figure 1. Cross-sectional view of the investigated blanket

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. (Berwald and Duderstadt, 1979).

Figure 2 shows 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 ($V_A/V_F = 1$).

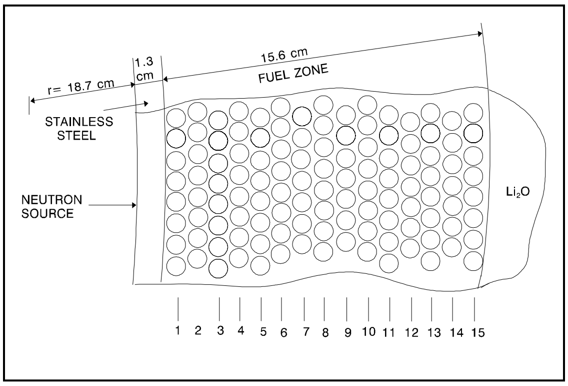


Figure 2. Hexagonal arrangement of the blanket with fifteen of rows air-cooled fuel rods in the radial direction

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.

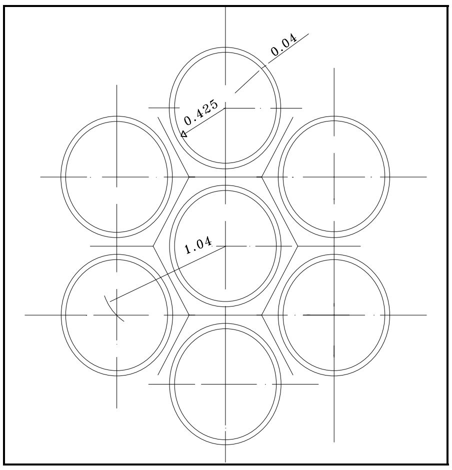


Figure 3. Cell dimensions in the hexagonal structure of the fuel rods in the investigated blanket

Table 1 shows the material composition and the dimensions of the investigated hybrid blanket zones.

Table l. 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 thicknessof 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 satellite (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 (Kumar and Şahin, 1984; Şahin, Al-Kusayer and Abdul Raoof, 1985; Şahin, Al-Kusayer and Abdul Raoof, 1986), so that this option has been used for a better neutron economy.

Table 2 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 |
| Reflector | Graphite | Carbon | 1.12840E-1 |

[a]Read as 7.87287 X $10^{-4}$.

## 2. 2. Numerical Results

The neutronic analysis has been performed in $S_8$-$P_3$ approximation with the help of the neutron transport code ANISN (Engle, 1970) using the neutron group data of CLAW-IV Library (AI-Kusayer, Şahin and Drira, 1988).

The numerical output of the ANISN calculations were further processed with the help of the auxiliary code ERDEMLI (Şahin, Yapıcı and Ünalan, 1991) to evaluate specific information for this work.

Catalyzed (D, D) fusion driver:

$$D + D \longrightarrow T + p$$
$$D + D \longrightarrow {}^{3}He + n$$
$$T + D \longrightarrow {}^{4}He + n$$
$$^{3}He + D \longrightarrow {}^{4}He + p$$

$$6D \longrightarrow 2\,{}^{4}He + 2p + 2n + Q(43.2\ \text{MeV}) \quad (1)$$

[illegible]atalyzed (D,D) fusion driver, it is assumed that the fuel zone is irradiated with a first wall neutron flux of $10^{14}$-2.45 MeV.n/cm$^2$.s and $10^{14}$-14.1 MeV.n/cm$^2$.s. This corresponds to a first wall neutron load of 2.64 MW/m$^2$. The plant factor 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. The temporal change of the fuel composition during hybrid reactor plant operation are evaluated for discrete time intervals $\Delta t$ = 6 months.

Figure 4 depicts main nuclear conversion reactions which have been considered in course of regeneration process.

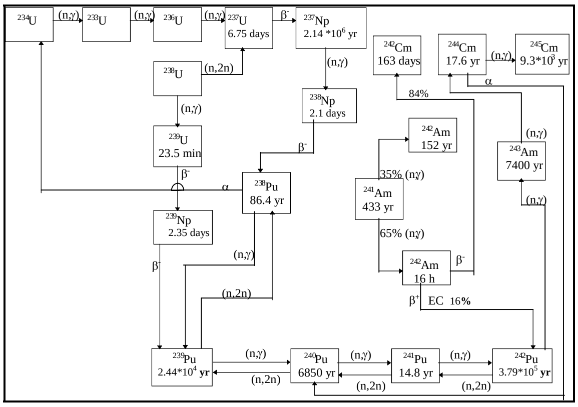


Figure 4. Main conversion reactions of the fissle nuclides

Table 3. Variation of Integral Neutronic Data During Plant Operation Per Catalyzed (D, D) Fusion Cycle in the Fast Fissioning Blanket (One Incident Neutron with 2.45 MeV and One with 14.1 MeV)

| | Plant Operation Periods (Months) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0 | 6 | 12 | 18 | 24 | 30 | 36 |
| $T_6$ | 2.09335 | 2.13413 | 2.17503 | 2.21598 | 2.25691 | 2.29777 | 2.33846 |
| $T_7$ | 0.08319 | 0.08372 | 0.08427 | 0.08483 | 0.08539 | 0.08597 | 0.08655 |
| T | 2.17654 | 2.21786 | 2.25930 | 2.30081 | 2.34231 | 2.38374 | 2.42501 |
| $\nu\Sigma_f$ | 1.49120 | 1.58220 | 1.67320 | 1.76406 | 1.85459 | 1.94465 | 2.03404 |
| $\Sigma_f$ | 0.43705 | 0.46861 | 0.50018 | 0.53170 | 0.56310 | 0.59434 | 0.62534 |
| M | 7.02547 | 7.37561 | 7.72584 | 8.07559 | 8.42420 | 8.77104 | 9.11539 |
| $\Delta$M | 0.00000 | 0.04984 | 0.09969 | 0.14947 | 0.19909 | 0.24846 | 0.29748 |
| $\Gamma$ | 1.93440 | 1.89706 | 1.86225 | 1.82962 | 1.79896 | 1.77004 | 1.74269 |
| L | 0.09550 | 0.09718 | 0.09888 | 0.10058 | 0.10229 | 0.10400 | 0.10571 |

Legend, $T_6$ = Integral tritium production through $^6$Li(n, $\alpha$)T reaction, $T_7$ = Integral tritium production through $^7$Li(n, $\alpha$, n')T reaction, T = $T_6$ + $T_7$ :Total Tritium production, $\nu\Sigma_f$ = Integral fission neutron, $\Sigma_f$ = Integral fission rate, M = (Blanket Energy Release in MeV)/16.55+1, $\Gamma$ = Peak-to-average fission power density ratio in the fuel zone, L = Radial neutron leakage fraction per fusion cycle

Table 3. 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.

Table 4. Cumulative Fission Fuel Enrichment Grades Throughout the Fissile Core of the Hybrid Reactor as a Function of Plant Operation Period

| Row Number (*) | Plant Operation Period (Months) | | | | | | |
|---|---|---|---|---|---|---|---|
| | 0 | 6 | 12 | 18 | 24 | 30 | 36 |
| 1 | 2.172[a] | 2.62121 | 3.05220 | 3.46622 | 3.86375 | 4.24531 | 4.61137 |
| 2 | 2.172 | 2.60571 | 3.02288 | 3.42461 | 3.81129 | 4.18331 | 4.54106 |
| 3 | 2.172 | 2.59561 | 3.00388 | 3.39782 | 3.77776 | 4.14399 | 4.49683 |
| 4 | 2.172 | 2.58771 | 2.98908 | 3.37706 | 3.75190 | 4.11385 | 4.46315 |
| 5 | 2.172 | 2.58119 | 2.97693 | 3.36010 | 3.73089 | 4.08951 | 4.43616 |
| 6 | 2.172 | 2.57581 | 2.96694 | 3.34624 | 3.71387 | 4.06996 | 4.41468 |
| 7 | 2.172 | 2.57156 | 2.95914 | 3.33556 | 3.70092 | 4.05533 | 4.39890 |
| 8 | 2.172 | 2.56863 | 2.95391 | 3.32860 | 3.69278 | 4.04652 | 4.38991 |
| 9 | 2.172 | 2.56740 | 2.95197 | 3.32646 | 3.69089 | 4.04531 | 4.38980 |
| 10 | 2.172 | 2.56852 | 2.95462 | 3.33102 | 3.69773 | 4.05477 | 4.40217 |
| 11 | 2.172 | 2.57436 | 2.96532 | 3.34683 | 3.71887 | 4.08145 | 4.43457 |
| 12 | 2.172 | 2.58464 | 2.98590 | 3.37773 | 3.76010 | 4.13301 | 4.49643 |
| 13 | 2.172 | 2.60508 | 3.02638 | 3.43789 | 3.83959 | 4.23144 | 4.61345 |
| 14 | 2.172 | 2.64627 | 3.10752 | 3.55783 | 3.99718 | 4.42557 | 4.84297 |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 15 | 2.172 | 2.74074 | 3.29291 | 3.83081 | 4.35445 | 4.86388 | 5.35913 |

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

The relative increase $\Delta M$ is rather modest for a hybrid reactor. Also, the variation of tritium breeding density in this self-sufficient blanket is seen.

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.

Table 4 shows 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 catalyzed fusion driver produces 2 neutrons per fusion cycle, equation (1), and is therefore richer in neutrons than the pure (D, T) 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, catalyzed (D, D) fusion-fission (hybrid) reactor would regenerate the spent fuel within about 18 months for the next utilization in a conventional LWR.

In figure 5 and figure 6, 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.

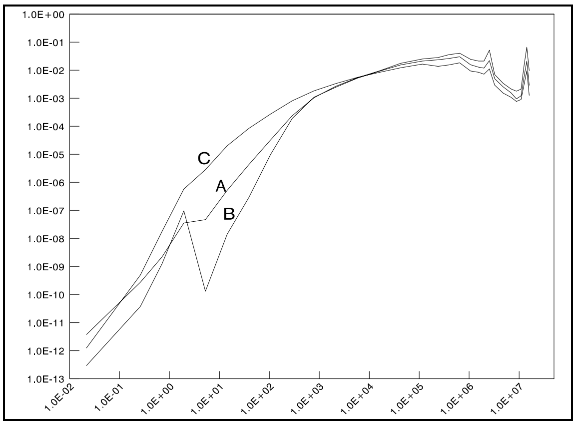


Figure 5. Neutron spectrum in the fuel zone at start-up: A, adjacent to first wall; B, in the middle of zone; C, adjacent to first $Li_2O$ zone

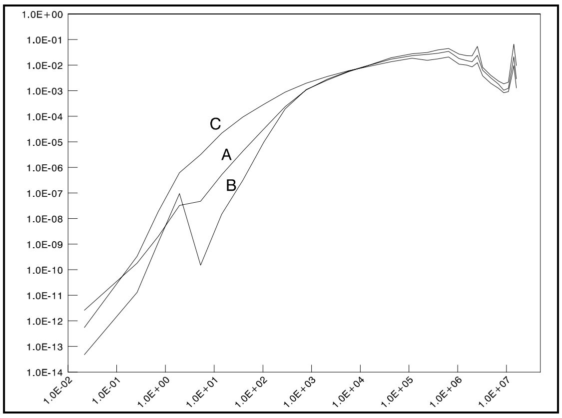


Figure 6. Neutron spectrum in the fuel zone after 36 months : A, adjacent to first wall; B, in the middle of zone; C, adjacent to first $Li_2O$ zone

## 3. CONCLUSIONS

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

a) A hybrid reactor driven by catalyzed (D, D) fusion neutrons 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 is short with catalyzed (D, D) fusion-driven hybrid reactor (18 months).
c) Fuel rods are regenerated more than the study in ref.7 during the same regeneration period (1.5 times fuel rods).
d) 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 isessential 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 (Mynatt, 1977).

### 3. 1. Nomenclature

E = Neutron energy
L = Radial neutron leakage fraction per fusion cycle
M = Blanket energy multiplication factor
t = Plant operation time
V = Volume

**3. 2. Greek**

Γ = Peak-to-average ratio of fission power density
ν = Neutron production per fission
Σ = Macroscopic cross section
ϕ = Neutron flux

3. 3. Subscript

A = Air
F = Fuel
f = Fission

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