Erc.Üni.Fen.Bil.Derg., 5, 1-2 (1989), 855-870

ENRICHMENT OF CANDU SPENT FUEL IN A FAST HYBRID REACTOR

Sümer ŞAHİN, Gazi Üniversitesi, Teknik Eğitim Fak. Beşevler, ANKARA

Hüseyin YAPICI, Ertuğrul BALTACIOĞLU
Erciyes Üniversitesi, Mühendislik Fakültesi KAYSERİ

ABSTRACT

The suitability of a fast hybrid blanket is investigated for the regeneration of Canada deuterium uranium (CANDU) spent fuel. The fast fissioning blanket has an appropriate neutronic economy and is investigated to achieve different enrichment grades of fissile isotopes (EGFI) for three different applications:

1. Recyling in a conventional commercial CANDU reactor, (EGFI=0.71) to 0.9%), regeneration period (RP)=6 to 9 months.

2. Recyling in a advanced CANDU reactor concept with high burnup rate (EGFI=1%), RP = 10-16 months.
3. Recyling in an advanced breeder with thorium fuel (EGFI 1.5%), RP > 18 months.

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 Canada deuterium uranium (CANDU) power plant is limited mainly because of nuclear fuel burnup and consequent criticality and less because of material damage. The conservative burnup values in a heavy water reactor (HWR) is on the order of 10000 MWd/t, although most fuel rods would support higher burnup rates, namely 200.000 MWd/t and higher [1,2] (*). Hence, a fuel regeneration would have the following economic potentials for an energy-producing system,

1. Higher nuclear fuel exploitation.

2. Elimination of the need for new fuel rod fabrication for an utilization a hybrid reactor and during reutilization in a conventional reactor.

3. Drastic reduction for nuclear fuel reprocessing per unit of total energy production, notably in conventional reactor, during regeneration in a hybrid reactor, and in the recycling phase in a conventioanal reactor until final utilization.

4. Lower nuclear fuel waste output per unit of total energy production.

In the past, this option was investigated to some degree for LWR spent fuel [3,4]. Little attention has been paid to the regeneration of spent CANDU fuel, although this type of reactor has the lowest burnup rate per fuel mass and, consequently, the highest ratio of spent-fuel mass/burnup. At present, CANDU spent fuel is stored without any clearly defined plans for future use.

In this study, the behaviour of CANDU spent-fuel regeneration in a hybrid blanket has been investigated for three different applications:

(*) A CANDU fuel mod may not attain a burn-up value of 200.000 MWd/t. But one can expect that it is able to with stand a much higher burn-up 10.000 MWd/t.

1. 0.7 to 0.9% enrichment for recyling in conventional CANDU reactors, taking in the account that in a CANDU reactor spectrum, the reactivity effect of ^{239}Pu is smaller than that of ^{235}U, for the following reasons:

   a) The fission to parasitic absorption ratio of ^{235}U is about twice higher than that ^{239}Pu for thermal energies.

   b) The parasitic absorption in ^{239}Pu produces the ^{240}Pu isotope. The latter has a strong absorption resonance in thermal energy region, and with that detrimental effects on reactor reactiviyty.

2. 1.0% enrichment for use in an advanced type of commercial HWR; recent studies indicate that the burnup rate in a CANDU reactor canbe increased substantially if the fuel charge is slightly enriched [5].

3. Minimum enrichment of 1.5% for use in an advanced CANDU breeder concept with thorium fueling. This option requires an average enrichment level of 1.5% [6]. In order to achieve a commercially reasonable breeding capability of ^{233}U from ^{232}Th.

II. DISTRIPTION OF BLANKET

The neutronic investigations aré performed on an experimental hybrid blanket design concept in order to allow comparison of the integral neutronic data with previous work [7-10]. Figure 1 shows the blanket geometry for neutronic investigations.

The fissile zone is composed of typical CANDU fuel rods with Zircolay cladding and contains spent fuel with an isotopic composition that corresponds to the highest burnup rate of 12252 MWDd/t for a CANDU pressurized heavy water reactor (PHWR) operating on a once-through natural uranium cycle, as indicated in ref [11].

The spent fuel rods are arranged in the fissile zone hexagonally.

Figure 2 depicts the cross section view of the fast fissioning blanket with air cooling.

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

III. BLANKET PERFORMANCE

A neutronic analysis has been carried out with the help of the ANISN neutron transport code [12] in S_4-P_1 approximation using the neutron group data of the GAT library [13]. This library has a fine group resulation for lower neutron energies to allow proper treatment of neutron thermalization. The 200 group structure of the original library is reduced to 97 energy groups (47 fast +50 thermal groups below 2.3 eV), which has treated the resonance self shielding effects by Nordheim's Integral Methods [14,15].

To study major temporal effects in the blanket, it is assumed that the fuel zone is irradiated with a first wall neutron load of 2.25 MW/m^2, with a plant factor (PF) of 75%.

The temporal variations in the fuel composition under this neutron flux load are evaluated by considering the following reactions:

1. Breeding reactions (σb):

   $^{238}U\ (n,\gamma)\ ^{239}Pu$
   $^{239}Pu\ (n,\gamma)\ ^{240}Pu$
   $^{240}Pu\ (n,\gamma)\ ^{241}Pu$
   $^{241}Pu\ (n,\gamma)\ ^{242}Pu$

2. Depletion reactions (σ dep) for ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{241}Pu and ^{242}Pu.

Figure 3 shows the cumulative fissile isotopes during plant operation. It is possible to draw from this figure the following conclusions:

1. The CANDU spent fuel becomes reusable in a natural uranium CANDU reactor after a regeneration time of about 6 to 12 months.

2. A regeneration period of 10-16 months is necessary to increase the cumulative fissile fuel content to about 1%.

3. The advanced CANDU breeder concepts with thorium-fueling require an average enrichment of $> 1.5\%$ at start-up [11]. This grade of regeneration occurs in the investigated hybrid blanket after a plant operation period of about > 18 months.

Figure 4 to 6 shows the variation of the isotopic enrichment grades of Plutonium in the spent fuel in the cours of the regeneration process. The ^{239}Pu enrichment increases gradually, but approaches to an assymptotic value far below 90%. The enrcihment grade of ^{240}Pu decreases, but remains at each fuel rod position above 10% throughout the regeneration period. Hence, the spent CANDU fuel keeps its denaturated character during a regeneration in a hybrid blanket, and does not even approach the grade for any violation of the non-proliferation stand point. The Plutonium fuel would cause a problem of proliferation, only if the ^{239}Pu enrichment would increase beyond 95%, and consequently the ^{240}Pu enrichment would decrease below 5% [16,17].

Figure 7 shows the average burn-up rate of the fissile fuel in the hybrid blanket as a function of plant operation period. One can see that the burn-up rate remains rather modest during the process of spent fuel regeneration in the hybrid blanket. Hence, one can conclude easily that it will be possible to repeat the process of fuel regeneration in the hybrid reactor followed by a re-utilization of the same fuel rods directly in a critical reactor over several cycles. This reduces the cos for fuel reprocessing drastically.

IV. DISCUSSION

The main conclusions of the present study on the possibility of regenerating CANDU spent fuel in a hybrid blanket can be cited as blanket follows:

1. A fast fissioning hybrid blanket using CANDU spent fuel a very good neutronic performance. It can produce electricity in the course of regenerating the CANDU spent fuel for different purposes.
2. The regeneration period of spent fuel is relatively short for a reuse in a conventional CANDU reactor (6 to 9 months). This period increases for an applications in advanced CANDU reactor concepts.
3. The spent fuel remains denaturated in the period of regeneration.

REFERENCES

[1] LEGGERIT, R.D., OMBERG, R.P. : Mixed Oxide Fuel Development, Proceedings of the International Conference on Fast Breeder Systems, Pasco, Washington (1987).

[2] WALTAR, A.E., DEITRICH, L.W. : Status of Research on Key LMR Sayefty Issues, Nucrear Safety, Vol.29, No.2, pp.125-148 (1988).

3] GREENSPAN, E. : Fusion-Fission Hybrid Reactors, Advances in Nuclear Science and Technology (editors J.Lewins, M.Becker), Vol.16, p.289, Plenum Press (1984).

CONN, R.W., KANTROWITZ, F. and VOGELSANG, W.F. : Hybrids for Direct Enrichment and Self-Protected Fissile Fuel Production, Nuclear Technology, Vol.49, p.458 (1980).

[5] BROOKS, G.L.: Advances in Commercial Heavy Water Reactor Power Stations, Proc.from the 6th Pasific Basin Nuclear Conference, Trans.Am. Nucl.Soc., Supl. to Vol.56, p.41 (1988).

[6] Status and Prospects of Thermal Breeders and their effects on Fuel Utilization, Technical Report Series No:195, International Atomic Energiy Agency, Vienna (1979).

[7] ŞAHİN, S., AL-ESHAIKA, M. : Fission Power Flattening in Hybrid Blankets Using Mixed Fuel, FUSION TECHNOLOGY, Vol.12, No.2. pp.315-354 (1987).

[8] ŞAHİN, S., ERİŞEN, A., ÇEBİ, Y. : Realization of a Flat Fission Power Density in a Hybrid Blanket Over Long Operation Periods, FUSION TECHNOLOGY, Vol.15, pp.37-48, (1989).

[9] ŞAHİN, S. : Power Flattening in a Hybrid Blanket Using Nuclear Waste Actinides, KERNTECHNIK, Vol.53, No.4, pp.285-290 (1989).

[10] ŞAHİN, S., YAPICI, H. : Rejuvenation of CANDU Spent Fuel in a Hybrid Blanket, Trans. of the American Nuclear Society 1989 Annuel Meeting, Vol.59, pp.105-106, Atlanta (June 4-8, 1989).

[11] DATA BASE for a CANDU-PHW operating on a once-through natural uranium cycle., Atomic Energy of Canadian Limited AECL-6593 (1979).

[12] ENGLE, W.W., Jr. : ANISN, A One-Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering; K-1693, Oak Ridge National Laboratory (1970).

[13] ADIR, J. and LATHROP, D.: Theory of Methods Used in GGC-4 Multigroup Cross Section code GA-9021. General Atomic (1968).

[14] NORDHEIM, L.W. : A Program of Research and Calculations of Resonance Absorption, GA-2527, General Atomic (Aug.1961).

[15] NORDHEIM, L.W. : The Theory of Resonance Absorption, Proc. Symp. Applied Mathematics, Vol. 1, (1961).

[16] ŞAHİN, S., LIGOU, J. : The effect of the spontaneous fission of Pu-240 on the energy release in a nuclear explosive Nuclear Technology, Vol. 50, no.1, pp.88-94 (1980).

[17] ŞAHİN, S. : Nonproliferation, Nature, Vol.287, No.5783, p.578 (16 October 1980).

FIGURE CAPTIONS

Figure 1 : Hexagonal arrangement of the blanket with air cooled fuel rods forming ten rows in the radial direction.

Figure 2 : Cross-sectional view of the investigated blanket.

Figure 3 : Accumulation of the fissile isotopes in different fuel zones of the air-cooled fast blanket during a 2 years plant operation period.

Figure 4 : Temporal variation of the enrichment grade of 239Pu in the spent fuel during regeneration period.
A:Adjacent to the firs wall
B:Centre of the fuel zone
C:Adjacent to the firs Li2O zone

Figure 5 : Temporal variation of the enrichment grade of 240Pu in the spent fuel during regeneration period.

Figure 6 : Temporal variation of the enrichment grade of 241Pu and 242Pu in the spent fuel during regeneration period.

Figure 7 : The burn-up rate of the spent fuel during the regeneration process in the hybrid blanket.

TABLE I

Dimensions and Material Composition of the Blanket Zones

ZONE	MATERIAL	DIMENSION (CM)	NUCLIDE	NUCLEI DENSITY (a) (10e+30/M3)
Cavity	Air	0 to 18.7	-	-
First Wall steel	Type 316 stainless (b)	18.7 to 20		
Fuel	CANDU spent fuel	20 to 35	Zircolay	6.07936E-3
			Oxygen	2.75098E-2
			Hydrogen	-
			U-235	1.48003E-5
			U-236	1.22668E-5
			U-238	1.36573E-2
			Np-237	6.99437E-7
			Pu-239	3.75646E-5
			Pu-240	2.42361e-5
			Pu-242	2.90366e-6
Tritium Breeding	L_2O	35 to 47	Li-6	4.63794E-2
			Li-7	5.70367E-2
			Oxygen	3.08374E-2
			Aluminum	3.01356E-3
Reflector	Graphite	47 to 51	Carbon	1.12840E-1
Tritium Breeding	Li_2O	51 to 56	(ibid)	(ibid)
Reflector	Graphite	56 to 62		
Tritium Breeding	Li_2O	62 to 66	(ibid)	(ibid)
Reflector	Graphite	66 to 82		

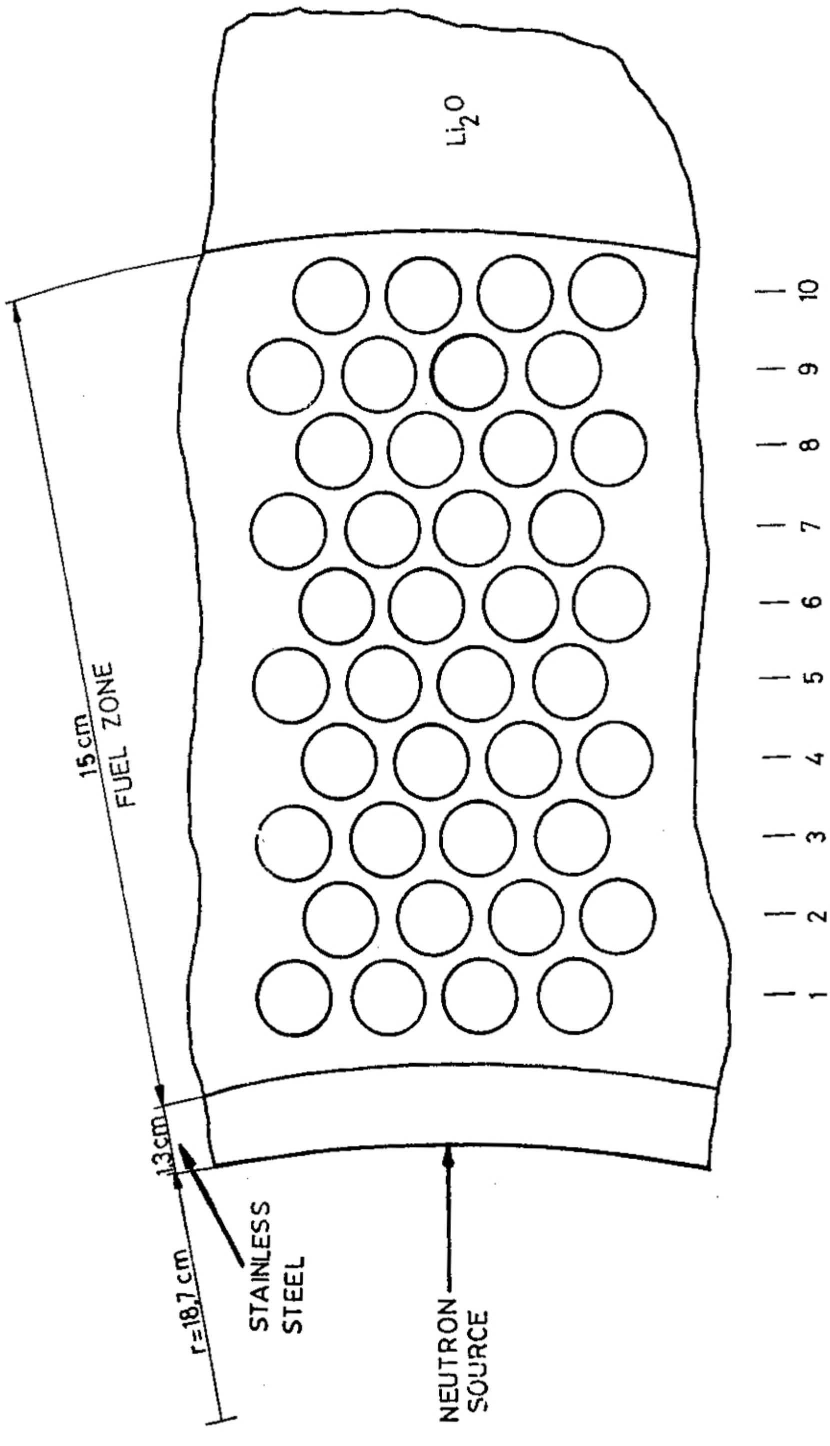

Figure 1 : Hexagonal arrangement of the blanket with air cooled fuel rods forming ten rows in the radial direction.

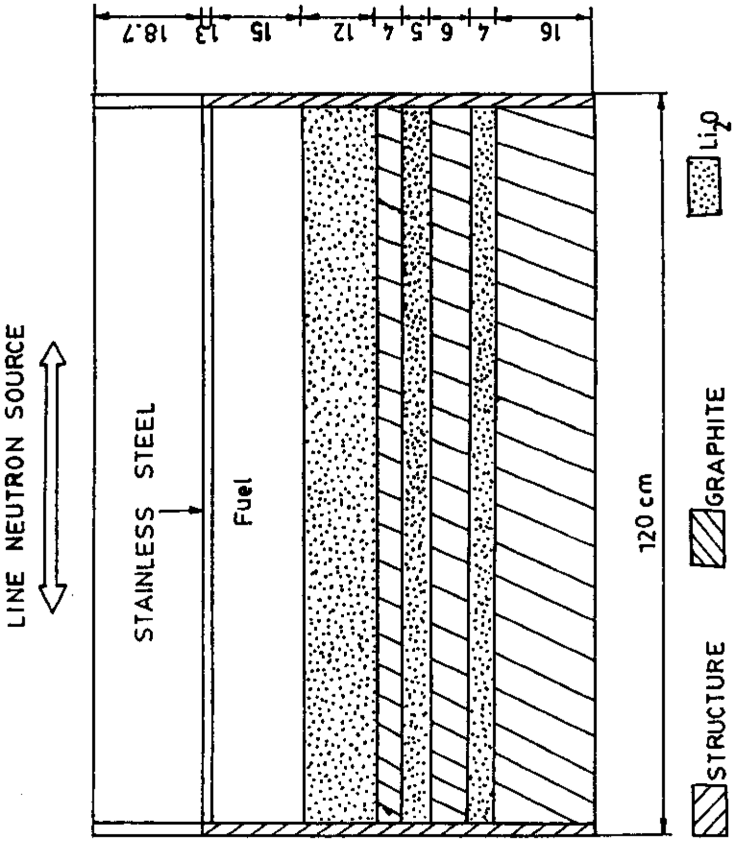

Fig.2 Cross-sectional view of the investigated blanket.

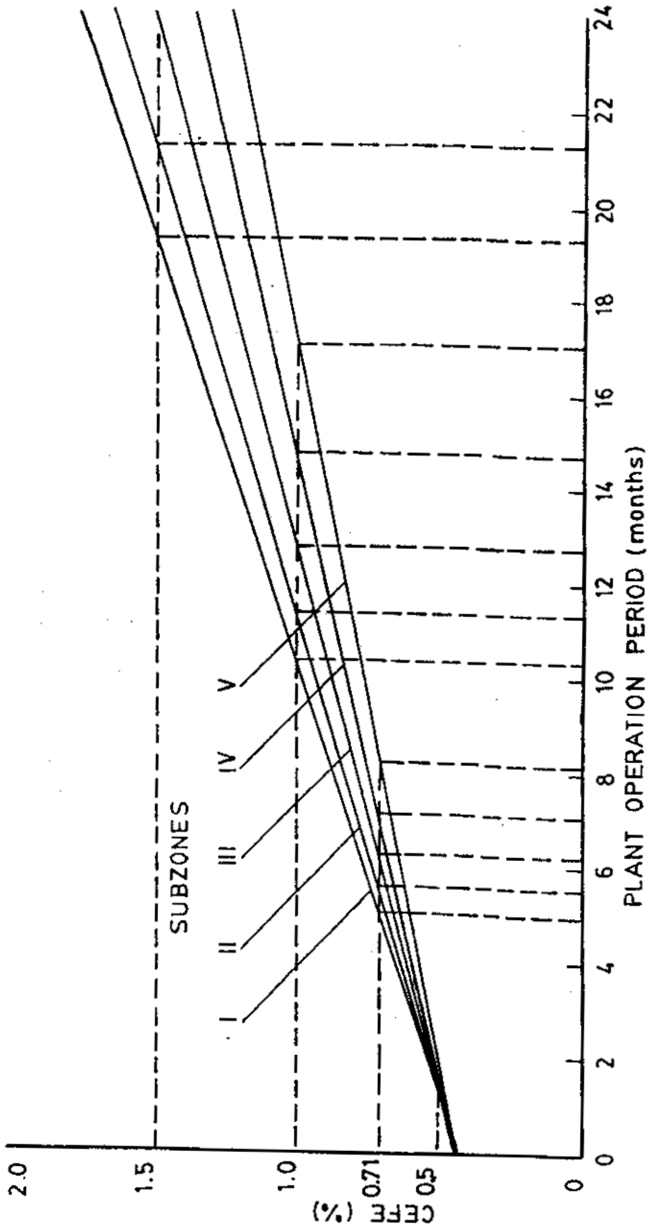

Fig.3 Accumulation of the fissile isotopes in different fuel zones of the air-cooled fast blanket during a 2-yr plant operation period.

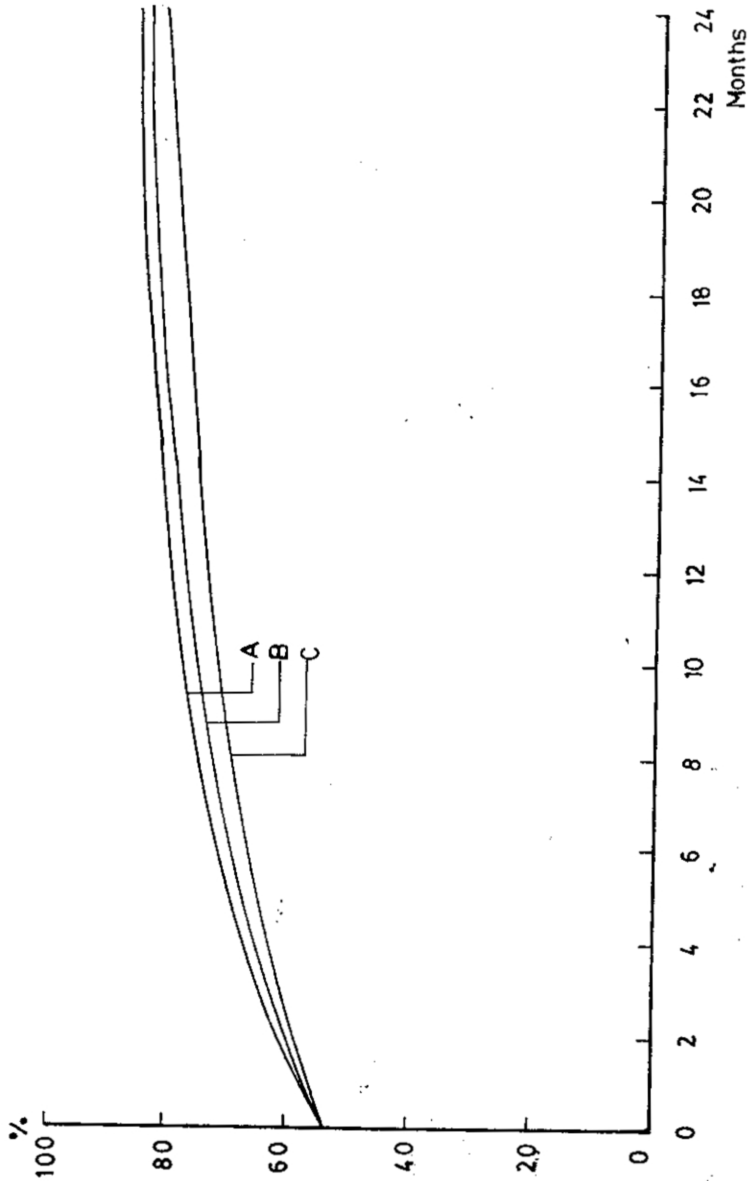

Fig.4 Temporal variation of the enrichment grade of Pu-239 in the spent fuel during regeneration period.
A: Adjacent to the first wall
B: Centre of the fuel zone
C: Adjacent to the first Li_2O zone

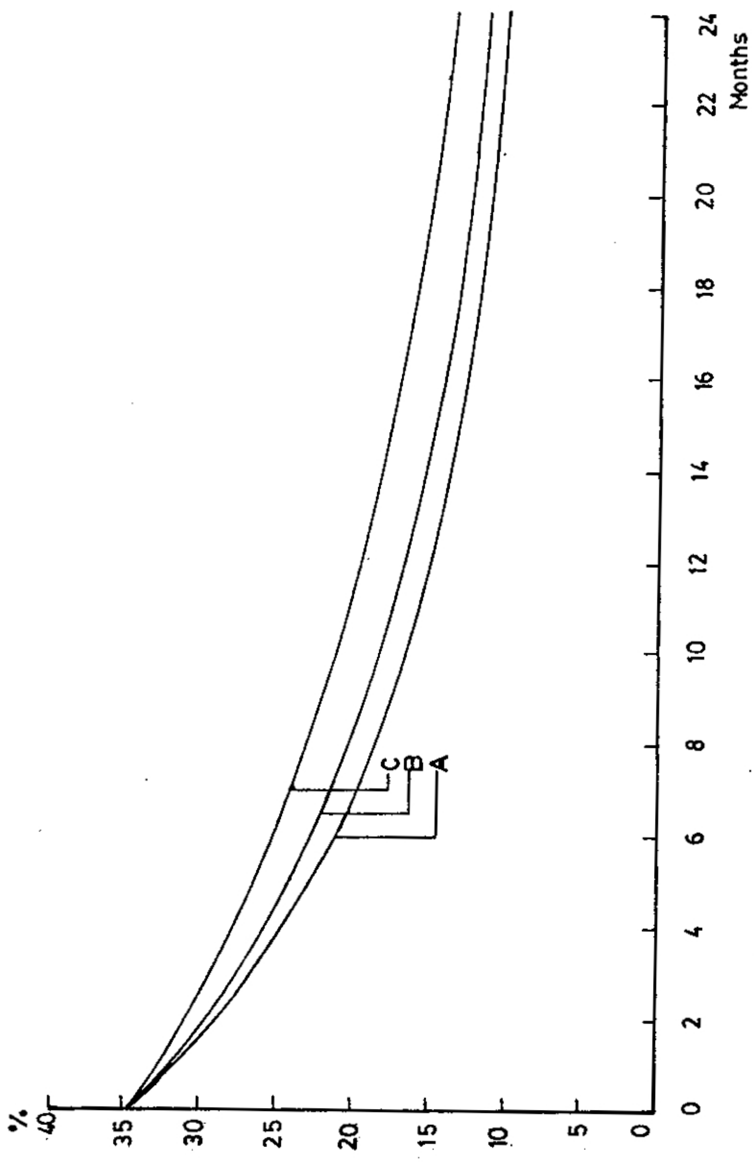

Fig. 5 Temporal variation of the enrichment grade of Pu-240 in the spent fuel during generation period.(Legend see Fig. 4)

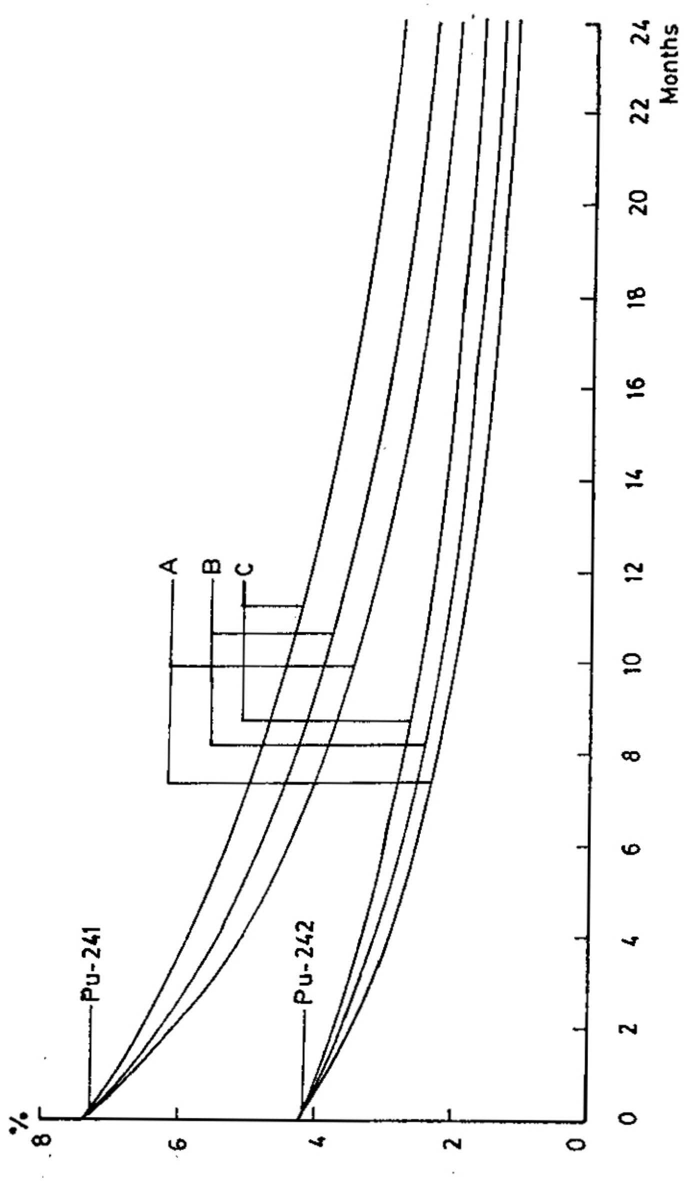

Fig.6 Temporal variation of the enrichment grade of Pu-241 and Pu-242 in the spent fuel during regeneration period.(Legend see Fig.4)

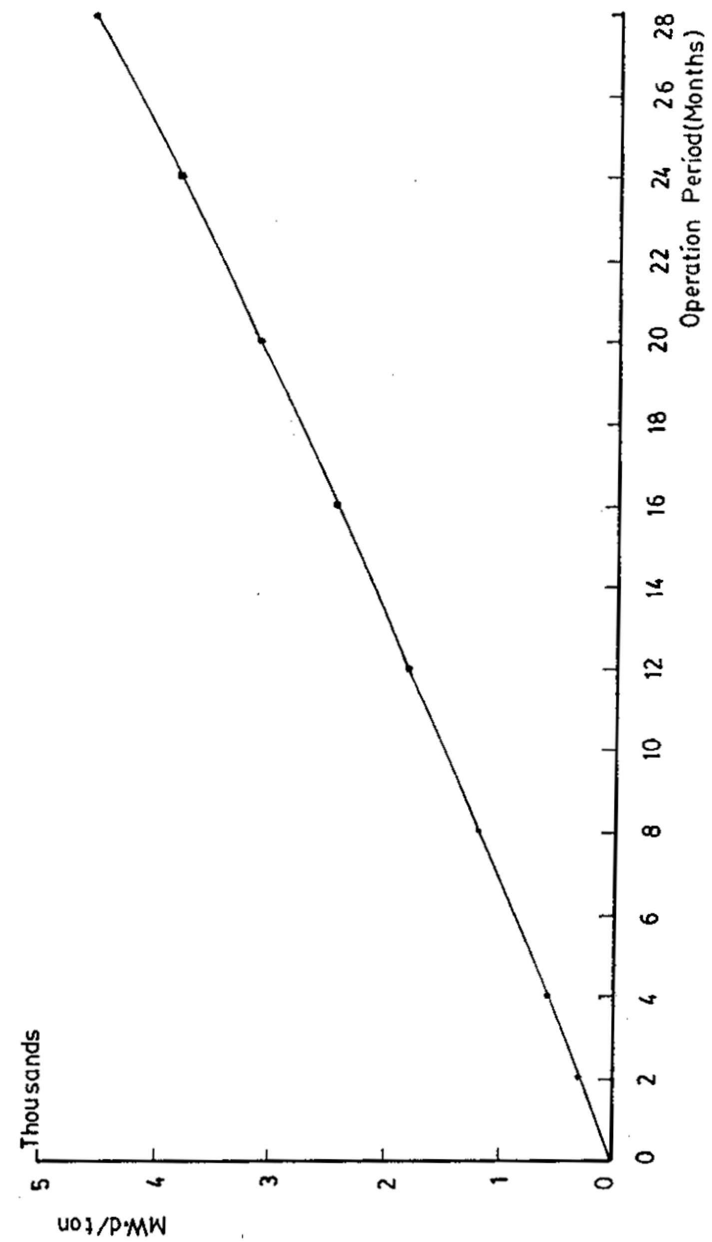

Fig.7 The burn-up rate of the spent fuel during the regeneration process in the hybrid blanket.