Energy Conversion and Management 49 (2008) 2032–2046

Contents lists available at ScienceDirect

Energy Conversion and Management

journal homepage: www.elsevier.com/locate/enconman

Reactor physics ideas to design novel reactors with faster fissile growth ☆

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ARTICLE INFO

Article history:
Received 8 July 2007
Accepted 25 February 2008
Available online 14 April 2008

Keywords:
Fast reactor
Thorium breeders
Fissile growth
Two year cycle length
Minimum control maneuvers

ABSTRACT

There are several types of fission reactors operating in the world adopting generally the open fuel cycle which considers the naturally available fissile nuclide, viz., ^{235}U. The accumulated discharged fuel is considered as waste in some countries. However the discharged fuel contains the precious man-made fissile plutonium which would provide the sole means of harnessing the nuclear energy from either depleted uranium or the natural thorium in future. It must be emphasized that the present day power reactors use just about 0.5% of the mined uranium and it would be imprudent to discard the rest of the mass as waste. It is therefore necessary to explore ways and means of exploiting the fertile mass which has the potential of providing the energy without the green house effects for millennia to come. This has to be done by innovating means of large scale fertile to fissile conversion and then using the man-made fissile material for sustenance as well as growth of fission nuclear power. This paper attempts to give a broad picture of the available options and the challenges in realizing the theoretical possibilities.

1. Introduction

The present day power reactors use mostly uranium based fuel. ^{235}U is the only natural fissile isotope serving as a match-stick to start the fission chain reaction process. It is necessary to breed new fissile material long before the ^{235}U would get exhausted. Nuclear reactors have the unique property of rejuvenating themselves, i.e., when fissile materials are consumed for power generation, new fissile materials can also be produced from neutron capture by fertile materials. When the latter process equals, or better, exceeds the former we can have continuous stock of fissile material for several centuries or even millennia. The research on accelerator driven subcritical system (ADSS) are being pursued vigorously world over to use the spallation source of neutrons as a means of rapid fissile growth. The accelerator technology, sustenance of high current ion beam, ion beam focusing, design of spallation target size and shape, effective heat removal in the surrounding blanket region near the intense neutron source and maintenance of reasonably low and constant subcriticality in the surrounding reactor medium are all challenges still defying satisfactory solutions. It is therefore useful to look into the principles of maximizing the fissile conversion process in the existing or some innovative variant version of nuclear reactors, which may also be suitable for ADS application eventually.

The basic principles used in the physics design of power reactors so far is to minimize the quantum of fuel required for a given power generation. The design parameters gets more or less converged to a narrow range once the choice of type of reactor, say fast or thermal is made and the choice of coolant, moderator, structural materials and control are made accordingly. In thermal power reactors using natural uranium as fuel the minimum fissile mass is obtained by choosing optimum fuel assembly pitch or V_m/V_f for the fresh state of the fuel. In thermal reactors using enriched fuel, a choice of V_m/V_f less than the optimum value is preferred to ensure negative coolant void or temperature coefficient of reactivity. However there may not be any loss in the discharge burnup since due to hardened neutron spectrum the fissile conversion ratio gets improved and at the end of life, the reactivity may, in fact, be higher than that for an optimum V_m/V_f. This advantage is absent in the reactors using natural uranium. In fast reactors all cross-sections tend to be one or two orders of magnitude lower and it is necessary to use nearly 5–10 times fissile content even for criticality. The fuel cost of fast reactors is thus apparently dearer than that of thermal power reactors, and not surprisingly, therefore, there are very few operating fast reactors, perhaps merely by way of demonstration of technology. It must be however recognized that fast reactors have to wait for the accumulation of the necessary stockpile of Pu from thermal power reactors.

^{232}Th and depleted uranium (^{238}U) are equal candidates for breeding of fissile material since they produce, respectively, ^{233}U

☆ Paper Submitted to: ICENES 2007 13th International Conference on Emerging Nuclear Energy Systems, 3–8 June, 2007, Istanbul, Turkey.

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doi:10.1016/j.enconman.2008.02.019

and ^{239}Pu on neutron capture and subsequent two 'β' decays. The ^{233}U isotope has an excellent neutronic characteristic. It has the least capture to fission ratio and hence the 'η' value, which is the number of fission neutrons produced per neutron absorbed in the fissioning atom, remains close to or above 2.3, in the entire energy range, except near 2 eV. Fig. 1 gives a comparison of the 'η' values for the three nuclides ^{235}U, ^{239}Pu and ^{233}U. It is this property of ^{233}U which opens the possibility of a thermal breeder. However in the fast energy range, the 'η' value of ^{239}Pu is much higher and hence is a preferred nuclide for fast breeder reactors (FBR).

Fast reactors are believed to be natural breeders since they have the best rejuvenating potential, i.e., they can produce more fissile material than what they consume over a period of time. This is facilitated by two distinguishing features in a fast reactor. One is large scale loading of pure fertile mass in what is called the 'blanket' regions and the second is, despite using highly enriched fuel, the level of neutron flux is higher by at least one order of magnitude due to low cross-sections compared to thermal reactors of comparable power levels. In this context, it must be noted that there is a net fissile material loss in the core region which is sought to be made up by the production of fissile atoms in the blanket region by capturing of neutrons leaking out of the core. Since the flux level in the blanket region is usually lower by 3–6 times and the average capture cross-section of fertile atom like ^{238}U is lower than the absorption cross-section of fissile atom like ^{239}Pu by similar order, the replenishment of fissile loss may be feasible only by using much larger fertile mass. The fertile mass in blanket region is typically more than twice that of the core mass in fast reactors. The blanket region surrounds the core both radially and axially. But for the blanket regions, even the fast reactors cannot have the breeding potential, in spite of the fact that the value of 'η' is about three in the fast energy range.

Before we discuss the means of enhancing the fissile conversion rate in some hypothetical reactor, we shall attempt to qualitatively inter-compare the conversion rates in different types of thermal power reactors and fast reactors which are already designed and operational.

Among the thermal power reactors, pressurized heavy water reactors (PHWR) have the highest fissile conversion rate since the flux level is high and there is least competition for absorption of neutrons from the small and depleting seed content of 0.71% or less of ^{235}U. In Light Water Reactors (LWR) using enriched uranium the conversion rate is lower for similar reasoning. However the fuel residence time in LWR is much longer, which is important for prolonging the duration of cumulative fertile capture. In PHWR, under nominal conditions, the fuel is discharged after 6–12 months of residence time. In LWR the residence time can be 3–4 years. In all reactors the conversion rate is the highest at the time of discharge since the competition of absorption of neutrons from the seed content is the least. The energy extracted from unit mass is called the discharge burnup and compares as 7 MWD/kg for PHWR using natural uranium and 35 or 43 MWD/kg in LWRs using 3.3% (western pressurized water reactors – PWR) or 4% (Russian VVER) enriched fuel, respectively. The discharge Pu content in PHWR is under 3.5 g/kg, while in LWR it is 10–11 g/kg. For the same quantum of energy generation, the PHWRs can produce more than double the quantity of Pu compared to LWRs. However, this Pu would be contained in 6–8 times larger discharged fuel mass. Hence due to lower specific Pu content the reprocessing loss of Pu in case of PHWR would be higher. The residual non-depleted ^{235}U in the discharged fuel of PHWR is around 2–3 g/kg while in LWR it is about 10 g/kg.

In fast reactor the initial fissile feed in the core region is typically 20–30% of reactor grade Pu with 74% fissile. After extraction of energy of the order of 100 MWD/kg the initial fissile seed content would have decreased by 10% and would have been partially replenished by fissile conversion in the core region to the tune of 4–5%. Thus the discharge fuel would still contain 15–24% of seed material with somewhat degraded fissile fraction. As will be seen in the ensuing discussions, the asymptotic fissile accumulation potential in fertile blanket region is 90–100 g/kg in a typical fast neutron spectrum with peak near 100 keV. In practice, however, the blanket fuel may be discharged much earlier due to provision of less coolant flow compared to core region and the blanket

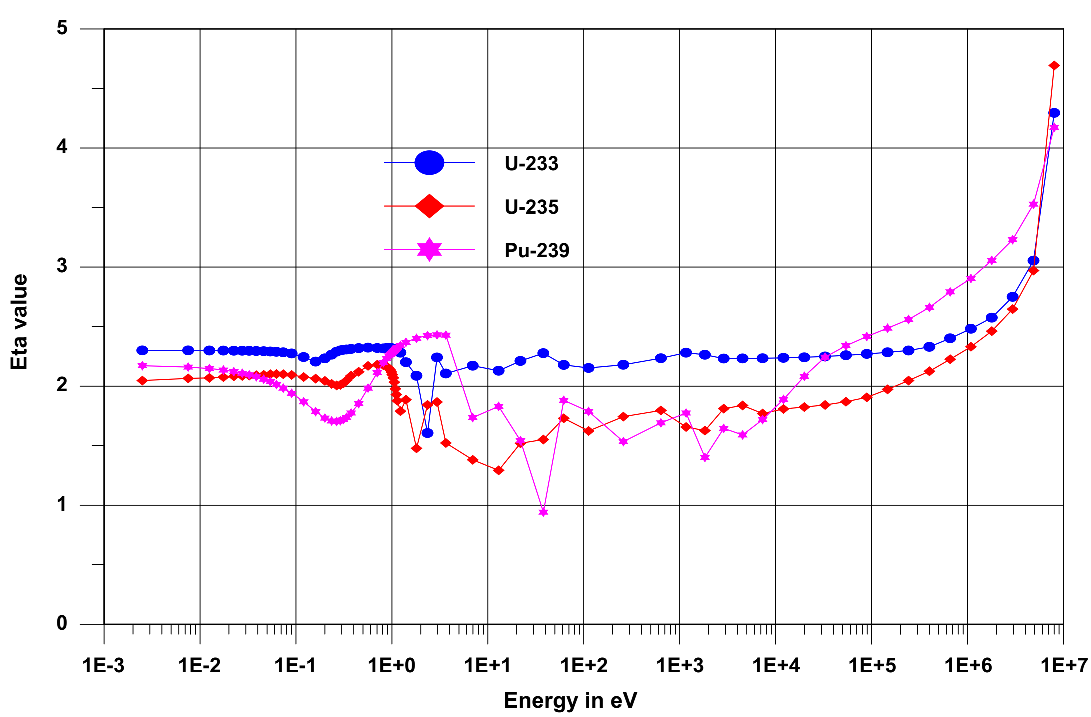

Fig. 1. Comparison of η-values of ^{233}U, ^{235}U and ^{239}Pu.

discharge criteria is based on the maximum power allowable as per the flow provided in the respective region.

From the above discussion, it is possible to identify the parameters which would help in maximizing the conversion rate of fertile to fissile and also enable accumulation of as large an amount of fissile material after a given quantum of fission energy generation. The fertile capture is proportional to the volume or mass of fertile material loading, prevalent flux level, the spectrum averaged capture cross-section of fertile nuclide and the duration of the fuel cycle campaign. One must attempt to maximize each of these parameters individually.

Depleted uranium and pure thorium are the best fertile materials to allow high conversion rates. PHWR using natural uranium has very small excess reactivity and hence can ill-afford any fertile mass loading, except in its initial core. They have the least parasitic absorption in D_2O coolant/moderator. However it considers large mass of zirconium as structural material, which may somewhat off-set the above advantage. Boiling water reactors (BWR) have higher conversion rate compared to PWR due to harder neutron spectrum. High conversion PWR with tight lattice pitch was seriously considered in nineties, but was found to have positive coolant temperature coefficient. The conversion ratios of PHWR and PWR compare as ~0.7 and ~0.6, respectively.

In the fast reactor the blanket region is usually beyond the active core. The blanket region receives the neutrons leaking out of core. Also due to leakage at the outer boundary of blanket, the flux level would continuously diminish radially and axially outward. Hence fissile formation rate per unit mass in blanket region would be a fraction of fissile destruction rate per unit mass in the core region. The seed refueling schedule and the blanket refueling schedule may not match. In view of this mismatch in fissile depletion and formation rates, the fast reactors may sometimes reduce to mere converters rather than breeders or may have unacceptable large doubling time for the fissile material hold ups both in pile and out of pile.

In fast spectrum, fission cross-section of ^{232}Th is lower than that of ^{238}U by about four times. The absorption and fission cross-sections of ^{233}U are lower by factor of two in thermal energy range compared to those of ^{239}Pu, while they are higher for ^{233}U in fast energy range. When thorium or depleted uranium is irradiated in thermal, intermediate or fast neutron spectra, the formation of U or Pu would monotonically increase and finally saturate to an asymptotic value. This asymptotic fissile accumulation potential in any energy range can be approximately inferred from the condition that:

$$\sigma_c^{232} N^{232} = \sigma_a^{233} N^{233}$$
$$\sigma_c^{238} N^{238} = \sigma_a^{239} N^{239}$$

which gives,

$$N^{233}/N^{232} = \sigma_c^{232}/\sigma_a^{233}$$
$$N^{239}/N^{238} = \sigma_c^{238}/\sigma_a^{239}$$

Here N^{233}, N^{232} are the nuclide concentrations of ^{233}U and ^{232}Th, $\sigma_c^{232}, \sigma_a^{233}$ are the effective microscopic capture and absorption cross-sections of ^{232}Th and ^{233}U, respectively in a given spectrum. N^{239}, N^{238} are the nuclide concentrations of ^{239}Pu and ^{238}U, $\sigma_c^{238}, \sigma_a^{239}$ are the effective microscopic capture and absorption cross-sections of ^{238}U and ^{239}Pu, respectively, in a given spectrum. Fig. 2 gives plot of these ratios in the entire energy range. It is seen that the ^{233}U can accumulate up to 1.3% (13 g/kg), while ^{239}Pu can accumulate up to 0.35% (3.5 g/kg) in thermal energy range. In intermediate energy range, these ratios show several spikes reaching up to 20% corresponding to the sharp resonances of the capturing nuclide. The ratio is seen to be higher for thorium capture in the energy range 100 eV to 10 KeV. The actual asymptotic fissile content can be obtained from the spectrum average cross-section values in a given reactor application.

A hypothetical study was done to observe the fissile accumulation in fertile fuel rods irradiated in a typical thermal or fast flux ambience at different flux and fluence values. Fig. 3a shows the plot of fissile formation in thorium and natural uranium fuel rods exposed to a constant one group flux level of 10^{14} n/cm^2/s in a thermal reactor ambience. It is seen that the U^{total} in Th accumulates up to 17 g/kg with 85% fissile in a typical thermal spectrum, while the Pu^{total} accumulation in U is seen to be under 7 g/kg with 60–65% fissile, after irradiation time of 1500 days. Fig. 3b shows similar plot for thorium and depleted uranium fuel rods irradiated at constant one group flux level of 1×10^{15} and 4×10^{15} n/cm^2/s in a fast reactor ambience. As seen in Fig. 3b, in fast neutron spectrum

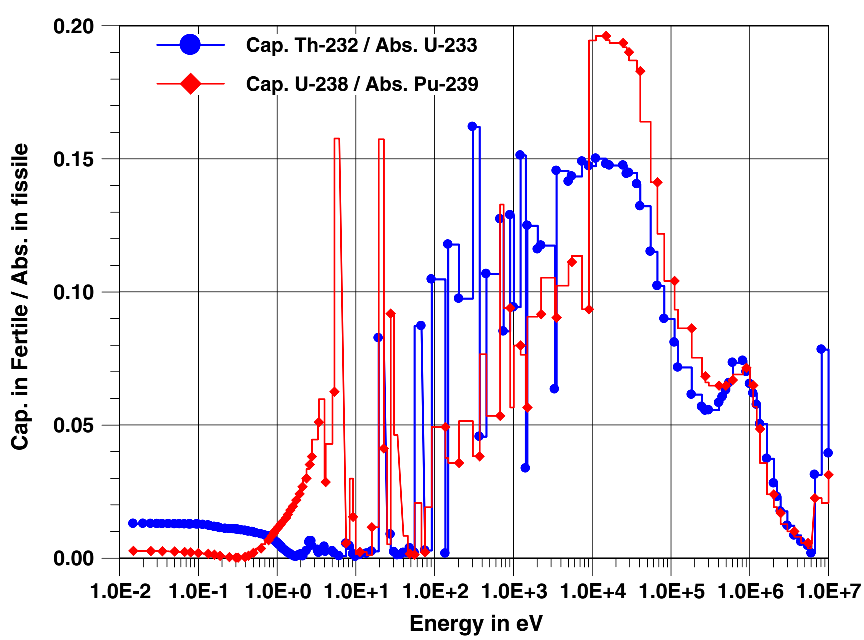

Fig. 2. Asymptotic fissile accumulation potential in each energy group.

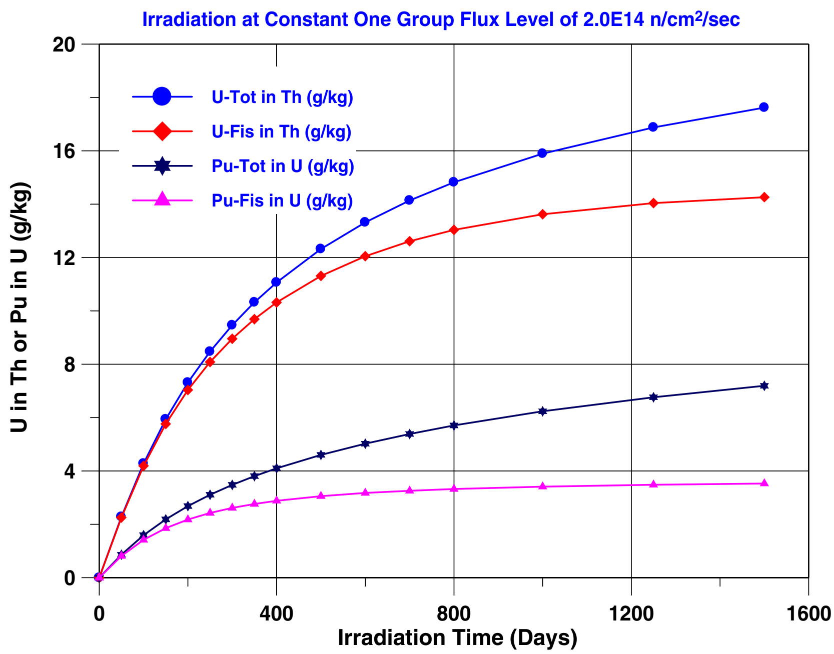

Fig. 3a. Production of U from Th or Pu from U in A thermal spectrum.

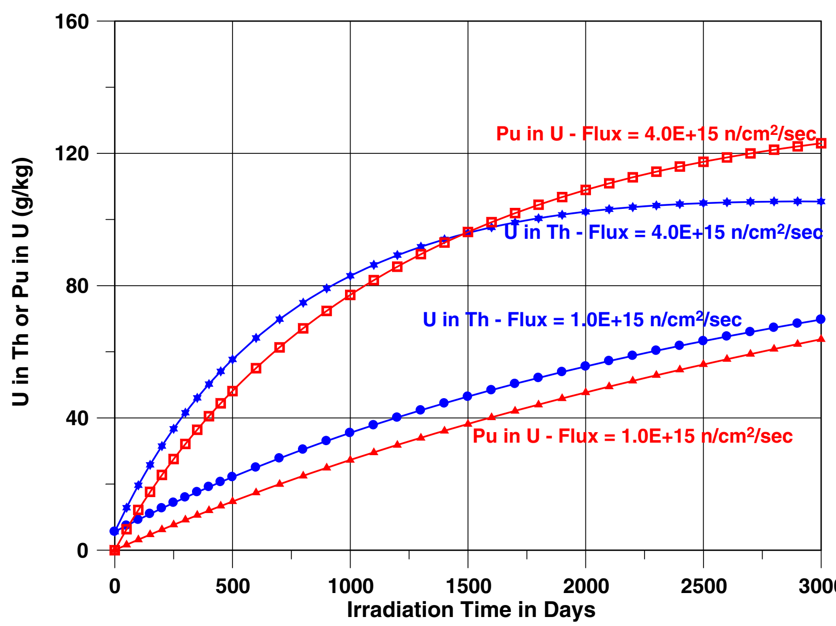

Fig. 3b. Production of U from Th or Pu from U in a fast spectrum. (*Note.* Thorium rods are pre-irradiated for a fluence of 2×10^{14} n/cm^2/s for 700 days).

both U in Th and Pu in U accumulate to about ~100 g/kg, the value being slightly larger for Pu in U, albeit with lesser fissile content. Unlike in thermal spectrum, the fissile fraction remains above 90% for very large fluence levels up to 1000 days or so at the above flux levels.

One of the major characteristics of fast reactors is relatively high power density which results in high neutron flux level, rapid fuel depletion and short refueling intervals. However, a low neutron flux would enable in decreasing the depletion rate of seed fuel and hence a longer fuel cycle duration for a given reactivity inventory. Since fertile zones reach nearly the same asymptotic fissile content, albeit at different times, a lower flux ambience is preferable from the following considerations. A core design with seed to fertile fuel mass ratio of 50:50, with somewhat larger fissile content in the seed and a considerable power share from fertile blanket zones would decrease the mean power density and hence the ambient flux level. This would help in conserving the fissile content in seed zone for longer duration and allow the fertile zones to reach the asymptotic fissile content.

Two other key parameters which influence the design consideration are k_∞ and burnup variation of fertile zone after a given neutron fluence. It was seen that the k_∞ of fertile zone rises from nearly zero to about unity after an irradiation time of about 3000 days at a flux level of 2×10^{15}n/cm^2/s. In view of this characteristic of fertile zones, it should be possible by judicious mix of seed and fertile zones to obtain a core design with a small and flat core excess reactivity for as long a duration as possible. The burnup accumulated in fertile zone compare as 75–85 MWD/kg at the above fluence while the seed zone attains burnup of ~100 MWD/kg for the above flux level in about 2200 days. These features suggest that one can consider a three batch fueling with a cycle length of about 720 days where the seed fuel should reside in the core for three fuel cycles and the fertile rods should reside for at least four fuel cycles. The prevalent flux level in seed and fertile zones should be comparable.

Table 1
Core design parameters of the conceptual fast thorium breeder reactor (FTBR)

Core parameters	Units	Values
Thermal power	MWt	2500
Electric power	MWe	1000
Coolant		Sodium
Cycle length	Days	720
Cycle energy	MWD	1,800,000
Mean discharge burnup	MWD/T	100,000
Discharge mass	T	18
Fuel batch size		1/3rd core
Number of assemblies per batch		120
Number of Seed assemblies in core		360
Number of seedless blanket assemblies		120
Number of seedless blanket/control assemblies		25
Additional ThO_2 + Ni reflector assemblies		120
Hexagonal assembly lattice pitch	mm	184
Effective diameter of core + radial blanket regions	mm	4830
Number of steel reflector assemblies		96
Number of B_4C assemblies for shielding		–
Active core height inclusive of 200 mm internal blanket	mm	1500
Top axial blanket thickness	mm	300
Bottom axial blanket thickness	mm	300
Average linear heat rating	w/cm	150.8
Description of seeded fuel assemblies (360)		
Inner region of the seeded fuel assembly		
Number of seed fuel rods		217
Seed pellet diameter	mm	5.7
Steel clad OD	mm	6.6
PuO_2 seed content in Dep. UO_2	%	45
Fuel density (oxide)	g/cm^3	10
Hexagonal pin pitch	mm	8.3
Inner steel channel inner/outer dimension	mm	126/132
Number of fuel cycles		3
Outer region of the seeded assembly		
Number of irradiated fertile Dep. U fuel rods		90
PuO_2 Seed content	%	In situ (4–5%)
Pellet diameter	mm	8.8
Steel clad OD	mm	9.8
Hexagonal pin pitch	mm	12
Fuel density (oxide)	g/cm^3	10
Outer steel channel inner/outer dimension	mm	176/182
Number of fuel cycles		(1 + 3)
Description of seedless fertile blanket assemblies (120 + 25)		
Inner region of the blanket assembly		
Number of seedless ThO_2 rods		127
Pellet diameter	mm	8.8
Steel clad OD	mm	9.8
Hexagonal pin pitch	mm	10.85
Fuel density (oxide)	g/cm^3	9
Inner steel channel inner/outer dimension	mm	126/132
Number of cycles in this location		1–3
Outer region of the blanket assembly		
Number of Dep. U blanket fuel rods		90
Pellet diameter	mm	8.8
Steel clad OD	mm	9.8
Hexagonal pin pitch	mm	12
Fuel density (oxide)	g/cm^3	10
Outer steel channel inner/outer dimension	mm	176/182
Number of cycles in this location		1
Description of seedless ThO_2 + steel rods assemblies		
Number of such assemblies in the core		120
Location of ThO_2 rods in the assembly		Inner 127 locations
Description of ThO_2 rods		Same as above
Location of steel rods in the assembly		Outer 90 locations
Location in the core of these assemblies		Outermost layer
Number of cycles in this location for ThO_2 rods		One

It is necessary to consider as high a seed content and a fairly high discharge burnup so that the blanket regions are capable of accumulating fissile content close to the asymptotic value. In normal FBR, after accounting for leakage from core, one would still require appropriate control maneuvers for compensating the burnup reactivity swing. Here we would like to mention that the parasitic absorption in structural material, coolant and net leakage out of core/blanket regions are normally inevitable. They all cut into the availability of neutrons for fertile to fissile conversion. It must however be possible to minimize at least the control absorber inventory by suitably devising means of loading equivalent fertile

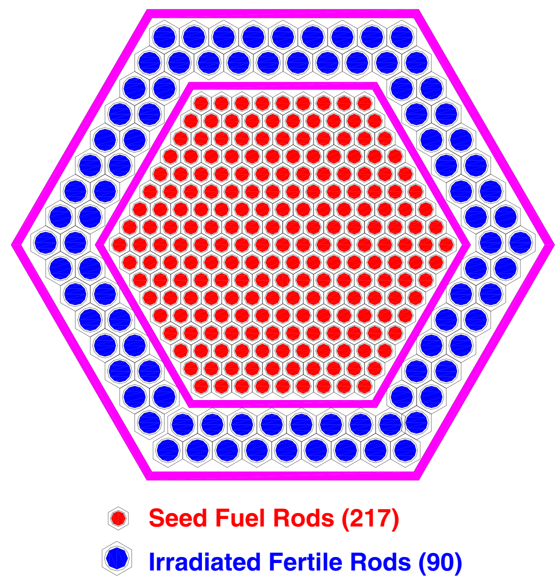

Fig. 4a. Two region – PuO_2 seeded MOX + one cycle irradiated Dep. UO_2.

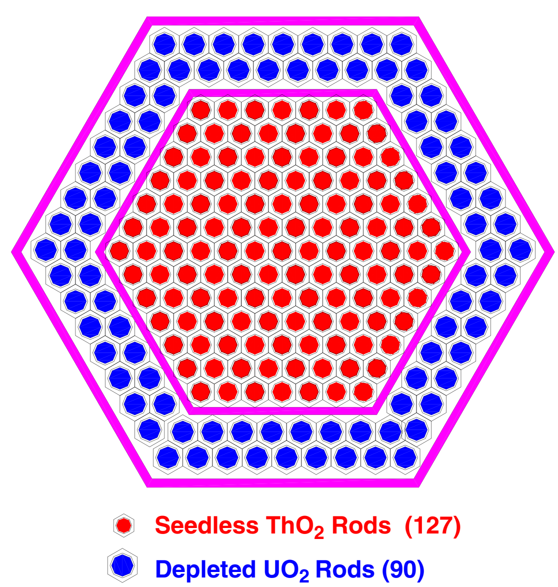

Fig. 4b. Two region seedless ThO_2 + Dep. UO_2 fuel assembly.

mass in the core. In addition, by bringing the fertile blanket region inside the active core, it is possible to achieve comparable flux levels in seed as well as blanket fuel rods. It is necessary to employ optimal fuel loading schemes to get a flat flux/power distribution throughout the core.

2. Fast thorium breeder reactor design

Fast reactors adopt a variety of fuel and coolant materials [1–3]. In India we consider oxide fuel and sodium coolant for the Prototype Fast Breeder Reactor which is under construction [4]. We have chosen the same materials for the conceptual design proposed here.

At present, there are no internal fertile blankets or exclusive fissile breeding zones in power reactors operating in the world. Loading of seedless thoria rods as inner blanket region was proposed by us in the thorium breeder reactor concept ATBR [5,6]. In this concept, it was shown that the breeding of ^{233}U in internal blanket zones could be utilized for achieving a fuel cycle duration of nearly two years (720 days) for a reactor power level of 600 MWe or 1875 MWt with no refueling and no major external control maneuvers. Applying the design principles of ATBR in evolving the conceptual fast thorium breeder reactor (FTBR) design was suggested earlier [7].

Table 1 gives description of the design of the FTBR core and the fuel types thereof. The reactor power is 2500 MWth. With 40% efficiency the electric power would be 1000 MWe. A fuel cycle length

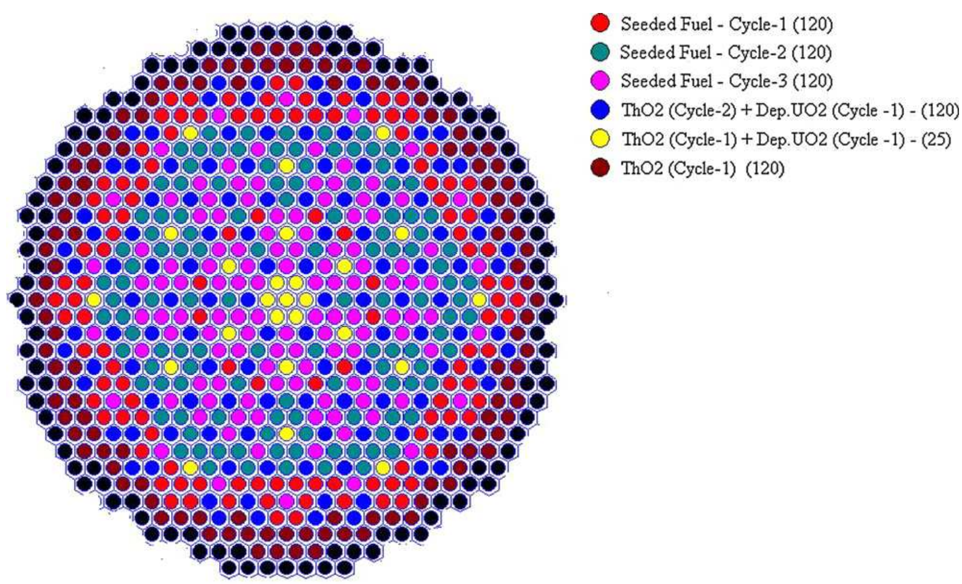

Fig. 5. Optimized three batch loading of conceptual FTBR.

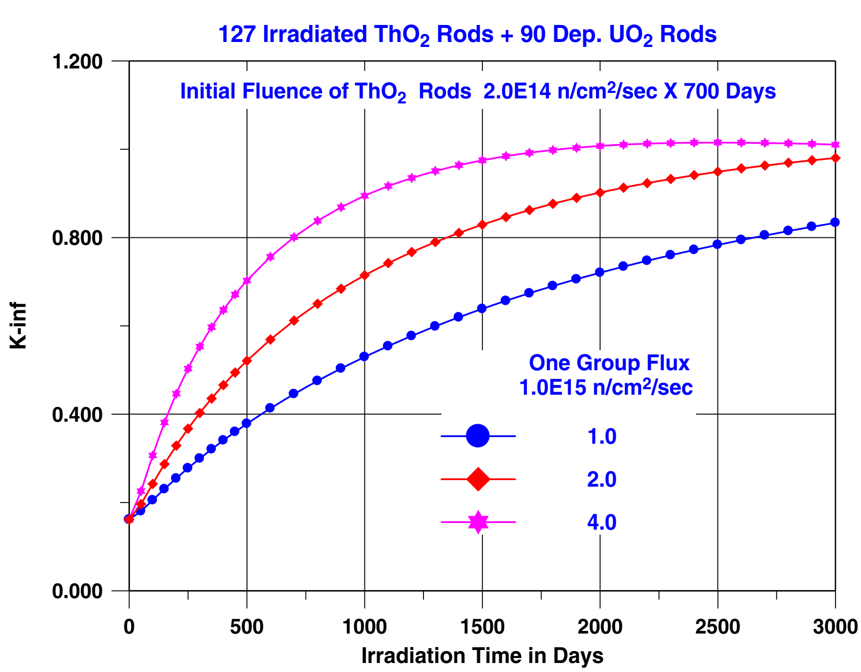

Fig. 6a. Variation of k_∞ for FTBR blanket assembly (127 ThO_2 + 90Dep UO_2 rods) as a function of fluence.

of 720 days with a three batch fueling scheme is considered. The gross energy in one fuel cycle is 1,800,000 MWD. The mean design discharge burnup is 100,000 MWD/T. The discharge fuel mass is 18 T per fuel cycle. As per the design principle of ATBR, a fuel assembly is considered to have seeded rods and fertile rods with a mass ratio of 50:50. Hence the seed and fertile mass discharged per cycle is 9 T each. A batch size of 120 fuel assemblies is considered. The core height is taken as 1500 mm. However, to limit the seed inventory and also to contain the axial peak factor, we propose an internal axial blanket region of 200 mm at axial centre symmetric to core mid-plane. This is known to help in negative or less positive coolant void reactivity in the BN-800 reactor [8]. A top and bottom blanket height is taken as 300 mm. The seed and blanket fuel rods have been taken as 217 and 90 in number. There are arranged in seven hexagonal layers for seed fuel and the blanket rods are arranged in just two layers. For the above discharge mass of 9 T, the seed fuel diameter can be estimated as 5.7 mm and the blanket fuel pellet diameter as 8.8 mm. The ratio of 1.54 in the diameters of seed and fertile rods would give a ratio of 2.4 for their cross-section area per fuel pin which is useful in

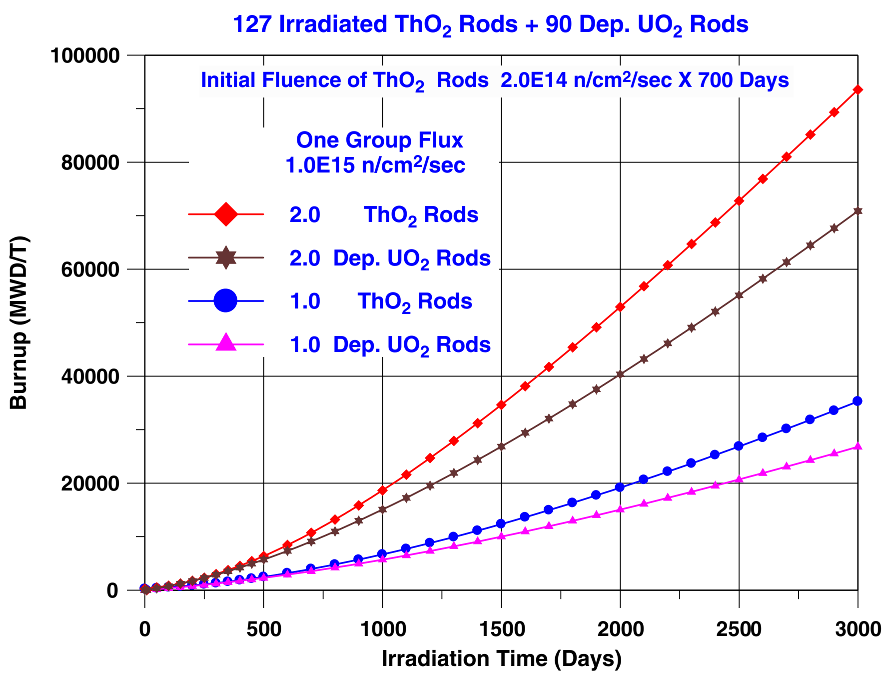

Fig. 6b. Variation of burnup for FTBR blanket assembly (127 ThO_2/90 DUO_2 rods) as a function of fluence.

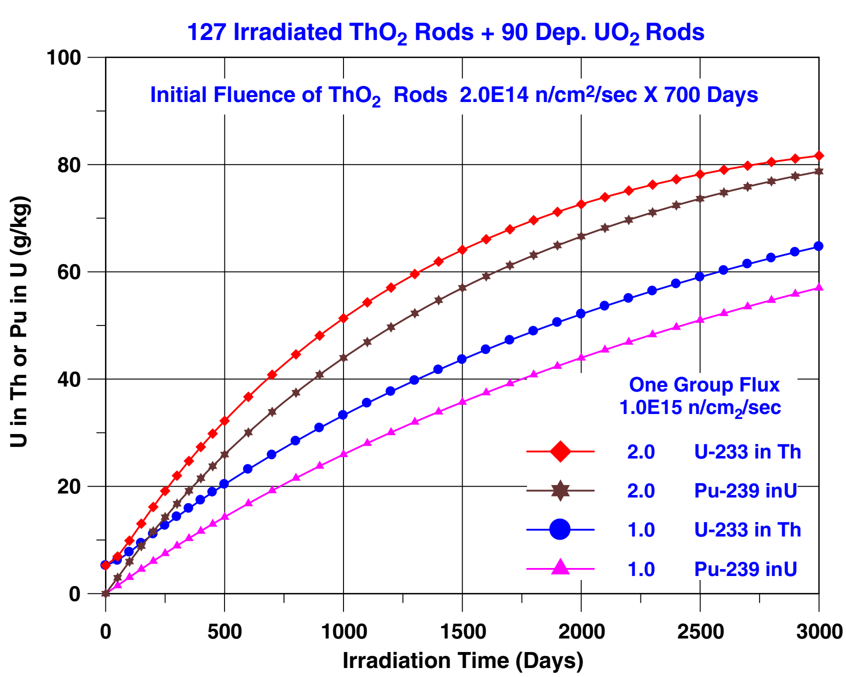

Fig. 6c. U in Th and Pu in U formation for blanket assembly as a function of fluence.

obtaining comparable relative pin power factors even with gross dissimilar fissile content. 217 seed fuel rods and 90 fertile blanket rods are assumed to be contained in two coaxial hexagonal steel channels with inner/outer dimensions of 126/132 mm and 176/182 mm, respectively. The hexagonal fuel assembly pitch is 184 mm. Fig. 4a gives a schematic diagram of the proposed FTBR seed fuel assembly. The fertile rods in this assembly are assumed to be depleted UO_2 (DUO_2). They are assumed to have been irradiated for one fuel cycle in the same reactor at selected locations vacated for the purpose. The pin pitch of seed region is 8.3 mm and that of DUO_2 rods is 12 mm.

The second type of fuel assembly where the above irradiation takes place consists of 127 ThO_2 rods and 90 DUO_2 rods. It may be mentioned that in ATBR core design (Refs. [5,6]), the fertile blanket assemblies considered a single ring of thoria rods and a beryllium block was present inside it as a neutron scatterer. In FTBR design, it was initially thought to load just 90 DUO_2 rods in the two layers of fertile zone, leaving the space in the interior region filled with slow moving sodium coolant. However this can adversely affect the coolant outlet temperature and it is also not advisable to waste the large flux trap volume available inside each fertile assembly. Hence it was decided to load 127 ThO_2 rods of

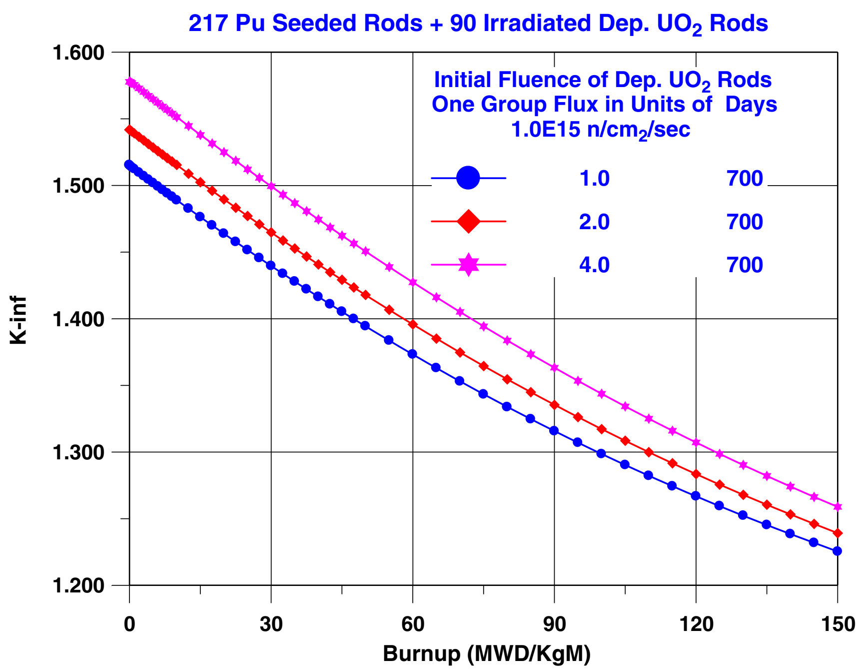

Fig. 7a. k_∞ of FTBR seed fuel cluster with 45% PuO_2 and different initial fluence on DUO_2 rods.

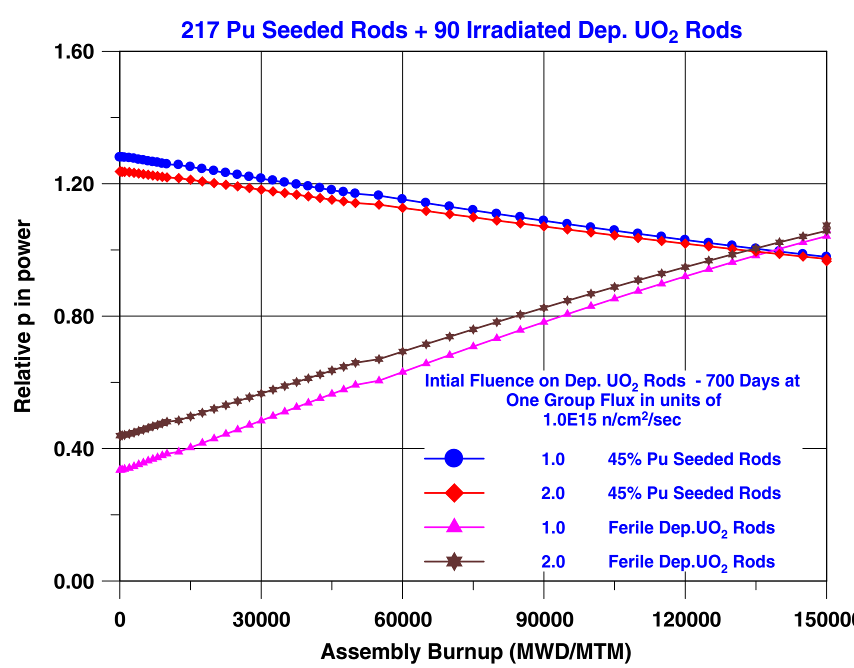

Fig. 7b. Relative P in power of FTBR seed fuel cluster with 45% PuO_2 and different initial fluence on DUO_2 rods.

same diameter as that of DUO_2 rods, but at a pitch of 10.85 mm. The schematic diagram of the ($127ThO_2 + 90DUO_2$) fuel assembly is shown in Fig. 4b.

The third type of fuel assemblies contain 127 fresh ThO_2 rods and 90 steel rods of outer dimension same as the DUO_2 rods. Structurally this assembly type would resemble type two assemblies shown in Fig. 4b.

Fig. 5 gives the optimized three batch fuel loading pattern. There are 360 assemblies of first type with seed and fertile rods, 120 second type of assemblies with only fertile rods and finally 120 assemblies of third type with ThO_2 rods. While the first two categories are appropriately arranged to obtain optimum power distribution, the third type of assemblies are invariably located at the peripheral layer of the core. There are 25 locations in the core which are designated for locating control rods to provide small reactivity maneuvers. At present they have been considered as fresh fertile assemblies of second type. The core thus consists of 625 assembly locations. The outermost two or more layers would consist of steel reflector assemblies and B_4C shielding assemblies.

In the second type of fuel assemblies DUO_2 rods are replaced every fuel cycle with fresh ones. ThO_2 rods in these assemblies may not all be fresh, but are assumed to follow an independent fuel management scheme. It is found to be necessary to allow them to reside in the core for two or three fuel cycles so that they can accumulate nearly the asymptotic fissile content prior to their discharge. The varying power fraction of these assemblies should be

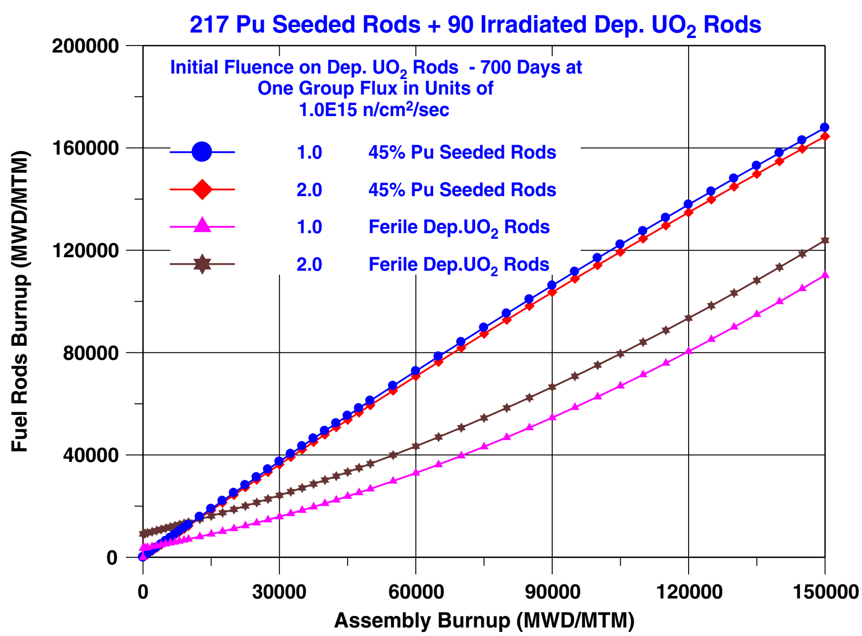

Fig. 7c. Fuel rod burnup of FTBR seed fuel cluster with 45% PuO_2 and different initial fluence on DUO_2 rods.

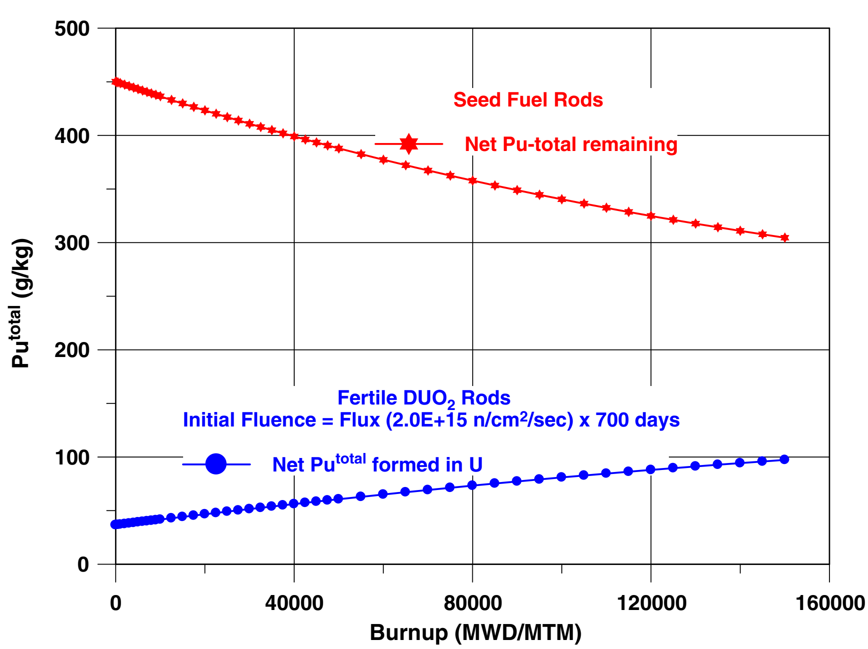

Fig. 7d. Net Pu^{total} in U in seed and fertile rods of seeded fuel assembly.

studied in detail and the coolant flow should cater to their maximum envisaged power. In the seeded fuel regions due to the use of two co-axial hexagonal channels it must be possible to send the coolant with higher velocity in the inner region while in the outer region the flow can be kept lower. A variable coolant flow during the course of fuel cycle had been suggested in some of the 'bull's-eye' concept of seed and blanket cores earlier in the late seventies [9].

3. Calculation model

The design described here is based essentially on neutronic behaviour of the Th, U and Pu fuels, as studied with the 172 group 'JEFF31GX' WIMS library obtained as part of WIMS Library Update Project from IAEA [10]. WIMS library has energy groups devised for thermal reactor design calculations. Further since there are no self shielding factors prescribed for diluent nuclides of Na coolant and SS clad materials, there may be some deficiency in performing the fast reactor calculations. The computations reported here have been performed with thermal reactor lattice and core computational code system PHANTOM [11] and TRISUL [12], which were used for the ATBR reactor design calculations. The CLUB module [13] of PHANTOM code system had been developed for annular ring type fuel clusters that are considered in PHWR. It was necessary to suitably alter the geometric modeling of the hexagonal fuel assembly of FTBR with fuel rods of two type of diameter and with

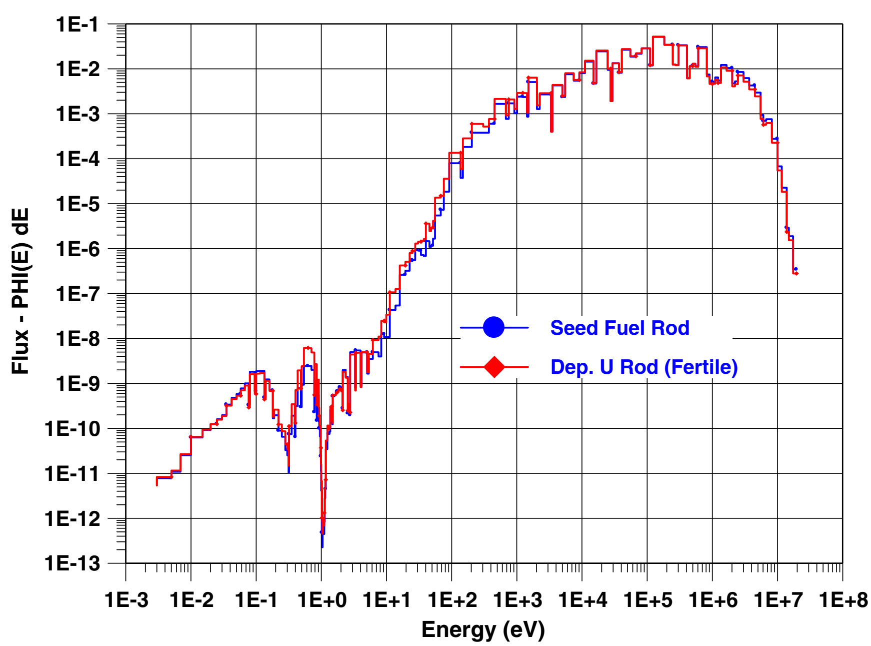

Fig. 8a. Typical neutron spectra in seed fuel assembly with 45% PuO_2.

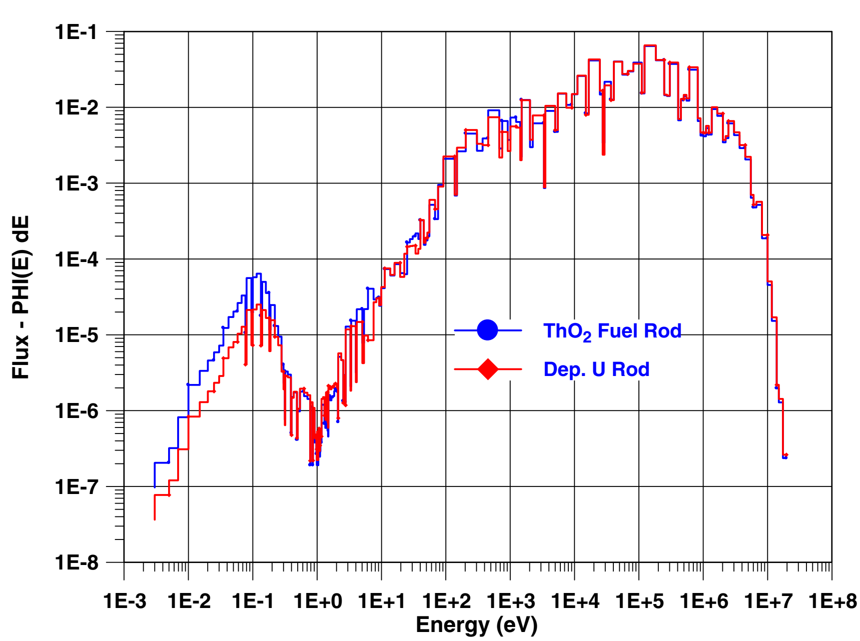

Fig. 8b. Typical neutron spectra in fertile blanket fuel assembly.

two different lattice pitches into equivalent ring type geometry. Monte Carlo validation in the fresh state of fuel assembly provided some degree of confidence in the calculation schemes adopted [14]. The fuel management schemes also presented some complications in tracking the flux history and fluence levels of fertile rods/assemblies which are irradiated in one location and subsequently shifted to various seed locations in the core in a three batch refueling scheme. Notwithstanding these disclaimers, it is believed that the ideas presented in this paper are tenable and it must be possible to evolve the correct design by specialists/experts from fast reactor discipline.

The PHANTOM simulations were done for the three types of fuel assemblies viz., third, second and first in that sequence. First, the 127 ThO_2 cluster and 90 steel rods were simulated. Irradiation was considered at average one group flux levels of (2, 4, 8) $\times 10^{14}$ n/cm^2/s (expected range of flux level at the core periphery). Secondly, the (127 irradiated ThO_2 rods + fresh 90 DUO_2 rods) assembly was simulated. The number density for fuel nuclides of ThO_2 rods were conservatively picked for the minimum flux level of 2×10^{14} n/cm^2/s at irradiation time of 500, 700 and 900 days. Finally the seeded fuel assembly was simulated. Flux values of (1, 2, 4) $\times 10^{15}$ n/cm^2/s and irradiation time of 500, 700 or 900 days were considered for the DUO_2 rods. The number densities of irradiated DUO_2 fuel nuclides were picked from the second set of calculations. Initially a range of Pu content values were studied for the seed fuel rods. After the initial trials between the lattice and core

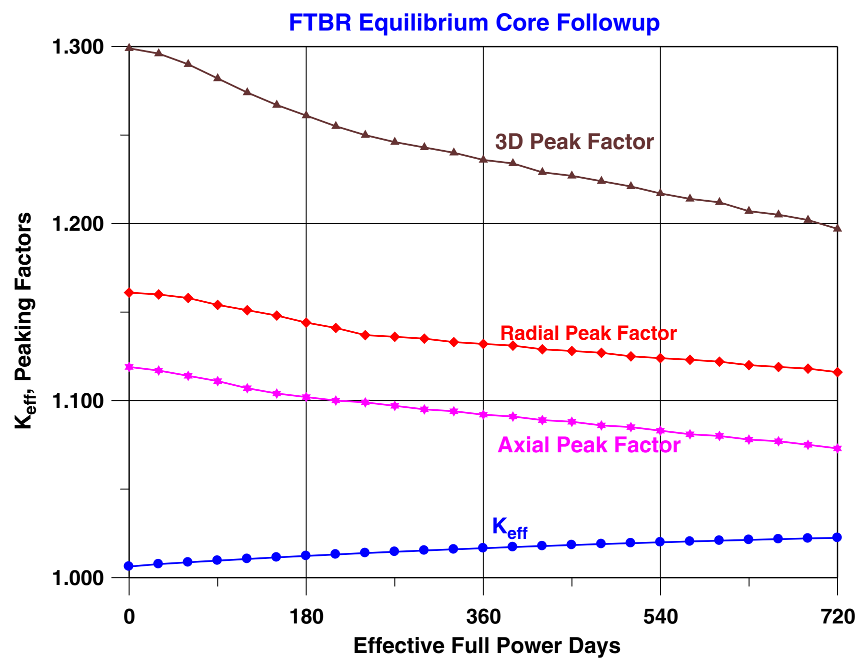

Fig. 9. FTBR equilibrium core – k_{eff} and peaking factors vs. cycle burnup.

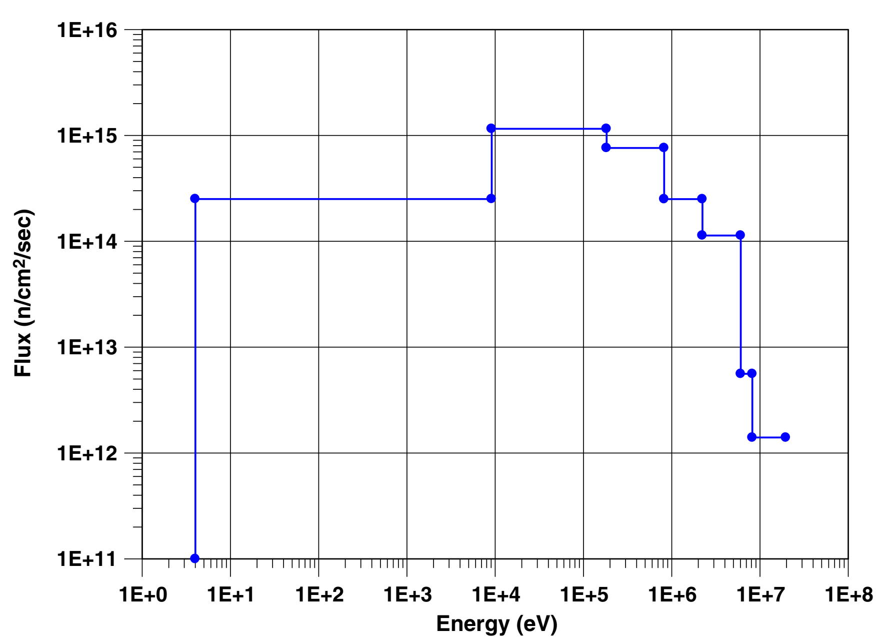

Fig. 10. FTBR equilibrium core – absolute neutron flux values in eight groups at peak location – core calculation by TRISUL code.

simulations, 45% PuO_2 content in the seed fuel rod was found to be adequate to ensure a core excess reactivity of about 10mk for a fuel cycle duration of 720 days. The results are presented in the ensuing section.

4. Results and discussion

4.1. Lattice results

Figs. 6a–6c give, respectively, the K_∞, burnup and U in Th or Pu in U accumulation as a function of fluence levels for the second assembly type ($127ThO_2 + 90\ DUO_2$). The K_∞ starts at $\sim$0.3 due to small initial content of ^{233}U in Th and ^{235}U in U. It reaches almost unity after about 2000 days for the highest flux level considered. However, as per the intended fuel management scheme, the DUO_2 rods will be taken out after its first fuel cycle and integrated with fresh 217 seed rods with 45% PuO_2 content. Hence the values plotted beyond one cycle duration of 720 days are of only academic significance.

Fig. 7a gives the K_∞ variation as a function of burnup for seeded fuel assembly with 45% PuO_2 in the seed fuel rods and the outer 90 DUO_2 rods with in situ bred Pu corresponding to different initial fluence levels. The K_∞ monotonically decreases from $\sim$1.55 to $\sim$1.25 after an assembly burnup of 150 MWD/kgM. Figs. 7b–7d give, respectively, the relative power share, burnup accumulation and the total Pu content in the two types of fuel rods of seeded

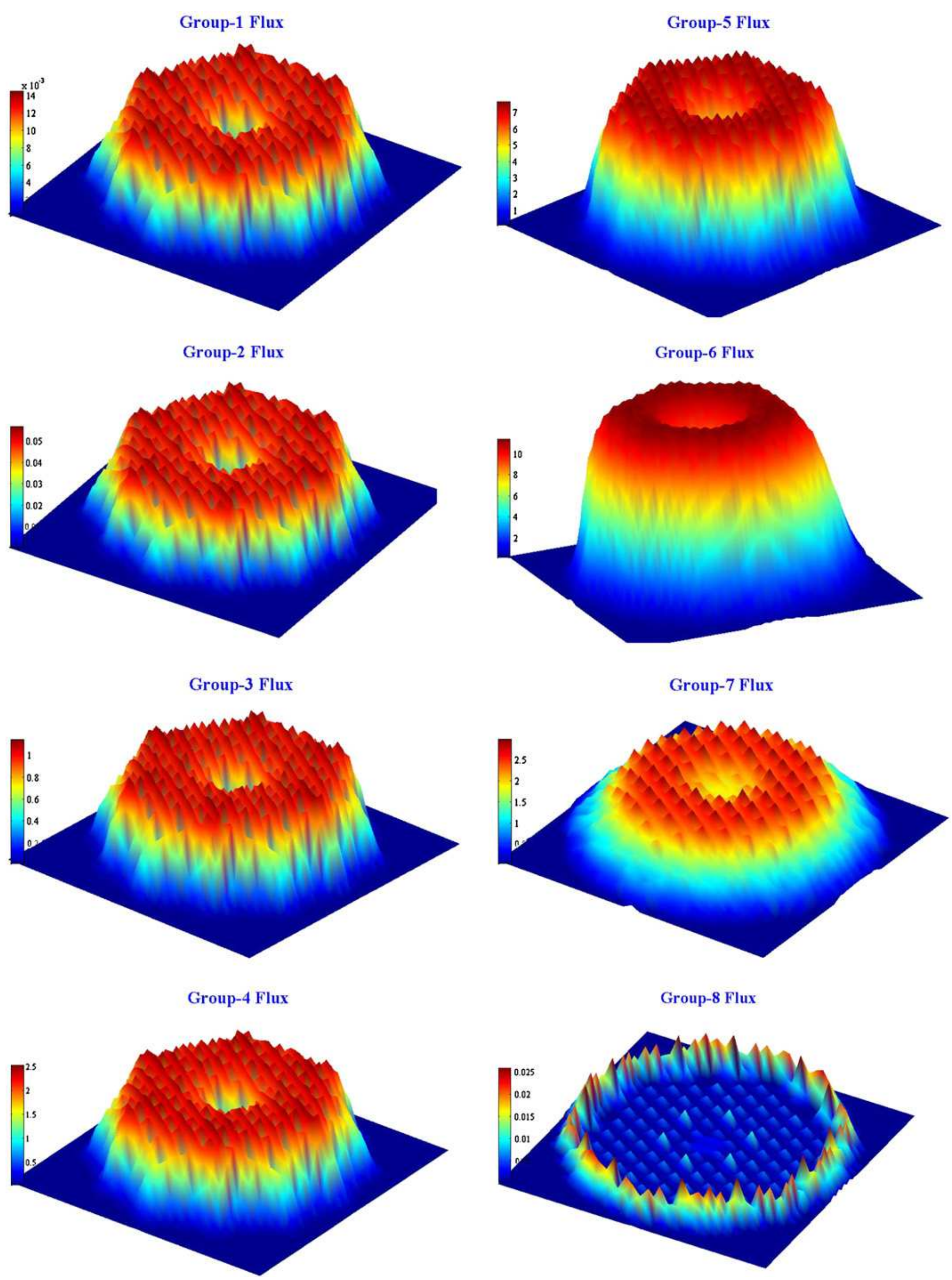

Fig. 11. FTBR equilibrium core – beginning of cycle 3D flux shapes for different energy groups at height = 172 cm.

assembly. It is seen that the power share between the inner seed zone and outer blanket zone in the seeded assemblies undergo significant change. The relative power of all seed rods is almost same and that of fertile rods is another single value. The seed fuel rods show a relative power factor of 1.25 at zero burnup while the outer fertile rods start with power factor of 0.4. It is seen that by using 1.54 times the diameter for fertile rods compared to seed fuel rods, the relative power share per blanket pin has become comparable to that of seed fuel rod despite having much lower seed content. The other factor which was useful in the thermal reactor design ATBR and not present here is the absence of strong flux gradients across the fuel assembly. The fast flux is normally monotonously flat and is not as malleable as the thermal neutron flux. Thus the fissile formation rates in fertile rods are essentially helped by large rod volume and a slight drift or softening towards the resonance region. As seen from Fig. 7b, the fertile rod power share rises progressively and if the assembly can attain a burnup of 140 MWD/kg, there could even be a cross over of the two power factors of seed and fertile zones. As may be seen from Fig. 7c, near a discharge burnup of 100 MWD/kg the seed fuel rods would achieve a burnup of

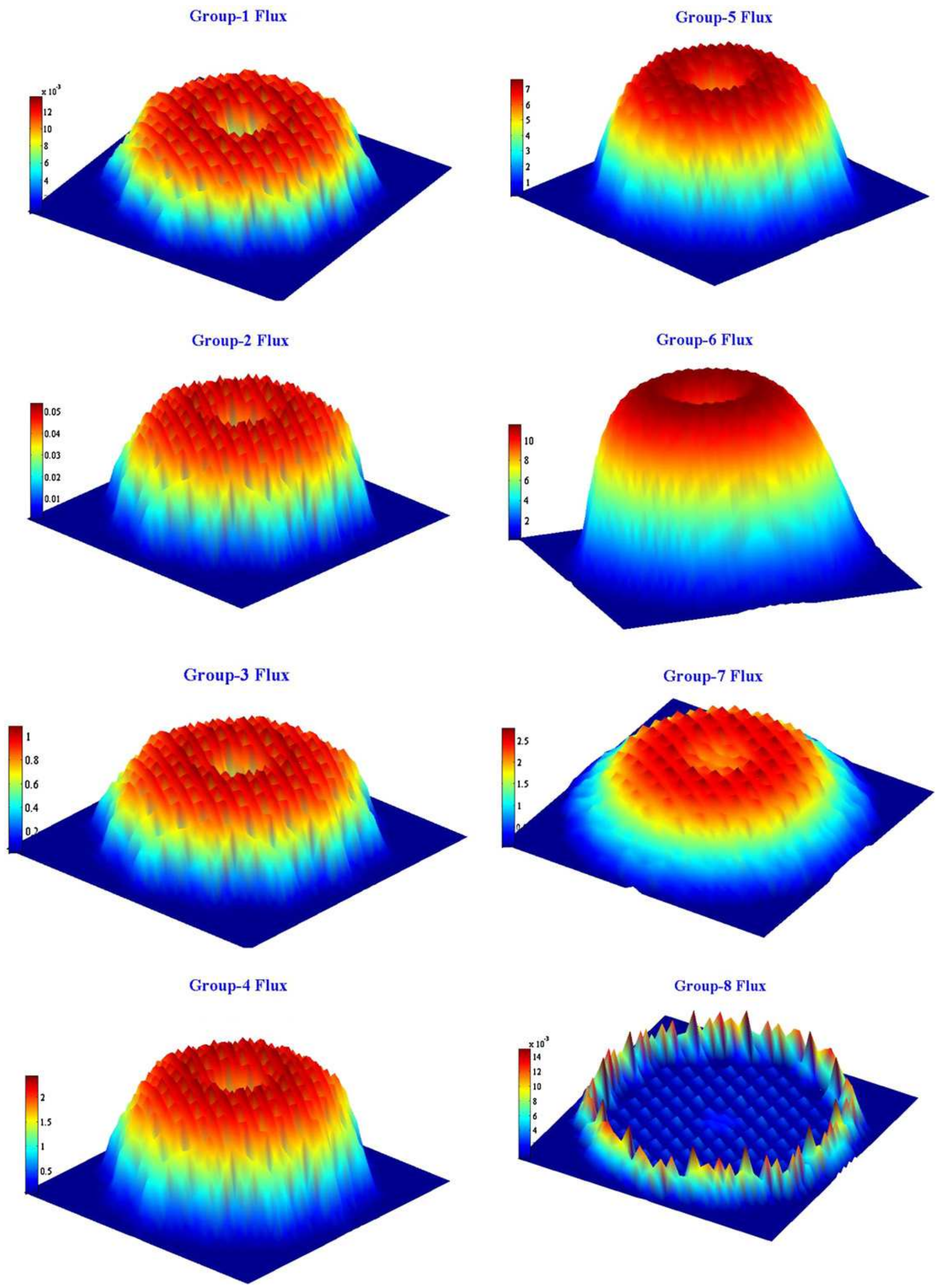

Fig. 12. FTBR equilibrium core – end of cycle 3D flux shapes for different energy groups at height = 172 cm.

117 MWD/kg while the fertile rods achieve a burnup of 75 MWD/kg. By judicious fuel management it must be possible to bring down even this difference. It is seen from Fig. 7d that the total Pu decreases from 450 g/kg to 340 g/kg at 100 MWD/KgM, while the net Pu in the DUO_2 rods (with initial fluence of 2×10^{15} n/cm^2/s $\times$ 700 days) starts from 36.5 g/kg and reaches 81 g/kg at the same assembly burnup.

Typical 172 group neutron spectra in the seed and fertile fuel rods are shown in Figs. 8a and 8b. The peak is seen near 100 keV. The resonance region flux (9 keV to 4 eV) is slightly higher in the fertile rods.

4.2. Core results

The core calculations were done using the TRISUL code. Eight energy groups were considered. The last two groups span the resonance and thermal energy range. The equilibrium core burnup distribution was iteratively converged for the optimized refueling scheme. It was important to prescribe appropriate flux and starting fluence values for the fertile assemblies which are linked by fuel management scheme. Fig. 9 shows the variation of core k_{eff}, radial, axial and global 3 D peaking factors as a function of effective full power days. It was seen that k_{eff} increases monotonically from 1.006 to 1.022. The peaking factors decrease monotonically. The radial, axial and 3D peaking factors decrease from (1.16, 1.12 and 1.3) to (1.12, 1.07 and 1.2), respectively. Fig. 10 gives the absolute flux values in eight energy groups at the peak flux location. This was seen to be nearly same at the beginning and end of equilibrium fuel cycle. Figs. 11 and 12 illustrates the flux distribution in eight energy groups at the beginning and end of equilibrium fuel cycle. While the fluxes in fast groups show an overall flat shape except for a central cavity where there is heavy loading of fertile assemblies, the flux in 7th group (resonance region) shows small local spikes near fertile assemblies. Thermal flux is literally absent in core region and is seen only in the outer reflector layer.

There is a tendency for the flux and power level to get flattened with burnup, when the power share of fertile rods progressively improves. Thus if the thermal hydraulic design is suitably designed, (read as over-designed), it must be possible to extract more power at the end of every fuel cycle. This will increase the flux le-

Table 2
FTBR equilibrium core – material inventory (BOC)

Fuel assembly type	Number of assemblies	Th (T)	U-tot in Th (T)	U-fis in Th (T)	U (T)	Pu-Tot in U (T)	Pu-fis in U (T)
Fuel mass at beginning of equilibrium cycle							
Fertile blanket assemblies							
Fresh ThO_2	120	15.396	–	–	–	–	–
Irr. ThO_2 + fresh DUO_2	145	18.098	0.449	0.439	14.690	–	–
Seed fuel assemblies							
Fresh 45% Pu seeded DUO_2	120	–	–	–	4.188	3.427	2.553
One cycle irr. DUO_2	120	–	–	–	7.316	0.142	0.132
Sum of seeded FA	120	–	–	–	11.504	3.569	2.685
Seeded FA I cycle	120	–	–	–	11.175	3.435	2.497
Seeded FA II cycle	120	–	–	–	10.587	3.246	2.401
Axial blanket							
Fresh	120	–	–	–	9.176	0.117	0.112
I cycle	120	–	–	–	9.051	0.226	0.217
II cycle	120	–	–	–	8.939	0.307	0.289
Total mass	625	33.494	0.449	0.439	75.06	10.96	8.263

Total mass at BOEC = 108.554 T (fertile) + 0.449 U in Th + 10.96 T (Pu seed in U) = 119.963 T.

Table 3
FTBR equilibrium core – material inventory – (EOC)

Fuel assembly type	Number of assemblies	Th (T)	U-tot in Th (T)	U-fis in Th (T)	U (T)	Pu-Tot in U (T)	Pu-fis in U (T)
Fuel mass at end of equilibrium cycle							
Fertile blanket assemblies							
One cycle ThO_2	120	15.378	0.019	0.019	–	–	–
One cycle ThO_2 + fresh DUO_2	145	17.564	0.814	0.787	14.248	0.371	0.356
Seed fuel assemblies							
Seeded FA I cycle	120	–	–	–	11.212	3.453	2.522
Seeded FA II cycle	120	–	–	–	10.760	3.299	2.286
Seeded FA III cycle	120	–	–	–	10.587	3.246	2.201
Axial blanket							
I cycle	120	–	–	–	9.041	0.227	0.217
II cycle	120	–	–	–	8.822	0.389	0.362
III cycle	120	–	–	–	8.717	0.454	0.415
Total mass	625	32.942	0.833	0.806	73.39	11.44	8.359

Total mass at EOEC = 106.332 T (Fertile) + 0.833 T U in Th + 11.44 T (Pu seed in U) = 118.605 T.
Pu seed input per cycle (feed) = 3.427 T.
Pu output in discharged seeded zone = 3.246 T.
Pu output in axial blanket = 0.454 T.
U output in thoria rods (1/3rd discharged) = 0.833 T/3 = 0.278 T (at least).
Total output = 3.978 T.
Breeding ratio = 1.16 (conservative value).
Only 1/3rd of the batch of thoria rods with maximum U to be discharged in each cycle. ThO_2 rods can be irradiated for 2–4 cycles.

vel, breeding rates in fertile mass and also help in higher fissile inventory for subsequent fuel cycle. It is therefore suggested that one may adopt a strategy of up rating the reactor power level at the end of every fuel cycle. The limitation would essentially come from the maximum burnup allowable for the seed/blanket fuel rods of different diameter. In addition, the fertile rods reside one more fuel cycle than the seed fuel rods and hence their clad design should allow for higher neutron fluence.

4.3. Material balance

The fuel inventory at the beginning and end of equilibrium core are presented in Tables 2 and 3. The Pu input fed externally in the fresh batch of seed fuel rods is 3.427 T every cycle. Total Pu in three batch of seed fuel is 10.96 T, inclusive of Pu in uranium of DUO_2 rods in seed assemblies and axial blanket regions. This is supplemented by in situ bred 0.449 T of U in thorium. At the end of equilibrium cycle the Pu content in seeded assemblies decreases to 3.247 T. The overall Pu inventory increases to 11.44 T and the U in thorium increases up to 0.833 T. Since the beginning of equilibrium fuel cycle requires about 0.45 T of U in thorium, for ensuring sufficient excess reactivity, one can discharge only a fraction of the thoria assemblies in each cycle. Thus the thoria assemblies would follow an independent fuel management scheme. It is seen that there is a net gain in the fissile inventory during the fuel cycle which is also reflected in continuous increase of k_{eff} during the cycle. It is judged that the breeding ratio can be about 1.16 to 1.2. It should also be pointed out that one component of fissile conversion in the blanket regions, their longer residence time and fuel movement schemes allow relatively large power share of the blanket regions and their discharge burnup is ~70% of the seeded rods with 45% initial PUO_2.

The Pu accumulation in DUO_2 rods and axial blanket regions should be adequate not only to replenish Pu loss in seed rods, but also for eventual growth as well. The additional ^{233}U bred from thoria rods can be utilized in a thermal reactor variant like ATBR. Since this uranium would have ~1000 ppm of ^{232}U it would require perfection of technologies of remote handling for reprocessing and refabrication in a massive scale. It may be mentioned that in a previous study [15] we had shown that one would require ~800 kg of ^{233}U for 17 months operation of a ^{233}U seeded ATBR-600 MWe design with initial fissile inventory of about two tonnes.

5. Conclusions

The possibility of implementing the ideas of internal fissile breeding zones in a fast reactor spectrum as was suggested in our earlier work [7] on ATBR has been studied in detail. The new reactor concept has been given the name 'Fast Thorium Breeder Reactor' (FTBR). This name has been retained though the design described in this paper suggests not only breeding of ^{233}U from thorium but also Pu from DUO_2 in the same reactor. Hence the acronym can also be expanded as 'Fast Twin Breeder Reactor'. The traditional blanket regions surrounding the core are brought in and accommodated in the interior region so that uniform high flux level is seen by both seed and fertile rods. The flux level is nearly halved but prevalent over the entire core (internal blankets). Hence the seed depletion rate is slower. To account for the reactivity load of huge blanket regions, 45% PuO_2 is needed in seed fuel rods. This may be justifiable in view of higher returns from the blanket zone which not only attain nearly 75% of the seed zone burnup but also contain nearly 80% of the maximum possible asymptotic fissile content at the time of discharge due to long residence time of ~3000 days. The higher fissile inventory may be justifiable if one considers it as the fraction of the integral hold up of fissile material in inpile as well as out of pile inventory of a closed fuel cycle.

In the present proposal, it is necessary to irradiate fertile DUO_2 rods separately for one fuel cycle and then integrate them with fresh fuel rods. The contrast in the power share of seed and fertile rods required that they are distributed separately in two co-axial hexagonal channels. Proper design of the orifice and possible online regulations of coolant flow during the fuel cycle may be needed. Though this task looks formidable, in our view, these thermal hydraulic design problems are far less challenging and far more promising than the similar challenges faced in the ADS type applications. Alternately one can consider separate fuel assemblies of seed and blanket type with same assembly pitch and contained in one steel channel. This would alleviate the problem of coolant flow design and also cut down parasitic capture in steel. This would be studied in our next paper. We would also explore the characteristics of different types of fuel and coolant combinations, retaining the idea of internal fissile breeding zones.

Acknowledgement

The first author would like to acknowledge with gratitude Dr. P. Mohanakrishnan, Head, Reactor Physics Division, IGCAR, Kalpakkam, for many enlightening discussions on fast reactor physics.

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