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Advanced fuel cycles for use in PHWRs

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A B S T R A C T

Pressurized heavy water reactors (PHWRs) were originally designed for employing once through fuel cycles with natural uranium. The excellent neutron economy and on-line fueling due to limited excess reactivity are important characteristics of these reactors. However, PHWRs have the main drawback of low burn-up, approximately 7500 MWd/T, due to the use of natural uranium. Use of neutron absorbers for control and power flattening further deteriorates the burn-up. All these aspects, specific to PHWRs, also lead to management of large quantities of: (i) initial fuel (ii) irradiated fuel, and (iii) radioactive wastes. Some of these drawbacks can be alleviated with high burn-up fuel, which also improves fuel utilization. Slightly enriched uranium and plutonium have been under consideration for this purpose. In situ production of U^{233}, by using thorium along with appropriate fissile feed, is one possibility. Alternatively, U^{233} can be generated externally in fast breeder reactors. It has been recognized that, when used along with thorium, PHWRs can also serve as efficient burners of excess plutonium accumulated over the years. Fuel cycles have been designed so as to completely reverse the isotopic composition (fissile to fertile ratio) which exists at the beginning of a cycle. These cycles also envisage producing proliferation resistant fuels containing high gamma-active decay products. Most of the reactor physics aspects of the various fuel cycles can be analyzed using simple methods of neutron physics and fuel burn-up. Multi-group techniques and explicit representations of the PHWR cluster geometry are essential. However, core physics and fuel management calculations can be simplified at an exploratory stage. Nevertheless, it is necessary to make sure, using core analyses, that the new fuel cycles do satisfy all the constraints of flux peaking, controllability, coolant void reactivity, etc. The main aim in this paper is to provide a comparative evaluation of the various advanced fuel cycles that are feasible in PHWRs.

1. Introduction

Nuclear power is once again in great demand due to the growing energy needs of the world population, particularly in the developing countries. The limited availability and environmental issues associated with fossil fuels are other contributing factors [1].

Two types of thermal reactor systems, namely, the light water reactors (LWRs), which include both pressurized water reactors (PWRs) and boiling water reactors (BWRs), and PHWRs, have been developed well and commercially proven. While PWRs requires about 3% U^{235} enrichment, PHWRs use natural uranium together with heavy water as moderator. Enhanced neutron economy of PHWRs, because of the negligible neutron absorption in heavy water, is one of its important characteristics. Even so, natural uranium provides only a small excess reactivity and hence low burn-up, thereby leading to on-power fueling. These aspects, specific to PHWRs, also require large quantities of initial fuel. As a result, it also produces large quantities of irradiated fuel and radioactive wastes. Use of neutron absorbers for control and power flattening further deteriorates the fuel burn-up. On-line fueling provides several possibilities in introducing new fuel cycles, even though it generates a few operational and engineering problems associated with the fueling machine.

The above considerations lead to a genuine interest in extending burn-up in PHWRs, which can be done by increasing the fissile content in the fuel. Slightly enriched uranium and plutonium have been under consideration for this purpose. Several other possibilities emerge with the Th–U^{233} cycles. In situ production of U^{233}, by using thorium along with appropriate fissile feed, is one such possibility. Alternatively, U^{233} can be generated externally in fast breeder reactors. Due to the good neutron economy, PHWRs also show the possibility of self-sustained Th–U^{233} cycles, however with lower burn-up.

Lastly, there is the possibility of using spent fuel from PWRs, containing nearly 1.56% fissile material, in PHWRs, thereby reducing the waste burden. There is no reprocessing in the DUPIC fuel, except removing the volatile fission products [2]. However, in the TANDEM cycle, uranium and plutonium together are separated and blended with Nat-U [3].

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2. Properties of nuclear fuels

There are only three fissile nuclides, namely, U^{235}, U^{233} and Pu^{239}, of which U^{235} alone is naturally occurring, while the other two have to be produced via artificial transmutation in reactors. Their nuclear properties determine all the features of various fuel cycles. While the thermal neutron (2200 m/s) capture cross-section of U^{235}, U^{233} and Pu^{239}, respectively, are 101, 46 and 271 barns, their fission cross-sections are 577, 525 and 742 barns. The number of neutrons emitted per thermal neutron absorbed, denoted as η, are 2.08, 2.29 and 2.12, respectively. However, in fast neutron spectrum reactors, η have values 1.93, 2.31 and 2.49. This indicates that while Pu^{239} is a better fuel in fast reactors, from the point of breeding fissile material, U^{233} is less sensitive to neutron spectrum. Further, it is also clear that U^{233} and Pu^{239} can provide better neutron inventory in thermal reactors.

The two important fertile nuclides, namely Th^{232} and U^{238}, have thermal capture cross-sections 7.4 and 2.73 barns, respectively. Therefore, thorium will invariably require larger fissile inventory for criticality in comparison to uranium. But then, thorium will also produce larger amount of U^{233} as the reactor continues to operate.

3. Calculation model

For developing a quantitative comparison of different fuel cycles, we have calculated the lattice multiplication factor (K_∞) and isotopic compositions of various nuclides as functions of burn-up. These lattice cell calculations were carried out by a 'state of the art' computer code CLUB [4], which uses the 69 group WIMS cross-section library [5]. This code solves the multi-group integral transport equation by a combination of interface current and collision probability methods. It also solves the burn-up equations for fuel and fission product for a given specific power, thereby providing isotopic compositions and cell K_∞ (B) as a function of burn-up (B).

To obtain the discharge burn-up, simple recipe is used [6]. Due to the on-power refueling, PHWR core carries a nearly continuous distribution of fuel burn-ups from fresh to discharge value. This spatial distribution of the core burn-up may be represented in terms of an average $\bar{K}_\infty$, defined as

$$\bar{K}_\infty = \frac{1}{B_d}\int_0^{B_d} K_\infty(B)\mathrm{d}B \tag{1}$$

The discharge burn-up, B_d is now obtained by requiring $\bar{K}_\infty = 1.045$, as the typical value of leakage in a PHWR is ~45 mk. This approximate method is used to compare the fuel cycles.

4. Different fuel cycles

The standard 37 fuel pin PHWR cluster is considered, which is schematically shown in Fig. 1. Typical dimensions of the fuel pin are also given. In this study we shall mainly consider cycles corresponding to burn-up up to about 25 000 MWd/T so that the peak power ratings are not more than 120% of the standard PHWR values. This is so because Canadian studies have indicated that, while the advanced 43 pin CANFLEX fuel does not change the reactivity very much, it reduces the peak power ratings by about 20% compared to the 37 rod cluster [6,7].

One of the parameters which can be used to compare different cycles is the advantage factor X_{ad} defined as the total energy produced in units of MWd/gm of fissile material that is actually consumed in the reactor [8]

$$X_{ad} = 10^{-4}\frac{B_d}{e(1-F_{ir})} \tag{2}$$

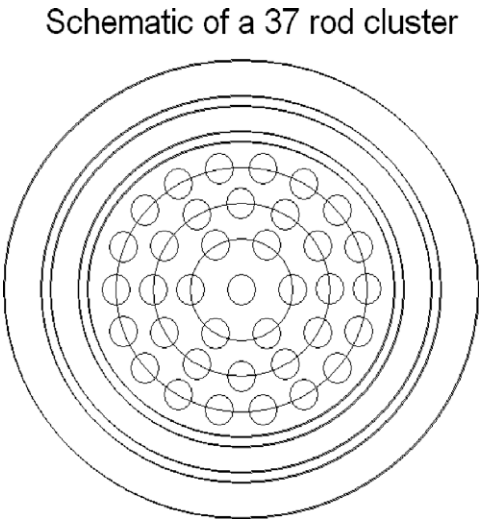

Fig. 1. Schematics of the 37 pin PHWR fuel cluster.

where 'e' is the percentage fissile enrichment, B_d is discharge burn-up in MWd/T and F_{ir}, is defined as the ratio of all fissile inventory present in the discharged fuel to that in the initial fuel. The parameter X_{ad} should be used only for cycles with $F_{ir} < 1$. Similarly, another parameter of interest is the energy utilization factor X_{ut} defined as

$$X_{ut} = 10^{-4}\frac{B_d}{e} \tag{3}$$

This defines the total energy produced in units of MWd/gm of the initial fissile inventory. A higher value of X_{ad} will be advantageous in closed fuel cycles as F_{ir} will be close to unity. For once through cycle it will be more appropriate to have maximum value of X_{ut}.

The different fuel cycles investigated can be grouped into three main categories: (i) slightly enriched uranium (SEU) cycles, (ii) plutonium cycles and (iii) thorium cycles. In all these cases, the fuel is considered in oxide form or as mixtures of different oxides.

5. Slightly enriched uranium

First of all, it is verified that the model outlined earlier generate the well known results of natural uranium (Nat-U) once through cycle (OTC). $K_\infty(B)$, and isotopes composition (g/kg) vs. burn-up are shown in Fig. 2(a) and (b). The recipe discussed earlier gives a discharge of 7210 MWd/T. It is found that 168 T of Nat-U will be needed per GWe-Y. It is also seen that 3.7 kg of plutonium are produced per ton of uranium, and the plutonium fissile ($Pu^{239} + Pu^{241}$) composition is 72%. These figures agree well with published results [9].

The fissile content of 0.71% U^{235} in Nat-U is not its most optimum value from the point of view of burn-up in OTC [9]. This is shown in Fig. 3(a) where the energy utilization factor X_{ut} has near optimum value of 2 MWd/g around 1.5% enrichment. In the same figure the variation of advantage factor X_{ad} is also shown. This as well as the burnt fissile inventory decreases with enrichment.

Fig. 3(b) shows that burn-up reaches up to 31 500 MWd/T at 1.5%. The corresponding requirements of Nat-U and enriched

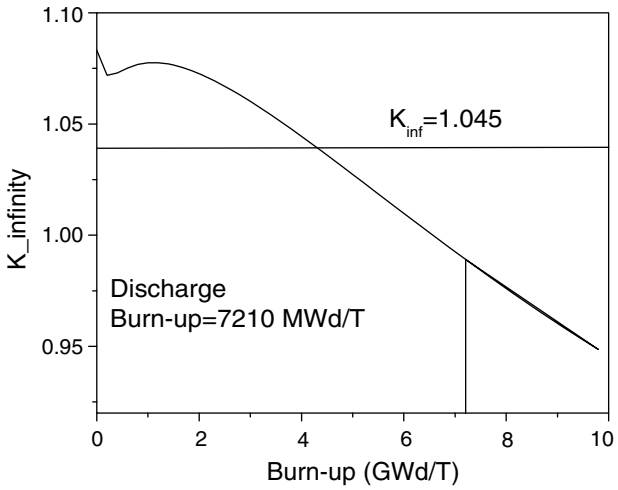

Fig. 2a. K_∞ vs. burn-up (Nat-U).

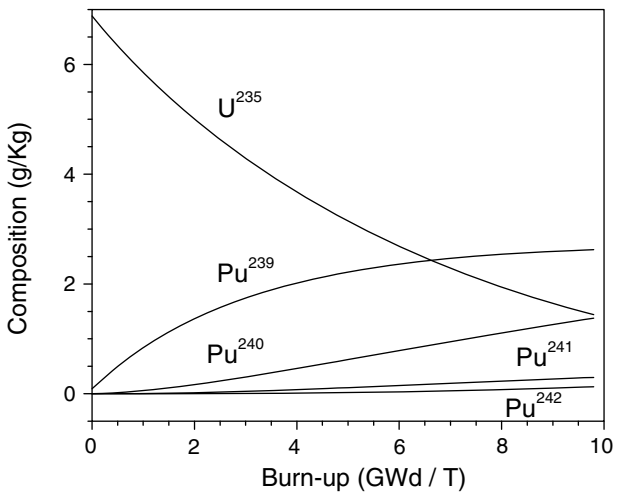

Fig. 2b. Compositions vs. burn-up (Nat-U).

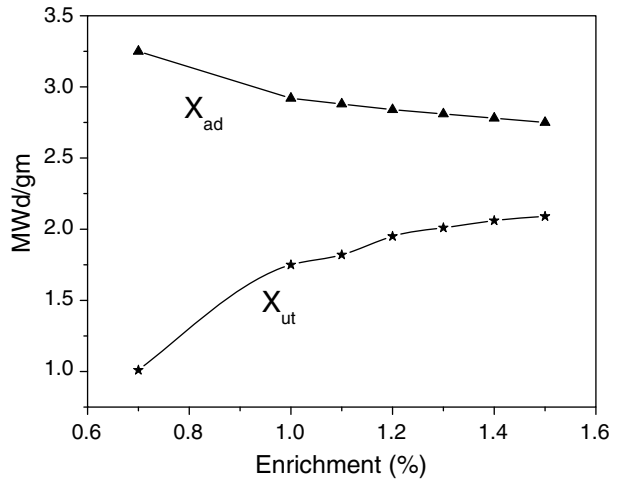

Fig. 3a. X_{ut} and X_{ad} vs. enrichment.

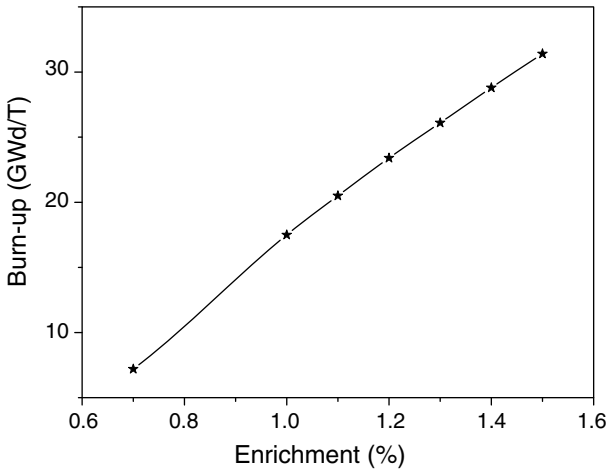

Fig. 3b. Burn-up vs. enrichment.

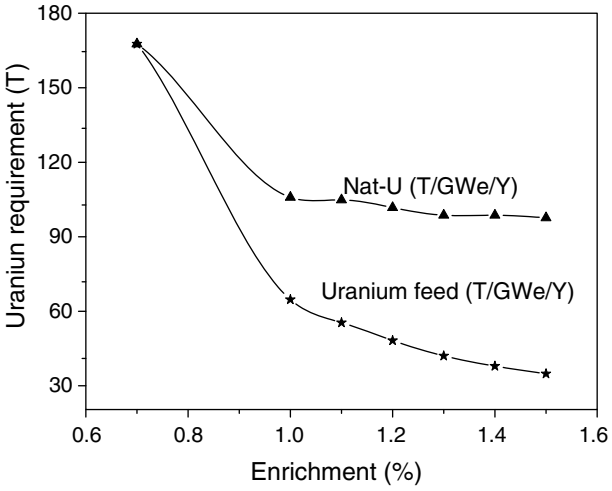

Fig. 3c. Uranium needed vs. enrichment.

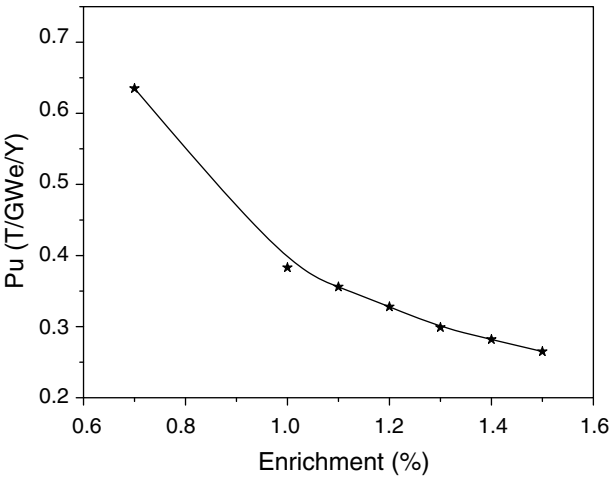

Fig. 3d. Pu production vs. enrichment.

uranium are shown in Fig. 3(c). The Nat-U requirement, calculated assuming a tail enrichment of 0.2%, becomes practically constant around 1.2%. The annual production of plutonium per GWe is reduced by ~50%, in comparison to Nat-U, as shown in Fig. 3(d). This is because more neutrons are absorbed in U^{235} than in U^{238} at higher enrichments. The ratio F_{ir} defined earlier, shown in Fig. 3(e), also decreases with enrichment.

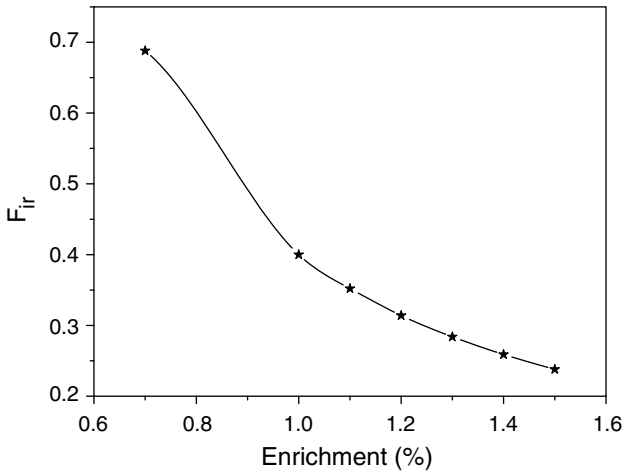

Fig. 3e. F_{ir} vs. enrichment.

6. Plutonium recycling

Plutonium recycling has been considered in two ways: First with Nat-U [10] and second with depleted uranium (Dep-U), which is left in the OTCs of PHWRs. Though these are possible cycles, it is noted that plutonium is better suited to fast reactor spectrum, and large scale use in PHWRs will not lead to significant growth of nuclear energy.

6.1. Pu–Nat-U cycle

In this case 0.45% of plutonium in Nat-U is considered in all the 37 pins of the cluster. Discharge burn-up that can be obtained is, approximately, 17 700 MWd/T corresponding to Nat-U consumption of 70 T/GWe-Y. This amounts to almost 60% saving of Nat-U resources. However, about 310 kg/GWe of plutonium will be consumed annually. The energy utilization factors, X_{ad} and X_{ut} are 2.88 and 1.74, respectively, which are nearly the same as in SEU case with 1.15% enrichment. As seen from Fig. 4, the discharge fuel consists of 6.7 kg of plutonium per ton of uranium with fissile (Pu^{239} + Pu^{241}) to fertile (Pu^{240} + Pu^{242}) ratio ~50:50.

6.2. Pu–Dep-U cycle

Next, 0.9% of plutonium in Dep-U with 0.3% of U^{235} is considered in all the pins of the cluster. This provides a discharge burn-up of 16 200 MWd/T, which is slightly lower than the earlier case because the total fissile content is less. Due to the higher plutonium fraction, its annual consumption increases to, approximately, 675 kg/GWe. At the end of the cycle, 7.98 kg per ton of plutonium will be left behind with fissile to fertile ratio ~47:53. The energy utilization factors, X_{ad} and X_{ut}, are 2.9 and 1.67, respectively.

Both the plutonium recycling cycles can be repeated with an additional stage of reprocessing. However, the discharge burn-up will be slightly lower ~14 000 MWd/T because of the lower fissile content. Further repetition of the cycle may not be worthwhile due to deterioration of plutonium.

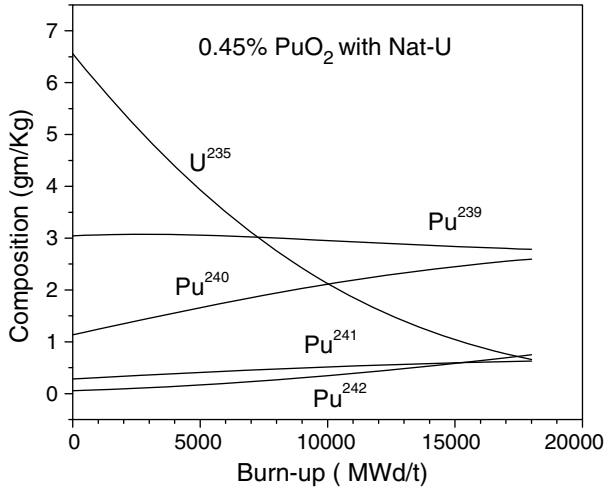

Fig. 4. Composition vs. burn-up.

7. Thorium utilization

Thorium, being a fertile material, has to be used with any one of fissile materials, U^{235}, Pu^{239} or U^{233}. Thorium based fuel cycles have intrinsic proliferation resistance due to formation of U^{232} via (n, 2n) reactions with Th^{232}, Pa^{233} and U^{233}. Half life of U^{232} is only 73.6 years. The daughter products have short half lives and two of these, Bi^{212} and Tl^{208}, emit strong gamma rays. In Th–U^{233} cycle, much lesser quantity of plutonium and long lived minor actinides (Np, Am and Cm) are formed as compared to U–Pu fuel cycle. This minimizes the radio toxicity associated in the spent fuel. However, in the backend of Th–U^{233} fuel cycles, there are radio nuclides such as Pa^{231}, Th^{229} and U^{230}, which may have long term radiological impact.

It is noted that thorium oxide has 30% higher thermal conductivity than uranium oxide and its melting temperature is also higher by 340 °C. Thus the fuel operating temperatures will be lower. In addition the chemical stability of thorium oxide helps in normal operations.

7.1. Th–U^{235} cycle

In the Th–U^{235} cycles, two possibilities are considered: (i) uniform enrichment of U^{235} in thorium oxide in all pins of the cluster and (ii) Thorium oxide in the 2 inner rings (7 pins) and enriched U^{235} in 2 outer rings (30 pins). In the first case the fuel is a mixture of ThO_2 and 10% enriched UO_2.

7.1.1. Uniform U^{235} in all rings

Table 1 shows the different features of this cycle as UO_2 fraction is varied. An appropriate value for UO_2 without significant peaking problems in this cycle will be 30% corresponding to U^{235} fraction of 3%, around which the advantage factor X_{ad} is 3.4. Annual enriched uranium needed is somewhat high, approximately, 6 T/GWe, though 205 kg of U^{233} and 23 kg of U^{235} are left in the discharge fuel.

In Fig. 5 the energy utilization factor X_{ut} and advantage factor X_{ad} of this cycle are compared with those of the SEU cycle. X_{ad} is higher in the thorium cycle and consequently a higher amount of fissile material will be left in the discharge fuel. However, energy produced per gram of initial fissile, X_{ut}, is smaller at lower enrichments. Thus it is favorable to consider higher enrichments within that allowable from power peaking considerations.

7.1.2. U^{235} in outer 2 rings

This case corresponds to 1.3% enriched U^{235} in the 2 outer rings of the cluster and thorium in the 2 inner rings [6]. The build up of U^{233} in each cycle is shown in Fig. 6. Other results provided in Table 2 show that burn-up progressively increases from 13 800 to 22 500 MWd/T in five cycles. Uranium consumption and its average cumulative values are also given in the table. Average uranium consumption at the end of five cycles is ~112 T/GWe-Y which is 35% lower than that in the Nat-U cycle. This cycle can be compared with the case of 1.2% SEU in all the pins of the cluster, where the

Table 1
Uniformly enriched U^{235} in all rings

U^{235} fraction	Burn-up MWd/T	Total heavy metal T/GWe-Y	UO_2 needed T/GWe-Y	U^{233} in discharged fuel kg/GWe-Y	U^{235} in discharged fuel kg/GWe-Y	Advantage factor X_{ad} MWd/gm
2%	20100	60.5	12.1	577	260	3.8
3%	60080	20.2	6.1	205	23	3.4
4%	89150	13.6	5.4	118	10	3.1

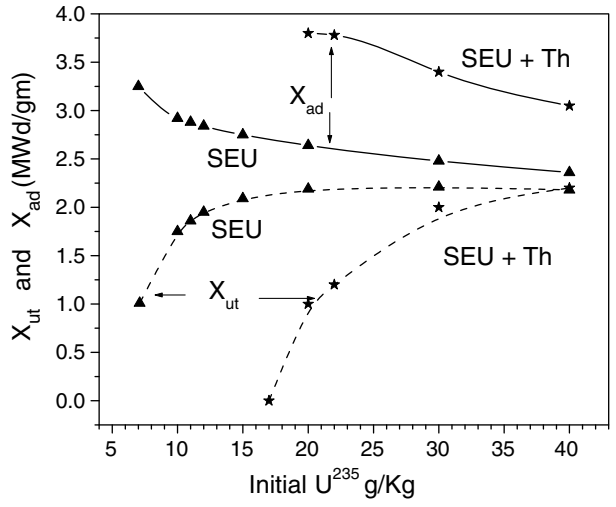

Fig. 5. Energy utilization in Th–U^{235} cycle.

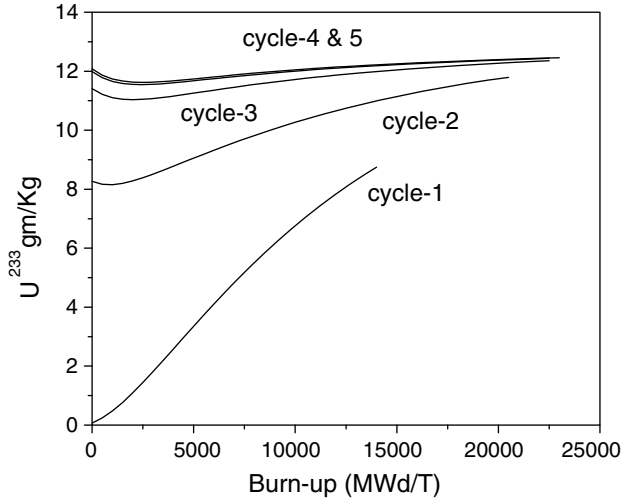

Fig. 6. Th–U^{235} (only in 2 outer rings) cycle.

Table 2
U^{235} in outer 2 rings

Cycle	Burn-up MWd/T	U consumption T/GWe-Y	Cumulative U consumption T/GWe-Y
1	13800	157.2	157.2
2	20000	108.5	132.8
3	22000	98.6	121.5
4	22500	96.8	115.3
5	22500	96.4	111.5

annual uranium consumption is only 105 T/GWe with 23500 MWd/T burn-up. It is important to note that there is no additional saving in Nat-U with the use of thorium in 2 inner rings. However, U^{233} fraction remains practically constant at 12.5 g/kg during the fifth cycle, thereby leaving 450 kg/GWe in the spent fuel. After each cycle, the Pa^{233} will also decay to U^{233} with a half life of 27 days and will add on to the fissile inventory.

Replacement of uranium pins in the outer 2 rings after each cycle is a difficult task though such demountable fuel bundles have been considered by AECL, Canada [6].

7.2. Th–Pu cycle

In the Th–Pu cycle also two possibilities are compared: (i) uniform fraction of Pu in thorium in all pins in the cluster [11] and (ii) pure thorium in the inner 2 rings and plutonium mixed with thorium in outer 2 rings. Both cases are useful to burn excess weapon grade or accumulated reactor grade plutonium [12]. Just as in the Th–U^{235} cycle, the outer 2 rings in second case will need to be replaced at the end of every cycle.

7.2.1. Uniform Pu in all rings

Table 3 shows the main features of this cycle as the Pu fraction is increased from 2 to 4%. The Pu-fissile content is 75% corresponding to the PHWR discharge fuel. Plutonium fraction without significant peaking problems is found to be around 3% from pin power distribution in the cluster. The advantage factor X_{ad} also has its maximum value around this enrichment. Annual plutonium requirement is quite high, approximately 1.55 T/GWe, though 493 kg of U^{233} and 114 kg of Pu are left in the discharge fuel. As seen from Fig. 7, this cycle is a plutonium burner as the fissile content (Pu^{239} + Pu^{241}) in the discharge fuel is only about 32%. Further, due to the U^{232} that will be present, this cycle will also be highly proliferation resistant.

7.2.2. Pu in outer 2 rings

For evaluating a typical case of this cycle, we consider 2.75% of plutonium in thorium in the outer 2 rings and pure thorium in the inner 2 rings. The discharge burn-up increases from 11600 to 16100 MWd/T after 5 cycles, while plutonium requirement decreases from 2.34 to 1.68 tons. Even though the fissile content (2%) here is more than that in the Th–U^{235} cycle (1.3%), the discharge burn-up is lower due to the higher absorption cross-section of thorium. U^{233} fraction is nearly constant around 12.5% during the 5th cycle as shown in Fig. 8. Approximately, 450 kg of U^{233} per GWe will be available at the end of 5th cycle.

7.3. Th–U^{233} cycle

As in earlier cycles, here also two cases are considered: (i) Th–U^{233} uniformly in all the pins of the cluster [13] (ii) outer ring (18 pins) containing Th–Pu mixture. The second case is found to produce excess U^{233} at the same time will be proliferation resistant and plutonium burner.

7.3.1. Uniform U^{233} in all rings

A low enrichment of 1.55% of U^{233} shows the possibility of realizing a nearly self-sustained system needing only 57 kg/GWe-Y, with a high advantage factor of 21. However it requires large initial inventory of 2.7 tons of pure U^{233} per GWe. But burn-up is quite

Table 3
Uniform Pu in all rings

Enrichment	Burn-up MWd/T	Total heavy metal T/GWe-Y	Pu needed kg/GWe-Y	U^{233} in discharged fuel kg/GWe-Y	Pu in discharged fuel kg/GWe-Y	Advantage factor X_{ad} MWd/gm
2%	6138	198	3960	846	1130	2.66
3%	23 497	51.8	1550	493	114	3.05
4%	42 351	28.7	1150	330	32	2.96

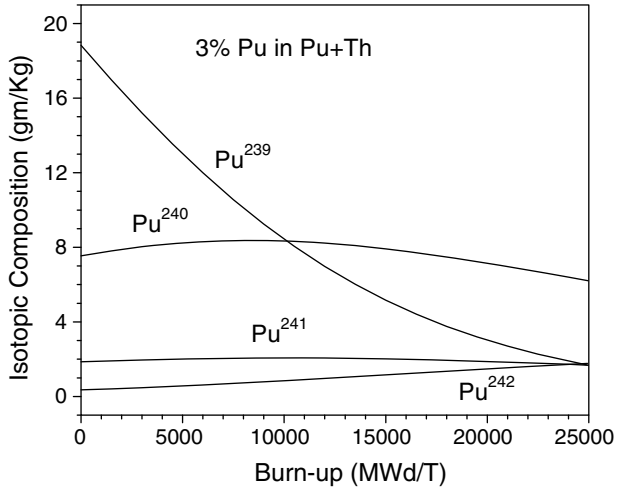

Fig. 7. Th–Pu (in all rings) cycle.

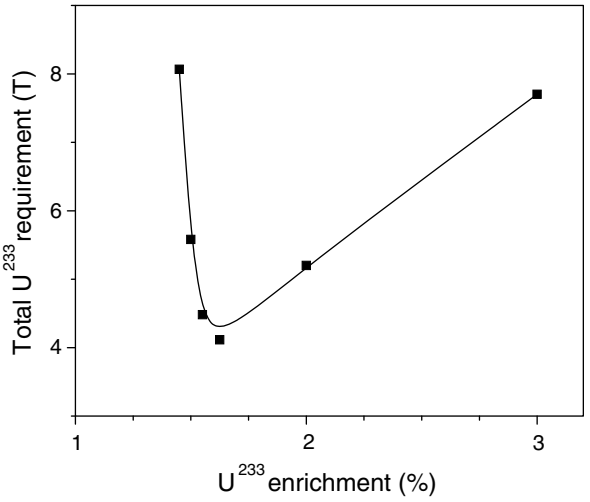

Fig. 9. Th–U^{233} (in all rings) cycle.

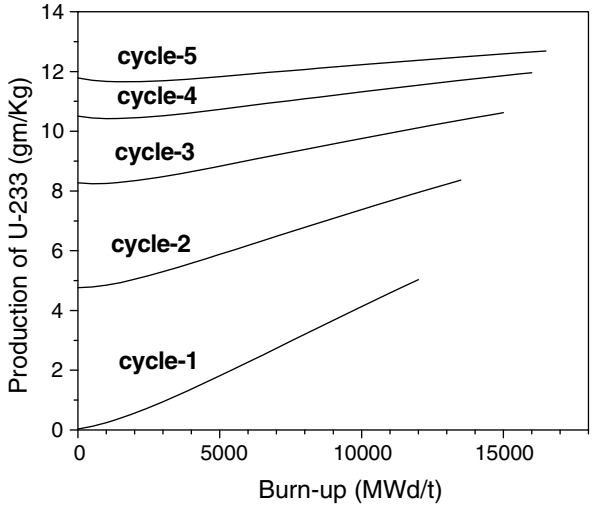

Fig. 8. Th–Pu (only in 2 outer rings) cycle.

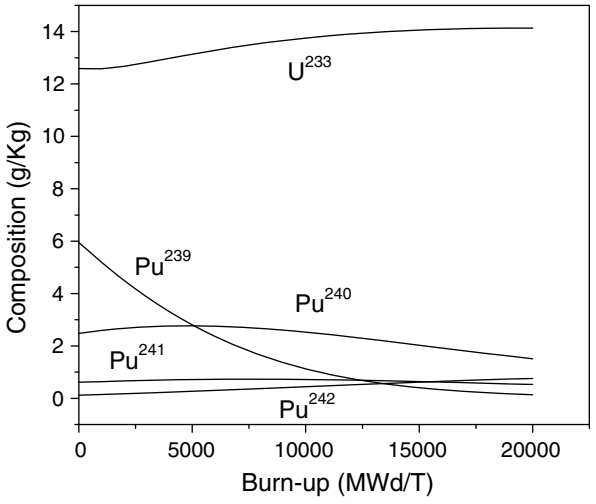

Fig. 10. Th–U^{233}–Pu (only in last ring) cycle.

low, 6800 MWd/T. With slightly higher enrichment, 1.625%, a higher burn-up of 12 500 MWd/T is obtained, though 83 kg/GWe-Y of U^{233} will have to be supplemented from external sources. As the enrichment increases further, burn-up increases but the advantage factor reduces thereby increasing the annual requirement. All these results are summarized in Table 4. Total amount of U^{233}

Table 4
Uniform U^{233} in all rings

U^{233} enrichment	Burn-up MWd/T	Total U^{233} consumption kg/GWe-Y	Total U^{233} produced kg/Gwe-Y	Net U^{233} required kg/Gwe-Y	Advantage factor X_{ad} (MWd/gm)
1.55	6800	2772	2715	57	21
1.625	12 500	1625	1542	83	15
2.0	37 000	672	521	151	8
3.0	82 800	443	222	221	5.5

needed vs. enrichment is plotted in Fig. 9 assuming 30 years reactor life. At the optimum enrichment of 1.625%, the life time requirement of U^{233} for a 1 GWe reactor is only ~4.2 tons.

7.3.2. U^{233} in inner 3 rings and Pu in last ring

In order to reduce U^{233} requirement, a cycle with 3 inner rings (19 pins) containing Th–U^{233} and the outer ring (18 pins) with Th–Pu mixture is studied. A typical example is 2.5% U^{233} enrichment in the inner rings and 2% Pu in the outer ring. This configuration generated a burn-up of 19 500 MWd/T. Annual Pu and U^{233} requirements in equilibrium are, respectively, 605 and 760 kg per GWe. However, 860 kg of U^{233} will be left in the discharged fuel. This is clear in Fig. 10 where U^{233} and Pu compositions are shown. The almost complete burning of Pu^{239} is also evident.

A higher burn-up cycle is with 3% U^{233} in the inner rings and 3% Pu in the outer ring. This cycle will generate 45 000 MWd/T burn-up. Annual Pu and U^{233} requirements in equilibrium are, respectively, 394 and 417 kg per GWe. However, only 392 kg of U^{233} will be left in the discharged fuel, thereby needing 25 kg/GWe-Y from external sources. Pu^{239} burns out completely here also.

Power fraction from U^{233} and U^{235}, as a function of burn-up, for three cases considered in Sections 7.1.1, 7.1.2 and 7.3.2 are shown in Fig. 11a–c. Power from Th is small in the first case because of its use only in 7 pins. In the second case, the cycle averaged power from Th is about 50%. Finally, in the third case, because of the U^{233} feed and larger amount of Th, it contributes almost 75% of power.

8. Effectiveness of fissile materials

In earlier sections the use of all the three fissile materials, U^{235}, U^{233}, Pu^{239} is discussed along with natural uranium and thorium. It

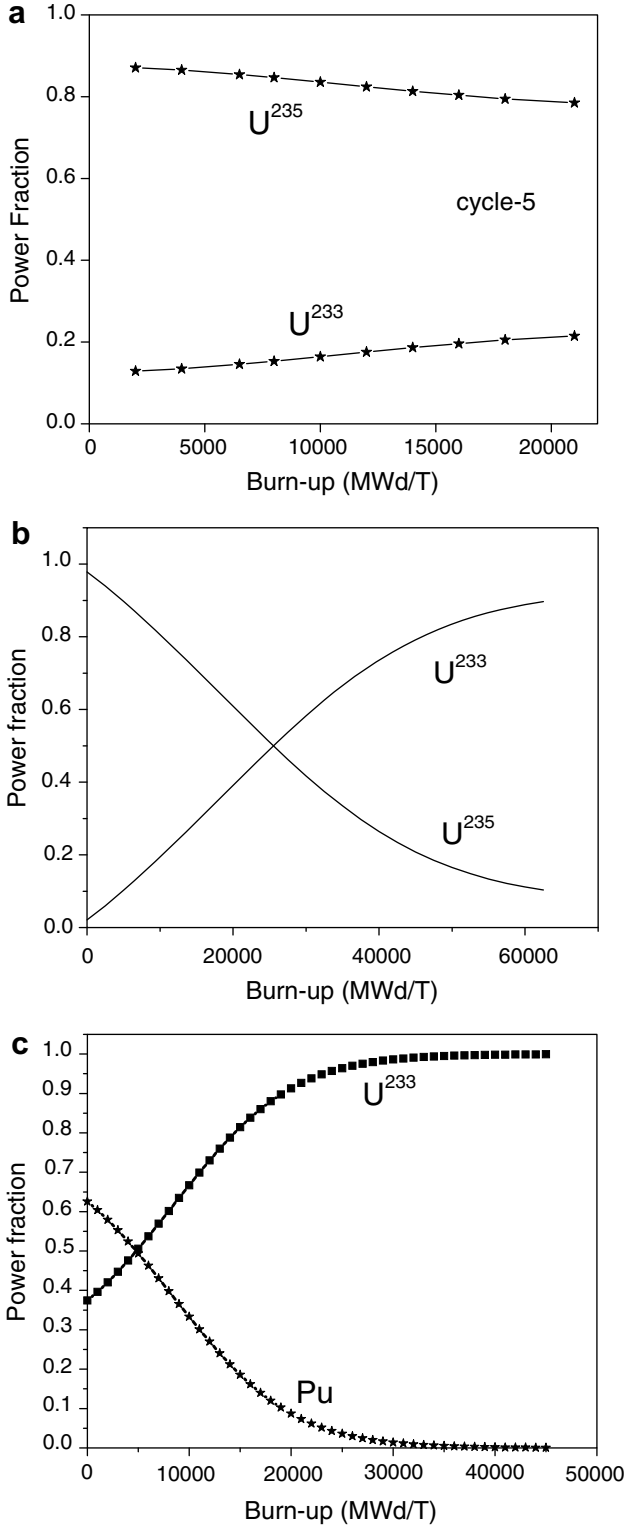

Fig. 11. Power fraction vs. burn-up.

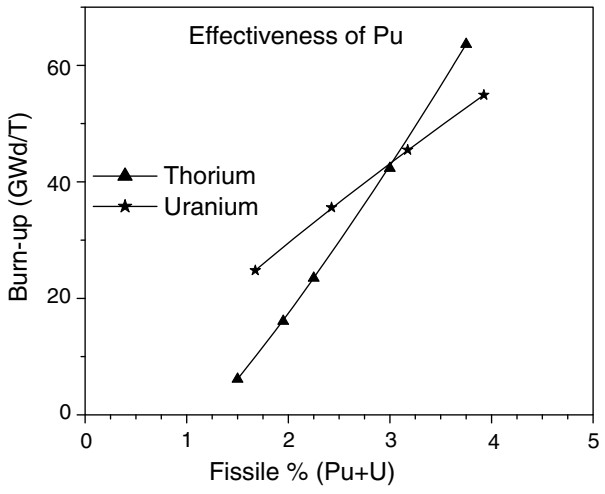

Fig. 12a. Burn-up vs. fissile% (Pu + U).

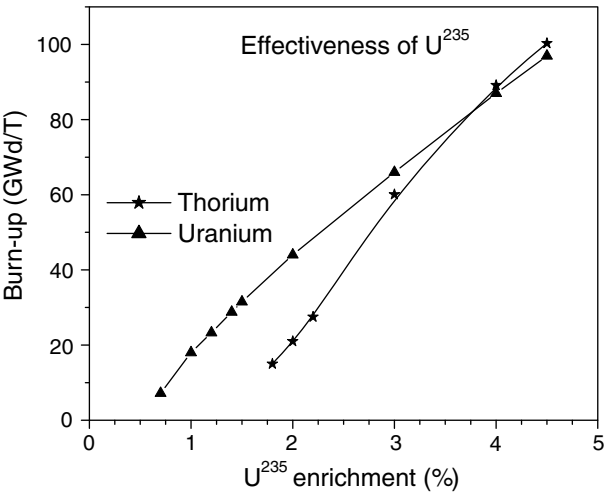

Fig. 12b. Burn-up vs. U^{235} enrichment.

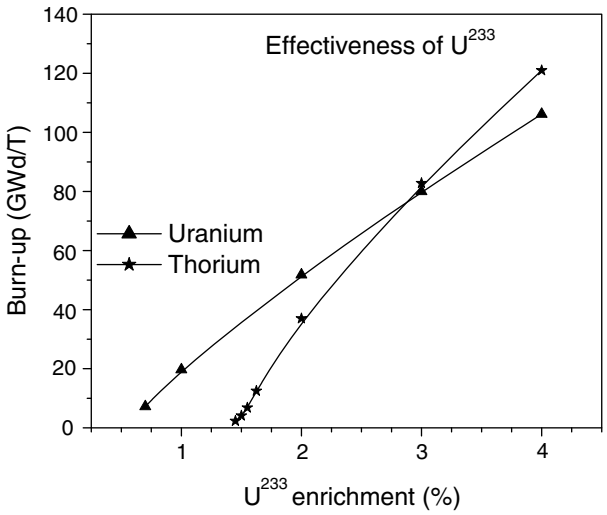

Fig. 12c. Burn-up vs. U^{233} enrichment.

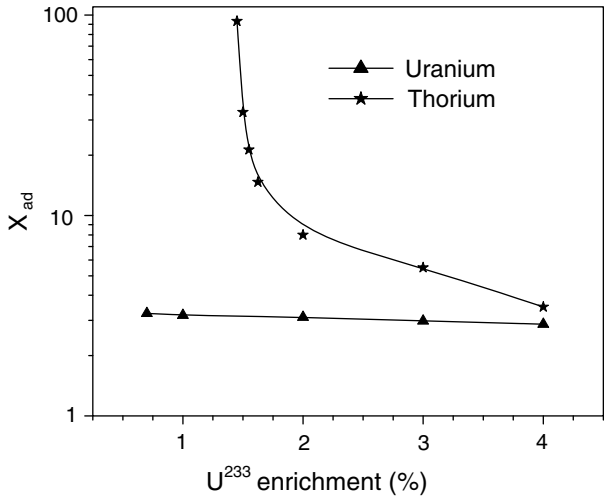

Fig. 13. X_{ad} vs. U^{233} enrichment.

is instructive to quantitatively compare their effectiveness from the burn-up point of view, that is, energy utilization factor X_{ut} defined earlier.

For simplicity, uniform enrichments of different fissile materials in all the pins of the cluster, either of Nat-U or thorium is considered. The fissile content of 0.7% of U^{235} in Nat-U is also added in case of U^{233} and Pu systems. The obtainable discharge burn-ups Vs fissile enrichment, shown in Fig. 12(a)–(c), indicates that thorium becomes more effective only after certain enrichment. For lower fissile enrichment, the uranium fuel will provide higher burn-ups. The cross over value of enrichment is lowest for U^{233} and highest for U^{235}. This is a manifestation of larger absorption cross-section of thorium and η of U^{233}. Finally, in Fig. 13 shows the advantage factor X_{ad} as a function of U^{233} enrichment in Nat-U and thorium systems. The possibility of self-sustaining system in Th–U^{233} cycle is evident from the high value of X_{ad} for low enrichment.

9. Conclusions

The salient features of the different fuel cycles considered in this paper are summarized in Table 5.

The main conclusions that emerge from these results are:

- SEU with 1.2% enrichment can enhance burn-up to 23 300 MWd/T and reduce Nat-U requirement by 40%. However, total amount of Pu produced will reduce by 50%.

Table 5
Comparison of different fuel cycles

Case	Fuel required T/GWe-Y	Nat-U saving T/GWe-Y (w.r.t Nat-U)	X_{ut} [a] (MWd/g)	X_{ad} [b] (MWd/g)	F_{ir} [c]	Burn-up (MWd/T)
Nat. U	168	Nil	1.0	3.2	0.69	7210
Slighly enriched uranium (SEU)						
1.2% U235	53	64 (38%)	1.9	2.8	0.32	23 300
1.5	39	68 (40%)	2.1	2.7	0.22	31 500
3.0	19	65 (38%)	2.2	2.5	0.12	66 000
Pu recycling						
0.45% Pu + Nat-U	70	98 (58%)	1.7	2.9	0.41	17 700
0.9% Pu + Dep-U	75	–	1.7	2.9	0.41	16 200
Thorium utilization						
ThO_2 with UO_2 (uniform)						
2% (U^{235})	60	No saving	1.0	3.8	0.74	20 100
3	20	48 (28%)	2.0	3.4	0.41	60 080
4	14	60 (36%)	2.2	3.2	0.31	89 150
ThO_2 with UO_2 (outer 2 rings)						
1.3% U^{235}	54	57 (34%)	1.7	3.2	0.47	22 500
Th with PuO_2 (uniform)						
2%	198	–	0.41	2.6	0.84	6140
3%	52	–	1.04	3.0	0.65	23 500
4%	29	–	1.41	2.9	0.51	42 350
ThO_2 with PuO_2 (outer 2 rings)						
2.75% Pu	75	–	1.7	3.2	0.47	16 100
Th + U^{233}						
1.625%	97	–	0.75	15	0.95	12 500
2%	33	–	1.81	8	0.77	37 000
3%	14	–	2.75	5.5	0.50	82 800
Th + U^{233}+Pu						
2.5% U^{233} + 2% Pu	62	–	0.96	5.1	0.81	19 500
3% U^{233} + 3% Pu	27	–	1.69	4.3	0.61	45 000

[a] X_{ut} is the total energy produced per gram of initial fissile inventory (MWd/gm).
[b] X_{ad} is the total energy produced per gram of net fissile (final-initial) consumed (MWd/gm).
[c] F_{ir} is the ratio of total fissile in discharged fuel to that in the initial inventory.

- Pu recycling yielding 17 700 MWd/T can save 60% of Nat-U. The plutonium composition in discharged fuel composition (Pu^{239}:Pu^{240}: Pu^{241}:Pu^{242}) is 42:38:9:11%. Thus the fissile content is reduced from 75% to 51%.
- From burn-up point of view, thorium utilization with U^{235} and Pu-fissile feed is found to be advantageous with enrichments higher than 3% and 4%, respectively.
- Efficient plutonium burning and proliferation resistant cycles are possible with Th–Pu feed in PHWRs.
- In situ production and burning of U^{233} is feasible in PHWRs, however, it will be competitive with uranium cycle only for enrichment ~3% as seen in Fig. 12C.
- A nearly self-sustaining Th–U^{233} system is feasible, however with low burn-up. With U^{233} generated externally, e.g. in fast reactors, intermediate or high burn-ups can be attained easily.

10. General remarks on fuel cycles

The possibility of employing different fuel cycles shown above indicates the flexibility of PHWRs to deal with advanced fuel cycle options. This is mainly due to its high neutron economy and on-power fuel management. Thus even the spent fuel from PWRs find great potential for use in PHWRs. Further, as it is shown, advanced fuels using SEU, MOX, Th/U^{233} cycles are feasible with high burn-up in PHWRs. Use of a new bundle with two pin sizes can reduce peak linear heat rating by about 20%.

Use of SEU has important advantages. For example, it provides 3–4 times the PHWR burn-up and reduces Nat-U requirement as much as 40%. The magnitude of spent fuel also is reduced by 60% which is significant from the point of view of reprocessing and waste disposal burden. However, the total annual production of Pu per GWe is also reduced. These features are generally true with regard to the cycles involving Pu recycling. SEU followed by Pu recycling with Dep-U can at best increase the energy potential by ~2 times that with OTC using Nat-U. However, it is well known that use of Pu in fast breeder reactors (FBRs) can provide very much higher installed capacity [1].

Use of Th in PHWRs needs either enriched U or Pu as fissile material. For example, 3% enriched U^{235} saves Nat-U by 28%. However, the amount of fuel which has to be handled annually, both at the front and backend, is just 20 tons per GWe, which is a significant reduction in comparison to 168 tons in case of Nat-U once through cycle. This is very well within the existing and proven technologies. There is an increase in installed capacity ~1.4 times and, additionally, production of ~200 kg of U^{233} per GWe in a year. This can further be used, say, in the 2% U^{233}–Th cycle providing 37 000 MWd/T burn-up. So the total capacity that can be generated is ~2.8 times that of Nat-U cycle. On the other hand, with a lower burn-up 1.625% U^{233}–Th cycle, 200 kg/GWe-Y can sustain two reactors, thereby increasing the capacity ~4.2 times. But this will require frequent reprocessing as burn-up will be only 12 500 MWd/T[1].

Pu requirement in Pu–Th cycles, generally, is much more than what can be produced using Nat-U in PHWRs of same capacity. For example, 4% Pu–Th cycle needs ~1150 kg/GWe-Y while a Nat-U fueled PHWR will provide only ~600 kg/GWe-Y. Thus the total installable capacity will be 1.5 times that with PHWR. However, the 330 kg/GWe-Y of U^{233} that will be produced can sustain two U^{233}–Th cycle (2%) thereby providing a total capacity of ~2.5 times. The combined Pu–Th–U^{233} is also similar as it can nearly be self-sustaining in U^{233} and Pu requirement can almost match with that produced in the PHWR. Thus these cycles can boost the capacity by ~2 times only. In short, direct utilization of Th in PHWRs will be constrained by the availability of fissile materials like U^{235} or Pu.

If U^{233} is produced elsewhere, e.g., in FBRs or accelerator driven sub-critical reactors, Th can increase installed capacity significantly. For instance, a net supply of 150 kg of U^{233} can sustain a 1 GWe reactor for one year with reasonable burn-up (37 000 MWd/T). The installed capacity will then be determined by the Th resources and technology for U^{233} production and reprocessing.

Thus the final conclusion is that maximizing the energy generation potential for a given amount of Nat-U must follow the well known three stage route, as enshrined in the Indian nuclear power programme.

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[1] This will require separation of U^{233} from other uranium isotopes. Schemes which do not need this isotopic separation are also feasible.