ALTERNATE FUEL CYCLES IN THE INDIAN PHWR

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1. INTRODUCTION

The Indian Pressurised Heavy Water Reactor (PHWR) has so far been working on the natural uranium once through cycle. But this reactor, with its advantages of on-load refuelling and short bundle length, is a versatile system which admits of many other fuel cycles. Every fuel cycle has its own distinct advantages and its own constraints. The selection of a fuel cycle for a reactor will depend upon the time and place, the economics and politics. Some of the performance criteria based on which fuel cycles are designed are (a) fuel utilisation, (b) shrinking of spent fuel inventory, (c) reduction of long-lived actinide wastes, (d) operational ease for high burnup cycles, (e) retrofittability into existing systems, (f) fuel cycle flexibility, and (g) dispositioning of weapons plutonium. In the international arena, fuel cycles designed for non-proliferation and safeguards considerations, and inert matrix cycles for plutonium destruction are also receiving attention. These cycles will however, not be addressed in this paper.

From the Indian perspective, the most important characteristic of any fuel cycle is the energy obtained per ton of uranium mined. From this point of view, a number of fuel cycles have been examined. These include (i) slightly enriched uranium (SEU), (ii) natural uranium with self-generated plutonium, (iii) recovered uranium with self-generated plutonium, (iv) natural uranium with dismantled weapons plutonium, (v) recovered uranium with dismantled weapons plutonium, (vi) thorium-U233 self-sustaining cycle with U235 makeup, (vii) thorium-U233 self-sustaining cycle with reactor grade plutonium makeup,(viii) thorium-U233 self-sustaining cycle with weapon plutonium makeup, (ix) thorium with U235 in high burnup cycle, (x) thorium with reactor plutonium in high burnup cycle, (xi) thorium with weapon plutonium in high burnup cycle. Then there are the various combinations that can be clubbed together as different versions of the once through thorium (OTT) cycle, viz., (xii) unenriched thorium and SEU in segregated channels, (xiii) unenriched thorium and natural uranium with reactor plutonium in separate channels, (xiv) unenriched thorium and natural uranium with weapon plutonium in separate channels, (xv) unenriched thorium and recovered uranium with reactor plutonium in separate channels, (xvi) unenriched thorium and recovered uranium with weapon plutonium in separate channels. Finally, we have also made a study of a BWR-PHWR tandem cycle.

In this paper, we describe all these cycles and the results of the studies. As mentioned, the major emphasis will be on the maximisation of energy from any given quantity of fissile material. But wherever a cycle offers other benefits, these have been pointed out.

The BWR-PHWR tandem cycle has been described in some detail. This has been done with the intention of showing the kind of studies that are needed before a cycle can actually be introduced into a reactor. Some of the problems that arise are pointed out and possible solutions discussed.

2. DESCRIPTION OF THE REACTOR

The Indian PHWR is a tube type reactor using heavy water as both coolant and moderator. The coolant is physically separated from the moderator by being contained inside the pressure tube where it is maintained at high temperature (~ 270 ^{o}C) and pressure (~92 bar). The moderator heavy water is at relatively low temperature (~ 55 ^{o}C) and is unpressurised.

The reactor core consists of 306 pressure tubes arranged along a square lattice of 22.86 cm pitch. The fuel pins and the coolant are contained within these pressure tubes. The direction of coolant flow in adjacent channels is in opposite directions. The fuel is in the form of a string of 12 bundles. Each bundle is a 19-rod cluster of 49.5 cm length. Of the 12 bundles, 10 are in the active portion of the core, the remaining 2, one on each end, are outside the core.

Refuelling is done on-power by simply pushing out 8 bundles from a channel on one end, while 8 fresh bundles are inserted from the other end. The direction of the bundle movement is the same as that of the coolant flow, so that alternate channels are fuelled in opposite directions. This helps in maintaining overall axial symmetry.

Table 1 gives a general description of some of the important physical parameters of the core of the Indian PHWR.

Table 1
DESCRIPTION OF THE PHWR REACTOR CORE

Number of fuel channels	306
Lattice pitch, cm	22.86
Calandria inner radius, cm	299.8
Calandria length, cm	500.0
Number of bundles per channel inside the active portion of the core	10
Extrapolated core radius, cm	303.3
Extrapolated core length, cm	508.5
Number of absorber rods (for xenon override)	4
Number of regulating rods (for reactor regulation)	2
Number of shim rods (to provide backup for regulation)	2
Number of mechanical shutoff rods (SDS-1)	14
Number of liquid poison tubes (SDS-2)	12
Total thermal power to coolant, MWth	756
Maximum channel power, MW	3.2
Maximum bundle power, kW	440
Maximum coolant outlet temperature, °C	297
Reactivity worth of SDS-1, mk	31.9
Reactivity worth of SDS-2, mk	32.1
Coolant inlet temperature, °C	249
Average fuel temperature, °C	625
Average coolant temperature, °C	271
Specific power, kW/kg	19.2

3. NATURAL URANIUM BASED CYCLES

Using natural uranium fuel as the base and increasing the fissile content by addition of external fissile material, a number of cycles are possible. The added fissile material could be self-generated plutonium or weapon plutonium (in which case the fuel is MOX), or U235 itself, leading to the most popular alternate cycle, viz., slightly enriched uranium (SEU). We consider SEU here in once through mode, and MOX is restricted to one recycle of plutonium.

Even though the correct way to evaluate resource utilisation is to express the results in terms of energy obtained per ton of uranium mined, the following difficulty arises. In the case of SEU, this conversion depends upon the amount of U235 lost in the enrichment plant tails. This value is sometimes quoted as 0.2%, sometimes as 0.1%, or at times even lower. In the case of weapon plutonium, one does not know how to make any assumptions on how it was produced or how much natural uranium was used to create it. It is only in the case of self-generated plutonium that any definitive numbers can be assumed. We obviate this difficulty by expressing our results in terms of energy per kg of U235 or plutonium. To make a meaningful comparison between SEU and MOX, we assume that the quantity of plutonium in the MOX is the sum of a certain "real" plutonium and "virtual" plutonium. This "virtual" plutonium is computed as the equivalent of the U235 content of natural uranium. It is defined as that amount of plutonium which would give a burnup equal to that of natural uranium.

Figure 1 shows this as a function of discharge burnup. Discharge burnup has been chosen as the independent variable since it is one of the most important performance criteria of any fuel cycle. The highest fuel utilisation is just above 1,700 MWD/kg fissile and is more or less the same for U235, reactor plutonium, or weapon plutonium. This comes in the neighbourhood of 27,000 MWD/T discharge burnup for SEU, and 20,000 MWD/T for plutonium. For high burnups, SEU gives much better utilisation than plutonium. This is only to be expected since higher initial enrichment will be needed for higher burnups. With the higher cross-section of plutonium, the MOX ends up burning more of the initial fissile material, while the SEU manages to convert more U238 to plutonium and burn it. There isn't much difference between the two plutoniums, but surprisingly, the reactor plutonium seems to have an edge over the weapon plutonium.

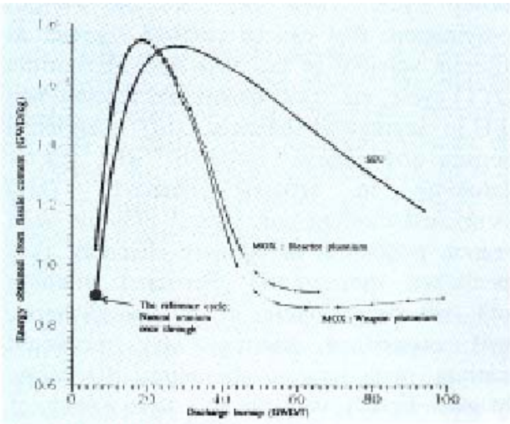

Figure 1 : Fuel cycles in PHWR with natural uranium base.

Again, may be it need not be such a surprise since the reactor plutonium contains Pu241 and also Pu240 which gets converted into Pu241, and Pu241 is a better fissile material than Pu239. All these numbers may be compared to the value obtained for the reference cycle, i.e., the natural uranium once through cycle, which has a discharge burnup of 6,270 MWD/T and can be said to have a fissile material utilisation of 896 MWD/kg fissile. This point is indicated in the figure by an asterisk. One can see that all the cases presented here have fuel utilisation characteristics superior to the natural uranium once through cycle.

4. RECOVERED URANIUM BASED CYCLES

Similar studies can be made for fuel with recovered uranium as base. The discharged fuel from the PHWR natural uranium once through cycle contains 0.255% U235. Since this is considered as discarded material in the reference cycle, only the added fissile content is used in the calculation of energy from fissile material. The SEU cycle in which recovered uranium is base, thus shows a higher fissile utilisation than the one with natural uranium base. Obviously this is because the residual U235 in the discharged fuel is being burnt. Once again, all three curves lie totally above the asterisk representing our reference case. This can be seen from Fig. 2.

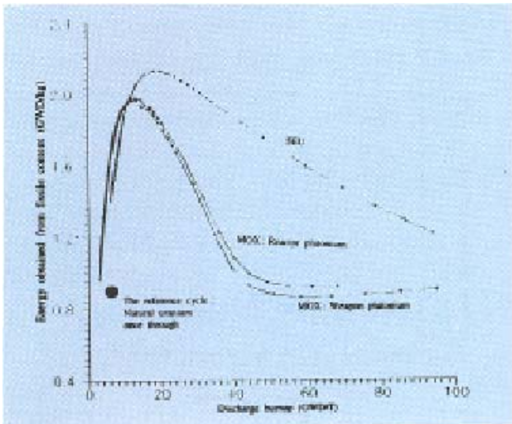

Figure 2: Fuel cycles in PHWR with recovered uranium base.

5. THORIUM CYCLES

The PHWR is one of the best reactors to use thorium, second probably only to the MSBR. Possible thorium cycles fall into three categories, which can be called the self-sustaining equilibrium thorium cycles (SSET), the high burnup open cycles, and the once through thorium (OTT) cycles.

5.1. THE SELF-SUSTAINING EQUILIBRIUM THORIUM CYCLE (SSET)

The fuel in this cycle is thorium-U233. The discharged fuel should contain enough U233 to provide the fissile content for the next fuel charge. Thus this cycle, once initiated, can go on indefinitely without any external fissile input. In the limit, its energy potential is theoretically infinite until thorium runs out.

There is the question of how the SSET is to be initiated. There are two ways of doing this. The first is to enrich the thorium with U235, extract U233 from the spent fuel and use it for the first charge of the SSET. About 250 tons of natural uranium will be required to place one PHWR on the SSET. The alternative route is the plutonium route, in which the reactor is first fuelled with natural uranium. The plutonium extracted from the discharged fuel is mixed with thorium and fed back into the reactor. This fuel when discharged, will contain U233, which can be retrieved and used to initiate the SSET. This route will require about 500 tons of natural uranium for one PHWR.

The discharge burnup possible with the SSET is rather low being about 11,000 MWD/T. For a cycle that depends on repro-cessing and refabrication of highly active U233 bearing fuel, this is too low a value to be economical. The burnup can be improved by adding a certain amount of makeup fissile in the new fuel along with the extracted U233. We have carried out studies with U235 makeup, reactor grade plutonium makeup, and weapon plutonium makeup. Figure 3 gives the results obtained with all the three. The weapon plutonium gives results that are marginally superior to reactor plutonium, while U235 is superior to both plutoniums.

Some safety related parameters of interest are as follows. The positive void coefficient of reactivity (on complete voiding) is 5.98 mk as compared with the value of 10.84 mk for the natural uranium core. The effective delayed neutron fraction is 2.75 mk (6.91 mk for the natural uranium core). In the event of a LOCA followed by over power trip, the peak power reached is 2.005 times full power (cf. 2.15), peak fuel temperature is 866 °C (cf. 864 °C).

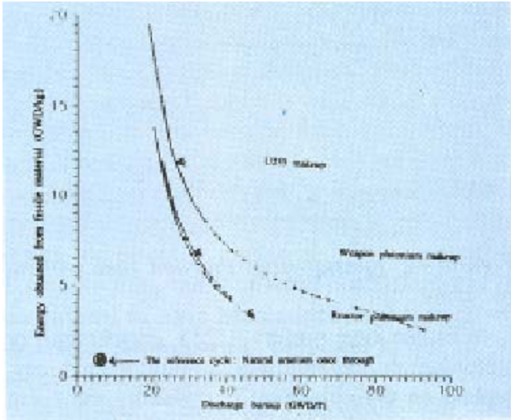

Figure 3 : Utilisation of makeup fissile material in SSET

As for operational parameters, the adjustor rod worth reduces from 8.25 mk for the natural uranium core to 6.28 mk for the SSET core.

5.2. THE HIGH BURNUP OPEN CYCLE

Here we do not close the cycle, and the intent is to get as high a burnup as possible. If direct disposal of spent fuel is contemplated, this cycle assumes importance in shrinking the spent fuel inventory. This cycle is also suitable for destroying the accumulated civil and military plutonium. The economics of this cycle is very favourable as compared to that of the SSET. Fuel utilisation is significantly lower. One possible way to improve resource utilisation is to store the spent fuel so that if at any time in the near or distant future there is a change in the position of cheap uranium, this fuel can be reprocessed to recover U233.

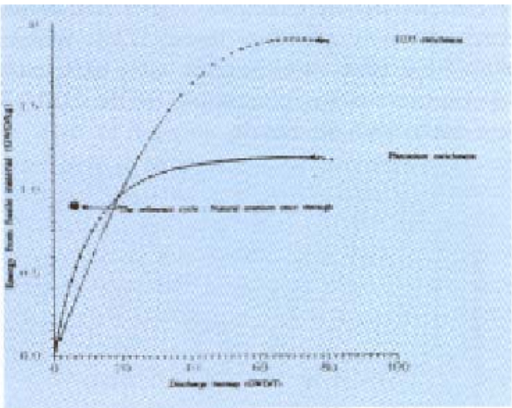

Figure 4 : Energy from thorium fuel without recycling

Whether one prefers U235 enrichment or plutonium enrichment will, among other things, depend on the objective. If shrinking the spent fuel inventory is the objective, U235 enrich-ment is more advantageous in that it gives better fissile utilisation. This can be seen from Figure 4 where the energy obtained from 1 kg of fissile material is compared for U235 and plutonium. On the other hand, if plutonium dispositioning is sought, this cycle is a very effective one. Figure 5 shows that this cycle can destroy up to 95% of the fissile plutonium.

5.3. THE ONCE THROUGH THORIUM (OTT) CYCLE

This scheme is based on reasons not really related to either fuel utilisation or fuelling costs, but rather on operational acceptablity. Two main arguments can be advanced for this cycle.

1. **Because of a general slowing down of the nuclear power programme the world over, it is difficult today to persuade utilities to load thorium. In the OTT scheme, we have a way of introducing thorium that causes the least disruption in the normal fuelling of the reactor.**
2. **One of the advantages of the thorium cycle over the uranium cycle has been the level of long-lived actinides which pose a waste disposal problem. This is orders of magnitude lower in the thorium cycle. But to take advantage of this, there is a need to keep thorium from being "contaminated" by the uranium cycle, i.e., direct mixing of the thorium with any part of the uranium cycle, e.g., U235 or plutonium, should be avoided.**

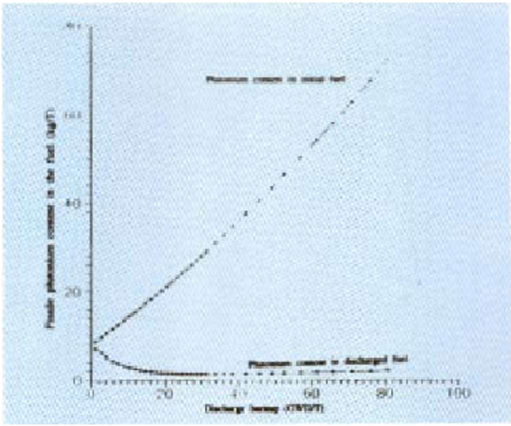

Figure 5 : Plutonium destruction in thorium-plutonium fuel

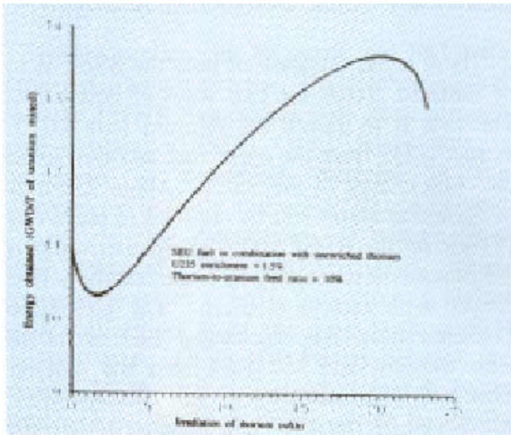

Figure 6 : Once through thorium (OTT) cycle in PHWR

The principle of this cycle therefore is to use plain unenriched thorium along with SEU in segregated channels of the PHWR. Being in different channels, their burnup can be varied independently. The presence of thorium

acts as a load on the SEU, which will therefore have to be discharged at a lower burnup. The total energy extracted will be the sum of the energy obtained from the thorium and the SEU. As the residence time of thorium in the core increases, the energy obtained from unit mass of mined uranium will first decrease, then increase, and finally become higher than it would have been had no thorium been present in the core. Figure 6 shows this graphically. Similar results are also seen when thorium is combined with MOX, i.e., natural uranium with either reactor or weapon plutonium.

6. THE LWR-HWR TANDEM CYCLE

Some work is being done in India on the possibility of using the discharged fuel from LWR as feed for the PHWR. At present, India has only two units of light water reactors of 160 MW(e) capacity each. They are boiling water reactors. The proposal is to subject the discharged fuel from the BWRs to a process known as coprocessing, in which the fission products are removed, but all the actinides are precipitated together. There is thus no separation of plutonium and uranium, a fact which enhances the value of this process from the point of view of non-proliferation. This mixture of heavy metals is then fabricated into the19 rod bundles of the Indian PHWR, and introduced into the heavy water reactor. A brief description of the BWR lattice is given in Table 2. The isotopic composition of the fuel which is discharged at a burnup of 21,000 MWD/T is as given in Table 3. Fuel of this composition, fabricated into 19 rod clusters and loaded into the PHWR can give a discharge burnup of 37,200 MWD/T.

This means a support ratio of about 1.7, which means that 1 GW(e) of BWR can support the fuelling requirements of 1.7 GW(e) installed capacity of PHWR. This ignores losses during fabrication, coprocessing, etc., but if all those factors were also taken into account, one can still expect a support ratio of 1.5.

Table 2
THE BWR LATTICE

Number of fuel assemblies	284
Mass of UO2 in one assembly, kg	160
Mass of heavy metal in one assembly, kg	141
Total uranium inventory, ton	40.0
Specific power, kW/kg	17.5
Fuel pellet radius, mm	6.18
Clad inner radius, mm	6.34
Clad outer radius, mm	7.15
Pin-to-pin pitch, cm	1.79
U235 enrichment (average), %	2.4
Discharge burnup, MWD/T	21,000

Table 3
ISOTOPIC COMPOSITION OF BWR DISCHARGED FUEL (%)

Uranium-235	1.02
Uranium-236	0.28
Uranium-238	95.15
Plutonium-239	0.92
Plutonium-240	0.21
Plutonium-241	0.13
Plutonium-242	0.026
Americium-241	0.006

The void coefficient of this lattice is about 6.7 mk (positive). This is lower than the value (+10.8 mk) for natural uranium. However, the delayed neutron fraction is 4.12 mk as opposed to 6.5 mk in the case of natural uranium. An approximate LOCA analysis was carried out to evaluate the effect of this. It was assumed that the PHT voiding process, at least in so far as the reactivity effect is concerned, is complete in 1 sec., so that the total void coefficient can be considered to have been added to the reactivity in 1 sec. Blowdown is supposed to last for 10 secs., during which period there is improved heat transfer. Thereafter, the ECCS will come in. However, in this calculation, this has been neglected and so beyond 10 secs. the heat transfer has been assumed to be negligible.

The shut down in this LOCA calculation is initiated by the over-power trip. Shut down is by moderator dumping. Among Indian PHWRs, there are four units built in the early days, which have shut down by moderator dumping. Later units have two independent fast acting shut down systems. The tandem cycle which we are examining is supposed to be introduced into one of the earlier reactors. Hence the use of moderator dumping for reactor shut down in the studies presented here.

The peak power in the tandem fuelled PHWR under LOCA reaches 1.93 times nominal power and peak fuel temperature of 862°C. This may be compared with peak power of 1.6 times nominal power and peak fuel temperature of 856°C in the case of the natural uranium equilibrium core and peak power of 2.15 times nominal power and peak fuel temperature of 864°C for the fresh natural uranium core.

The PHWR has a set of eight rods called adjuster rods (AR) to provide xenon override. A set of four rods called regulating rods (RR) are used for reactor regulation. In the natural uranium core, the ARs have a reactivity worth of 8.2 mk, and the RRs have a worth of 4.8 mk. In the tandem fuelled core, the AR worth is 5.2 mk, and the RR worth is 3.2 mk. This reduced AR worth is however, compensated by a change in xenon worth. Whereas the 8.2 mk in the natural uranium core could give a xenon over-ride time of 30 minutes, the 5.2 mk in the tandem fuelled core gives an over-ride time of about one hour. So the AR worth reducing to 5.2 mk is of no concern.

The reduction in RR worth may have some implications for operability, but with some backup worth provided by boron in the moderator, this can be managed.

The reactivity of the initial core will be very high, of the order of 200 mk. It is possible to suppress this either by loading a number of thorium bundles, or by using boron in the moderator. The boron level will have to be very high. But the major disadvantage of boron will be that power peaking in the initial core will necessitate the derating of the reactor. As core burnup proceeds, the flux will tend to flatten. By about 90 effective full power days, both bundle power and coolant outlet temperature will have reached acceptable values. Refuelling however, will be required only after about 900 EFPD and continued burnup will worsen the power distribution, because the low flow channels towards the periphery of the core will now produce more power than can be removed by the flow which has been designed to cater to the power distribution of the natural uranium equilibrium core.

Studies have shown that while the maximum bundle power in the core goes on decreasing until the core reactivity falls to zero at about 925 EFPD, the maximum coolant outlet temperature reaches a lowest value of 295.0°C at 116 EFPD, and thereafter starts increasing again. At 925 EFPD, it has touched 311.9°C. The three constraints on power are the bundle power, channel power, and coolant outlet temperature. Figure 7 shows, as a function of EFPD, three quantities, which we have chosen to call bundle ratio (B), channel ratio (C), and temperature ratio (T). The bundle ratio is defined as the ratio of the highest bundle power in the core to the maximum permitted bundle power. C is a similar quantity for channel power, and T for coolant outlet temperature.

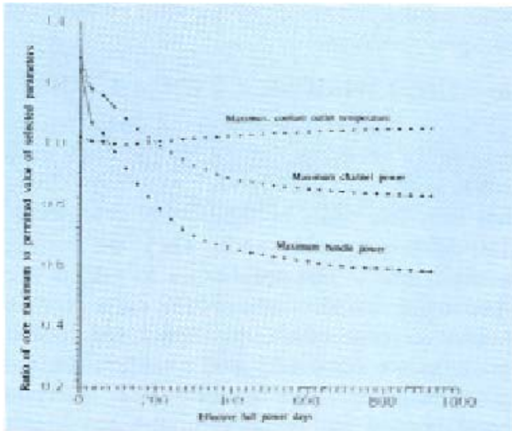

Figure 7 : Variation of power constraints wth burnup

For full power to be possible, all three of them should be less than unity.

When any of them exceeds unity, it also is the factor by which the reactor power needs to be derated in order to stay within design constraints. We see from Figure 7 that whereas B and C start from values exceeding unity at zero EFPD, and decrease consistently right up to the first refuelling at 925 EFPD, T goes through a minimum and then rises again, reaching quite high values towards the end. The reason for this is that we are retrofitting the tandem fuel into a core originally designed for natural uranium, in which the coolant flows have been already fixed through appropriate orificing to suit the natural uranium fuel. Over the extremely long pre-fuelling life of the tandem core, burnup induced power flattening reaches such proportions that the low flow channels at the core periphery end up producing much more power than they were designed to cater to.

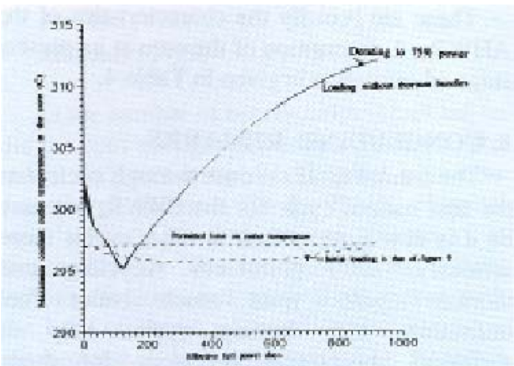

Figure 8 : Variation of maximum channel outlet temperature with burnup

Figure 8 shows the variation of maximum coolant outlet temperature with effective full power days of operation in the core without any thorium bundles. Also shown in Figure 8 is a similar curve for a core in which thorium bundles have been deployed to give full power in the beginning. (This loading is shown in Figure 9). While in the core without thorium bundles, the coolant outlet temperature crosses its permitted limit again by about 180 EFPD, in the core of Figure 9 this never happens. The core can give full power right up to the time that fuelling starts, beyond which of course, the power distribution is controlled by proper fuelling.

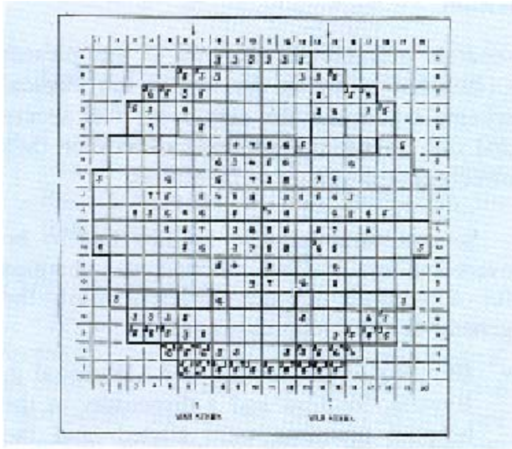

Figure 9 : Thorium bundle distribution in the tandem loaded core.

7. THE ADVANCED HEAVY WATER REACTOR (AHWR)

While the thorium cycle in the PHWR offers many attractive possiblities, the fact remains that all of them are retrofits into the PHWR which was designed with the uranium cycle in mind. The AHWR was designed to take advantage of the specific nuclear characteristics of thorium.

If we look at the SSET as a starting point for a thorium cycle, we should address its main weakness, viz., its low burnup. The burnup can be increased by adding plutonium makeup. The capture-to-fission ratio of Pu239 is rather high in the soft spectrum of the PHWR. A harder spectrum and tighter lattice are therfore advisable. In a uranium system, the harder spectrum will be to the detriment of the U235, but the eta of U233 is almost flat and so it does not matter.

Thermal absorption of thorium is about three times as large as that of U238. So parasitic absorption is less of a problem in thorium systems and one could try light water coolant. With light water coolant, it is possible to have boiling coolant and direct cycle. Since the spectrum is hard, things could be so managed that the coolant void coefficient is negative.

Table 4
DESCRIPTION OF AHWR AT THE PRESENT STAGE OF EVOLUTION

Reactor power, MWth	750
Reactor power, MWe	220
Fuel description:	
Number of pins	52
Number of Pu bearing pins	20
Number of thorium-U233 pins	32
Number of water tubes for ECCS injection	8
Plutonium content in MOX, %	2.7
U233 content in thorium	self sustaining
Coolant water density, (varies in the range)	0.50-0.55
Total number of channels	428
Number of seeded clusters	344
Number of clusters with all "thorium-U233" pins	84
Number of fuelling zones	2
Number of reconstitutions	1
Thorium pins discharge burnup MWD/T	20,000 approx.
MOX pins discharge burnup, MWD/T	20,000 approx.
Lattice pitch, cm	29.4
Active fuel length, cm	350
Moderator and reflector	D2O
Scatterer balls in the moderator	pyrocarbon
Calandria radius, cm	430
No. of adjustor rods	4
Worth of adjustor rods, mk	5.77
No. of regulating rods	4
Worth of regulating rods, mk	4.53
No. of SDS-1 rods	32
Worth of SDS-1, mk	54.5
Performance data in the equilibrium core Radial form factor	1.41
Hot spot factor in the seeded cluster	1.42
Hot spot factor in the thorium-U233 cluster	1.36
Maximum channel power, MWth	2.45
Maximum-to-minimum channel power factor	1.88
Fraction of power from thorium, %	75.4

With boiling coolant, the reactor has to be vertical. In this case, it can be designed to have 100% heat removal by natural circulation and also to have passive safety.

These are broadly the characteristics of the AHWR. A description of the core at its present stage of evolution is given in Table 4.

8. CONCLUDING REMARKS

The natural uranium once through cycle was the first natural cycle for the PHWR, but may be it is now time to look at other cycles more seriously. Both plutonium recycling and thorium cycles yield much better fuel utilisation, high burnup cycles help in shrinking spent fuel inventory for those countries for whom direct disposal is a constraint. These cycles are also very efficient in constraining plutonium where that is of importance. The choice of a cycle will depend upon the compulsions of any country or company or utility, but an array of options is available to choose from.