Nuclear power from thorium: different options

S. Banerjee, H. P. Gupta and S. A. Bhardwaj*

Thorium is a fertile material that has drawn attention as a potential source of nuclear energy since the 1950s due to several attractive features of the Th–U^{233} fuel cycle. In view of the renewed interest in thorium, the possibilities of thorium utilization in different reactor systems, namely pressurized heavy water reactors (PHWRs), light water reactors, molten salt breeder reactors (MSBRs), fast reactors and accelerator-driven sub-critical systems have been examined. Extraction of energy from thorium essentially requires prior conversion of thorium to fissile U^{233}. For in situ burning of thorium, a high burn-up is therefore essential. It is shown that the use of thorium in currently deployed PHWRs will reduce the requirement of uranium by about 30% in once through fuel cycle, while MSBRs with closed fuel cycle can achieve near breeding capability in thermal reactors. The most effective thorium utilization can be achieved only by adopting a closed fuel cycle which will not only enhance the fissile inventory many fold but also reduce nuclear waste burden significantly. While in conventional fast breeder reactors, thorium, partly converted into U^{233} in the blanket region, is reprocessed for the recovery of the fissile material; in the breed and burn concept, the converted material is transferred to the core region without any reprocessing. Availability of spallation neutrons produced by bombardments of high-energy protons on heavy nuclides can augment fertile to fissile conversion leading to thorium utilization. The various options, which appear technologically feasible for generating power from thorium and the key issues connected with these schemes, are discussed in this article.

Keywords: Accelerator-driven sub-critical systems, breed and burn reactors, fast reactors, light water reactors, molten salt breeder reactors, pressurized heavy water reactors, thorium fuel cycle.

NATURE has provided only one fissile isotope, U^{235}, splitting of which in fission chain reaction produces most of the present day nuclear energy. Natural uranium contains 0.7 wt% of U^{235}, which is fissile material while the remaining 99.3 wt% U^{238} is fertile. In contrast, natural thorium (Th^{232}) is only a fertile material with no fissile content in it. While examining the sustainability of nuclear power, the use of thorium was given due consideration from the early years of the nuclear power generation. In view of the fact that the total inventory of naturally available U^{235} is inadequate to provide nuclear energy for a long period (beyond a century or two), the idea of utilizing fertile material by first converting them into fissile and subsequently fissioning them to produce energy was conceived way back in the 1950s. It was also recognized that the fertile to fissile conversion could be made possible only if a steady supply of neutrons can be made economically. The process of nuclear fission produces neutrons in excess of what is required for sustaining the chain reaction and these excess neutrons are utilized for the generation of fresh fissile nuclides. While the fast neutron spectrum fissioning of Pu^{239} can generate more fissile nuclides than consumed, U^{233} (derived from Th^{232} by neutron absorption) can also do the same at a reduced level in a wide neutron energy spectrum from thermal to fast. It is this nuclear property, which was the main incentive of thorium utilization in the early years. Other advantages of the thorium fuel cycle are: reduced generation of higher actinides with long radioactive life and attractive thermo physical properties of ThO_2 (ref. 1).

Feasibility of the scheme of converting Th^{232} to U^{233} and subsequently fissioning U^{233} to produce nuclear energy was demonstrated in several countries, a brief account of which is provided later in this article. However no major power plant has been built with U^{233} as the main fuel. This is primarily because driving a thorium-based reactor requires a driver fuel containing any of the fissile nuclides U^{235}, U^{233} or Pu^{239}. The fissile fertile mix U^{235} + U^{238} in natural and enriched uranium is being used extensively over the last six decades for producing energy and generating fresh fissile nuclides. The capture cross-section of Th^{232} is nearly thrice higher than that of U^{238} in thermal neutron spectrum, therefore, introduction of thorium will invariably require larger fissile content in

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the reactor core for criticality in comparison to uranium. In order to extract energy from thorium effectively, it is necessary to allow continued conversion of thorium into fissile U^{233} *in situ* for longer periods. This requires relatively higher fuel burn-up. If one compares the effectiveness of the fertile nuclides, U^{238} and Th^{232}, mixed with identical fissile driver fuel, thorium-containing fuel will score better, only when fuel burn-up exceeds about 50,000 MWd/t.

In recent times, there has been a revival of interest in thorium[2]. A recent report issued by OECD-NEA[3] is indicative of the growing interest. The factors, which are responsible for this revival, are as follows

- Fissile inventory accumulated in several countries is adequate for initiating a fuel cycle with Th^{232} as the fertile constituent.
- There is an urgent need for nuclear incineration of higher actinides, for which thorium provides the best matrix (for deep burning of plutonium).
- Achieving a fuel burn-up much in excess of 50,000 MWd/t has now become possible with new cladding materials and better controllability of excess reactivity.
- Issue of the management of accumulated radioactive waste has gained prominence in countries operating large nuclear capacity for a long time.
- Thorium offers some distinct advantages in designing fuel, which cannot be diverted towards the weapon programme.

Coming back to the original point of long-term sustainability of nuclear power, it is clear that the thorium fuel cycle can display all its advantages only when one operates U^{233}–Th^{232} as the fissile–fertile mix. To achieve this goal, it is essential to generate enough of U^{233}, which again requires not only conversion of Th^{232} to U^{233}, but also the adoption of a closed fuel cycle involving reprocessing. The various options, which appear technologically feasible for generating power from thorium and the issues connected with various schemes, are discussed here.

Neutronic characteristics

A comparison of the nuclear properties of the three fissile nuclides U^{233}, U^{235} and Pu^{239} reveals the following: The thermal capture cross-section of U^{233} is much smaller than U^{235} and Pu^{239}, but the fission cross-section is of the same order[4].

The quantity $\eta(E)$ is defined as the average number of neutrons generated per neutron of energy E absorbed in the fuel. The variation of $\eta(E)$ as a function of neutron energy for all the three fissile nuclei, namely, U^{233}, U^{235} and Pu^{239} is shown in Figure 1. It is clear from this figure that U^{233} fission is most efficient in generating neutrons in the thermal and epithermal energy range while in the fast spectrum the $\eta(E)$ for Pu^{239} is maximum. While one neutron is required for carrying forward the fission chain reaction, the surplus neutrons, which escape capture and leakage, can produce excess fissile nuclides from fertile material. If the conversion of fertile to fissile more than compensates the loss of fissile nuclides, more fissile material is produced than consumed. This process is called breeding. From the consideration of breeding, Pu^{239} is potentially a better fuel in the fast neutron spectrum (because of high value of η). U^{233} can as well breed fissile material in a very wide neutron energy range, and is most effective in breeding in thermal and epithermal energies. Getting the η value as high as possible and reducing the capture and leakage are, therefore, the aim of reactor designers for achieving faster growth of the fissile inventory. While breeding is possible in fast spectrum by all the three fissile nuclides, breeding in thermal reactors can be achieved only with the Th^{232}–U^{233} cycle.

Breeding ratio (BR) defined as the ratio of the rate at which fissile material produced to that consumed, is smaller in thermal reactors fuelled with U^{233} compared to fast reactors fuelled with Pu^{239}. The period in which the excess fissile produced equals the initial inventory is defined as doubling time which is inversely proportional to breeding gain (BR-1) and directly proportional to specific fissile inventory (M/P kg/MWt, where M is the initial fissile inventory and P is the reactor thermal power). Thermal reactors require smaller specific fissile inventory compared to that of fast reactors. In spite of a smaller breeding gain of Th–U^{233} fuelled thermal reactors, the doubling time can come close to that of the fast reactor.

Thorium-232 has three times higher thermal capture cross-section than that of the other important fertile nuclide, U^{238}. While this results in a larger fissile requirement to achieve criticality, conversion of Th^{232} into U^{233} will be higher. Larger thermal capture cross-section of thorium also leads to lower losses due to parasitic captures. Nuclide chains originating from Th^{232} and U^{238} are

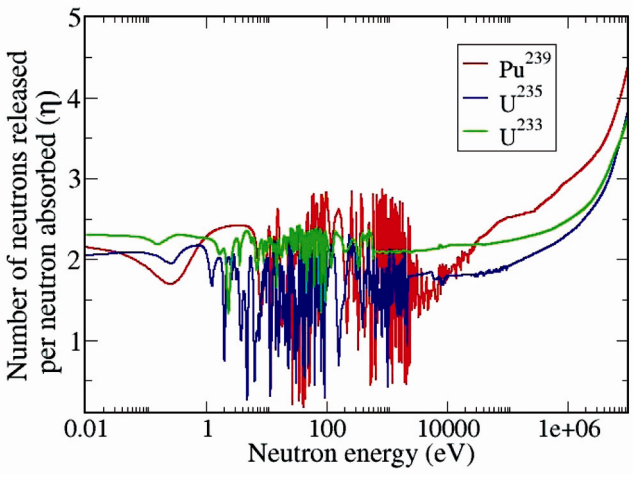

Figure 1. Variation of η with neutron energy.

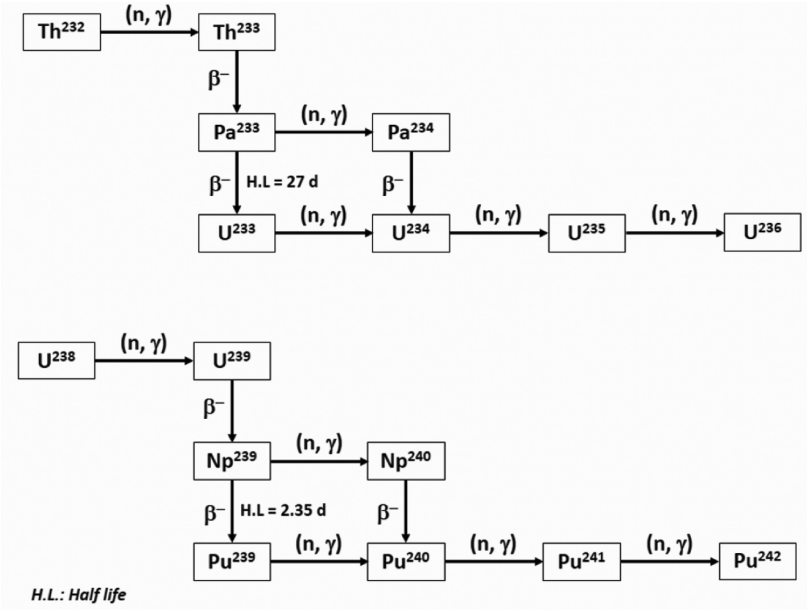

Figure 2. Nuclear reactions for conversion from fertile to fissile material.

shown in Figure 2. The generation of Pa^{233} in the thorium cycle, which has a higher half-life (~27 days) compared to Np^{239} (~2.35 days) in the uranium cycle and a high absorption cross-section for thermal neutrons, is responsible for the reduction in the breeding ratio.

Neutrons are an important resource to produce more and more fissile material from fertile materials available in abundance. It was recognized quite early in the development of nuclear energy that sustainability of nuclear power can be achieved only if neutrons are efficiently used in fission and conversion. For this purpose neutron economy should be given utmost priority while designing reactor systems and related fuel cycles. One could also consider the possibility of non-fissile neutron generation such as the spallation reaction (induced by bombardment of heavy nuclides with high energy (1–2 GeV) protons), which generates about 20–40 neutrons from a single event. Success of such an accelerator-driven technology on commercial scale has the potential to provide additional neutrons for rapid growth in the fissile inventory from the large reserves of fertile nuclides, U^{238} and Th^{232}, which can make nuclear energy sustainable for centuries to come.

Technological difficulties in utilization of thorium

Uranium/plutonium fuel cycles applied to thermal reactors is a proven and robust technology which is delivering power for over five decades. Further developments in fast reactors and in the back end of the fuel cycle are in the offing. On the other hand, the Th–U^{233} cycle is yet to be deployed and there are certain technology issues, which need to be resolved. The thorium-based fuel cycle is associated with the generation of U^{232} (half life ~68.9 years) by the several nuclear reactions, important among those are as follows

$$U^{233} + n^1 \rightarrow U^{232} + 2n^1,$$

$$Pa^{231} + n^1 \rightarrow Pa^{232} \xrightarrow{\beta^-} U^{232}.$$

Daughter products of U^{232} have short half-lives and two of these, Bi^{212} (half-life ~60.5 min) and Tl^{208} (half-life ~3.05 min), emit strong gamma rays. Therefore, U^{233} related activities such as fuel fabrication and subsequent fuel handling need shielding and remote access. The Thorex process for reprocessing of spent thorium fuel, which has been successfully demonstrated in somewhat smaller scale, needs to be deployed on a large scale for establishing a robust technology. The recent success of reprocessing of ThO_2 bundles irradiated in power reactors in kilogram scale has provided confidence to Indian scientists to go forward in the reprocessing of spent thorium fuel and subsequent fuel fabrication[5].

Radiotoxicity of spent fuel

Spent nuclear fuel in uranium fuel cycle consists of fission products, uranium and transuranic elements (TRU) such as neptunium (Np), plutonium (Pu), americium (Am) and curium (Cm). The composition of TRU inventory depends on factors such as initial enrichment of fuel, neutron energy spectrum, neutron flux and fuel burn-up. Spent fuel in typical light water reactor (LWR) contains approximately 95% uranium, 4% fission products and 1% TRUs. Radio-toxicity of spent fuel is dominated by

fission products for the first 100 years, subsequently, by TRUs mainly plutonium which is more than 90% of the TRU elements. Some of these TRU isotopes take nearly 200,000 years to achieve the radio-toxicity level of natural uranium ore. The variation of radio-toxicity as a function of time is shown[6] in Figure 3. The scheme of removal of these TRUs during reprocessing and transmuting them in nuclear reactors (fast reactors dedicated for burning long-lived radioisotopes) will enable reduction of radio-toxicity to the desired level to about 300 years, thus reducing the radioactive waste burden significantly.

In uranium cycle, U^{238} is almost entirely responsible for the production of TRU elements. In Th–U^{233} cycle, the production of long-lived TRUs is reduced by orders of magnitude to an insignificant level compared to those produced in uranium fuel cycle as shown in Figure 3. This is by virtue of thorium being lower in atomic number than uranium. Since thorium does not contain any fissile material, operating a reactor on Th^{232}–U^{233} fuel cycle will necessarily involve use of U^{235}/Pu^{239} in the initial stages of operation, which will produce some long-lived higher actinides.

Fuel utilization

In view of the fact that the availability of the fissile material is abysmally small compared to combined fertile (U^{238} + Th^{232}) material, the closed fuel cycle concept was introduced right at the inception of the nuclear energy programme. Alternative fuel cycles were conceptualized to convert fertile to fissile as efficiently as possible in order to achieve a near sustainability of nuclear power. In once through fuel cycle fuel utilization is enhanced by increasing the in-situ fertile to fissile conversion and subsequent efficient burning of the fissile content. In contrast in the closed fuel cycle the approach is to breed fissile material in excess of what is consumed, separate the fissile content by reprocessing and burn it in subsequent cycles. Multiple recycling options in fast breeder reactors are feasible due to the breeding capability and the higher tolerance of neutron poisons in fast neutron spectrum. Thus any reactor, thermal or fast, which has breeding ratio greater than one with reprocessing option, will have high fuel utilization. Considering the generation of higher actinides, which limit the ultimate fuel utilization it is estimated that fast reactors deploying the multiple U–Pu recycle can burn nearly 60% of uranium (U^{235} + Pu generates from U^{238}) atoms and thus supply energy for years to come. The introduction of thorium adds further to the supply of fertile nuclides and allows many fold extension to the supply of fissile nuclides by fertile–fissile conversion. The early interest on thorium is essentially based on this possibility.

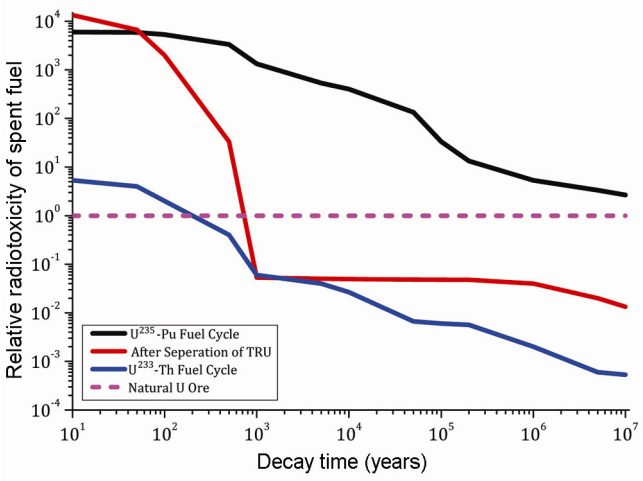

Figure 3. Relative radiotoxicity of nuclear waste in different fuel cycles as function of time.

At present, after extensive deployment of fission energy over half a century in the once through uranium fuel cycle, a substantial amount of fissile inventory of plutonium has been generated in the spent fuel in a few countries. The interest in burning these fissile nuclides with and without resorting to reprocessing of spent fuel has prompted the conceptual development of some new reactor systems and fuel cycle strategies.

Historical developments for utilization of thorium

There has been interest in the use of thorium in nuclear reactors from the beginning due to high η of U^{233} in thermal and epithermal energy range. Several countries have used thorium in conjunction with other fissile nuclides as nuclear fuel to gain experience on their irradiation behaviour. High temperature gas cooled reactors (TGR) in Germany and USA, BORAX-IV and Elk river boiling water reactors (BWR) in USA, KAMINI reactor in India, and Shippingport reactor in USA are some of the examples. The only thermal breeder reactor which was operated, was a pressurized light water reactor named Shippingport reactor in USA. This thermal breeder reactor operated during 1977 to 1982, produced 2.5 billion kWh of electrical energy. The non-destructive assay of 524 spent fuel pins and destructive analysis of 17 spent fuel pins showed that 1.39% more fissile material was present in the spent fuel compared to that present in the initial fuel[7], establishing that breeding had indeed occurred in the thermal energy spectrum.

W. B. Lewis in 1968 gave the concept of 'Valubreeder' to use thorium in pressurized heavy water reactors (PHWRs)[8]. In this concept, the initial fuel considered is thorium with slightly enriched uranium (nearly 1.8%) as the driver fuel. Enriched uranium provides the extra neutrons required for *in situ* breeding of U^{233} from thorium. The discharge burn-up for enriched uranium was considered at 20,000 MWd/t, and that of thorium was 35,000 MWd/t. The high burn-up of both these fuels keep

the cost of fabrication and reprocessing low. It was shown that the net fissile fuel credit could exceed basic inventory charges and thus economic feasibility was indicated.

The molten salt reactor concept was introduced in the aircraft reactor experiment (ARE) in 1954 with an operating power of 2 MWt. Alvin Weinberg at Oak Ridge National Laboratory (ORNL) developed 8 MWt molten salt reactor[9]. The fuel was in the molten salt form (Li^7F, BeF_2, ThF_4, UF_4), which was circulated through the heat exchangers and reprocessed online. Molten salt reactor experiment (MSRE) was conducted in 1964. All the three fissile fuels namely U^{233}, U^{235} and Pu^{239} were tested. ORNL designed 1000 MWe molten salt breeder reactors (MSBRs), using single fluid (with mixed fertile and fissile) and two fluids (separate fertile and fissile) concepts using thorium and calculated breeding ratios of 1.04 and 1.07 respectively.

In 1950, E. O. Lawrence proposed the idea of producing plutonium from depleted uranium using high power accelerators. In 1952, W.B. Lewis proposed the idea of producing U^{233} from thorium using a intense neutron source generator[10]. In 1993, Carlo Rubbia proposed thermal neutron energy amplifier system based on the thorium cycle[11].

Future with thorium

Having demonstrated about 15% of the total world electricity production by fission energy for many decades and consistently attaining capacity factors exceeding 80%, nuclear fission energy has indeed become a strong candidate for clean energy production in the world for many centuries to come. The concerns of rapid growth in nuclear energy arises mainly from factors such as long-term radioactive waste burden, diversion of fissile material for weapon production and availability of fissile material for an extended period. How adoption of the thorium fuel cycle addresses the above mentioned issues will now be discussed.

The various ways in which thorium can be deployed along with some driver fissile nuclides such as enriched uranium or plutonium can be broadly categorized as follows:

- Thorium as fuel in solid fuelled conventional reactors.
- Thorium as fuel in molten salt reactors.
- Accelerator-driven subcritical reactors using thorium fuel.

Use of thorium in solid fuelled conventional reactors

Thorium in heavy water reactors

PHWRs have excellent neutron economy due to use of heavy water as the moderator and coolant. These reactors adopt the pressure tube concept where hot pressurized heavy water is used for removing heat from the fuel pins put in fuel bundles, which are kept in pressure tubes. The design of small fuel bundles stacked in horizontal channels permit on-power fuelling with the help of fuelling machines. It is because of this design that the transition of one type of fuel to another can be done fairly easily in this type of reactor. These features make PHWRs suitable for use of different advanced fuels such as (U–Pu) mixed oxide (MOX) and ($Th–U^{233}$) MOX, etc. Since the reactivity margins (necessary for maintaining criticality after compensating leakage and parasitic absorption of neutrons) available in PHWRs are limited, a significant introduction of thorium in the reactor core will necessarily require addition of reactivity either by introduction of enriched uranium or plutonium as driver fuel.

There are three ways in which thorium can be introduced in these reactors[12]: (i) by fissile material (enriched uranium or plutonium oxide) and fertile material (thorium oxide) homogeneously mixed and put in all the fuel pins of the fuel bundle uniformly. (ii) Using two types of fuel pins, one containing fissile enriched uranium or plutonium oxide and the other containing fertile thorium oxide placed in a suitable configuration in fuel bundles (Figure 4 *a*). (iii) Using two types of fuel bundles, one having all ThO_2 and the other containing enriched UO_2 or (U–Pu) MOX pins distributed in separate fuel channels (Figure 4 *b*). In an alternative configuration, these different types of bundles can be staggered in a given fuel channel also (Figure 4 *c*).

Primary incentives of introduction of thorium in PHWRs are as follows. (i) Gradual build-up of inventory of U^{233}. (ii) Saving of natural uranium fuel. In order to estimate how much saving of natural uranium is possible by introduction of thorium in different ways, the lattice multiplication factor, k_∞, and the quantity of fissile nuclides present as functions of burn-up are calculated using the computer code CLUB[13]. The 69 group WIMS cross-section library[14] is used. The average $\bar{K}_\infty$, defined as

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

is maintained at a value, $\bar{K}_\infty = 1.045$, to account for a typical leakage value ~45 mk for PHWRs. As the fuel burn-up proceeds, the initial fissile inventory from the driver fuel is consumed with simultaneous fertile to fissile conversion. Increasing burn-up leads to accumulation of fission products eventually reducing the reactivity to a level where $\bar{K}_\infty$ drops below a threshold value (1.045 in PHWR) where the system no longer remains critical. At that stage, fuel needs to be discharged and the corresponding burn-up is called discharge burn-up. All reactors pass through three stages namely, fresh start-up,

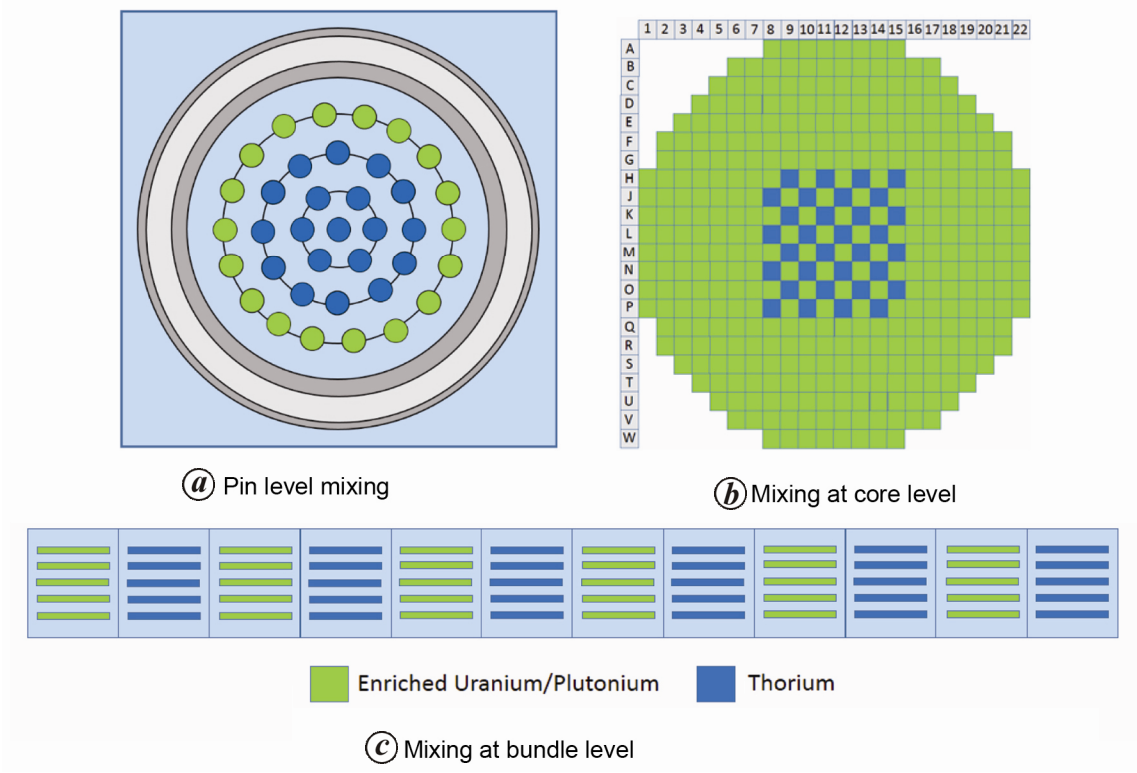

Figure 4. ***a***, Distribution of fissile and fertile material in different fuel pins of fuel bundle. ***b***, Distribution of fissile and fertile material in different channels in core (cross-section of core). Each square box represents a fuel channel with associated moderator. ***c***, Distribution of fissile and fertile material in different fuel bundles of the same channel.

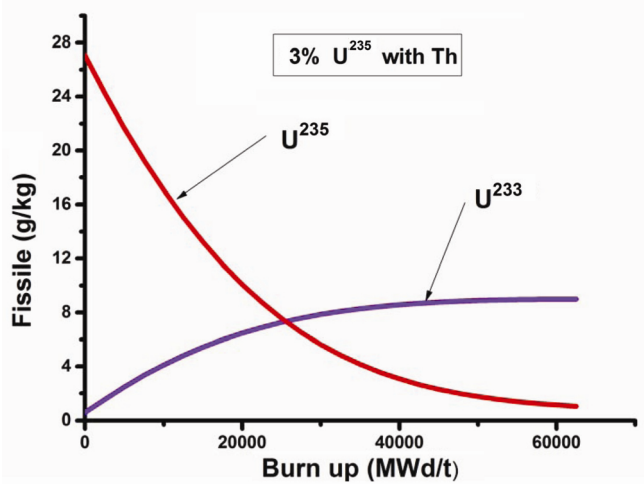

Figure 5. Variation of fissile material (g/kg of heavy metal) versus burn-up (3% enriched uranium case).

pre-equilibrium condition and equilibrium condition. Equilibrium condition is achieved after some time of operation of the reactor at rated power when average discharge burn-up becomes constant. The reactors remain most of the time in equilibrium condition during their lifetime.

Results obtained for a 1000 MWe PHWR in which fuel is made up of 70 wt% ThO_2 homogeneously mixed with 30 wt% UO_2 (containing 10% U^{235}) in all the fuel pins (case 1) shows that a discharge burn-up (B_d) of about 60,000 MWd/t can be achieved. Figure 5 shows how the contents of the fissile nuclides change in the core as the fuel burn-up progresses. While in the initial stage, fissioning of U^{235} is primarily responsible for energy production, the accompanying process of fertile-fissile conversion (Th^{232} to U^{233}) leads to the growth of U^{233} nuclides. Beyond the point of cross-over, U^{233} becomes the major contributor for energy production. The build-up of the other fissile nuclide, Pu^{239}, is however much reduced compared to a core containing all natural uranium for the obvious reason of reduced U^{238} in the starting fuel.

The annual requirement[15] of natural uranium for a 1000 MWe PHWR with all natural uranium core is about 170 tonnes. By using 70 wt% ThO_2 homogeneously mixed with 30 wt% UO_2 (containing 10% U^{235}) fuel, the annual natural uranium requirement can be decreased to 120 tonnes. Thus, there can be a saving of about 50 tonnes of natural uranium and generation of 200 kg of U^{233} per GWe annually. This way, one can derive the advantage of thorium conversion and *in situ* fissioning of U^{233} to extend the nuclear installed capacity by about 30% without resorting to reprocessing.

If a closed fuel cycle is deployed and uranium isotopes recovered from reprocessed spent fuel are used with topping up of enriched uranium or plutonium, a much larger

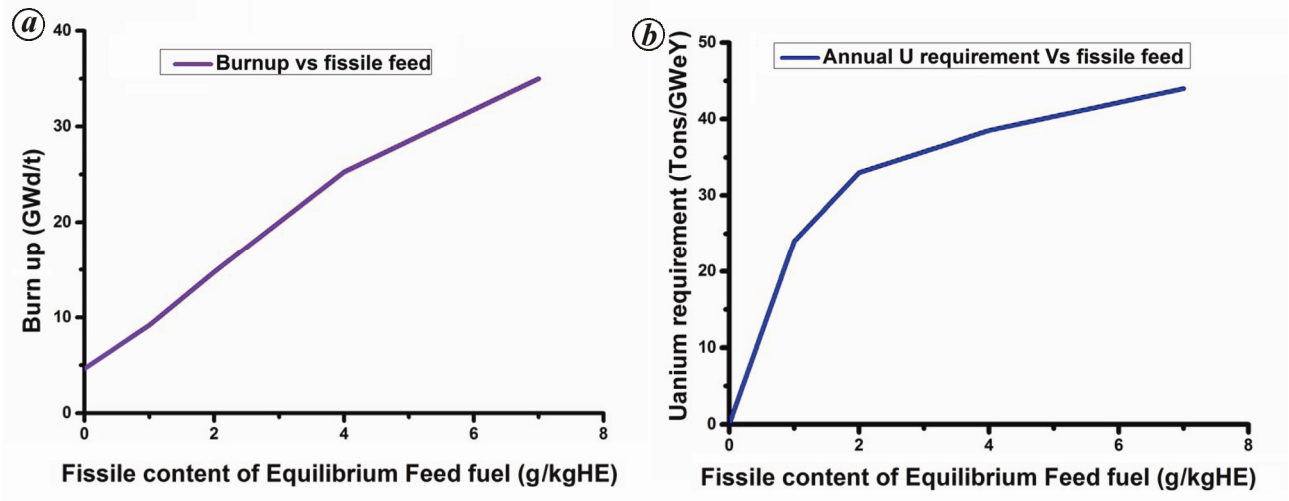

Figure 6. ***a***, Variation of fuel burn-up versus fissile feed (g/kg of heavy metal) (reprocessed case). ***b***, Variation of uranium requirement versus fissile feed (g/kg of heavy metal) (reprocessed case).

Table 1. Fissile topping amount, burn-up and natural uranium requirement

Fissile topping amount g/kg of heavy metal	Burn-up MWd/t	Natural uranium requirement (tonnes)
7	35,000	44.0
4	25,260	38.5
2	14,730	33.0
1	9200	23.8

saving of natural uranium can be achieved. This point can be illustrated considering a fuel containing ThO_2 and varying amounts of highly enriched UO_2. The first cycle is so chosen that the natural uranium requirement is nearly the same as in standard PHWR. With 88 wt% ThO_2 homogeneously mixed with 12 wt% UO_2 (containing 20% U^{235}) fuel, and the $\bar{K}_\infty$ attaining a value of 1.045 at the time of fuel discharge, the first cycle, attains a burn-up[16] of 32,150 MWd/t. After the first cycle, the uranium is recovered from the discharged fuel. This reprocessed fuel is topped up with varying amounts of highly enriched fuel ranging from 7 to 1 g/kg of heavy metal (hm) and is used in the next cycle. Table 1 shows fissile material topping amount in g/kg, burn-up achieved and natural uranium requirements (in tonnes). The variation of fuel burn-up and the annual requirement of natural uranium as a function of fissile topping expressed in g/kg is shown in Figure 6 *a* and *b* respectively. It can be seen that as the external fissile feed is reduced, the annual requirement of natural uranium decreases. It is to be noted, however, that while the lower fissile feed requires very small amount of uranium annually per GWe, the corresponding burn-up also reduces. This means that as the natural uranium requirement is reduced, the number of reprocessing cycle will increase and beyond a limit, this may not be economically attractive.

The second thorium cycle analysed[4], is the one in which 1.3% enriched uranium oxide is put in the fuel pins of the outer two rings (i.e. 30 fuel pins) of the fuel bundle and thorium oxide is put in the fuel pins of inner two rings (i.e. 7 fuel pins). As thorium does not have any fissile material, all the power in the beginning is produced in the outer 30 fuel pins. This applies restriction on the number of thorium pins that can be put in the bundle. Also, since thorium has high neutron absorption cross-section, discharge burn-up in the first cycle is small. The discharge burn-up progressively increases from 13,800 to 22,500 MWd/t in five cycles as production of U^{233} increases with each cycle. While the uranium pins in the outer two rings have to be replaced after each cycle, which is a difficult task, thorium pins are replaced only after five cycles. Average natural uranium consumption at the end of five cycles is nearly 112 tonnes/GWe year, which is 35% lower than that of the Nat-U cycle. After five cycles, 225 kg/GWe of U^{233} remains in the spent fuel. This fissile material can be recovered by reprocessing the spent fuel, the reprocessed U^{233} containing nearly a few hundreds of ppm of U^{232} making it unsuitable for weapon use.

The next case is where two types of fuel bundles, one having all ThO_2 and the other containing enriched UO_2 or (U–Pu) MOX pins are distributed in separate fuel channels. This case corresponds to what has been done in some of the Indian PHWRs for achieving neutron flux flattening which is essential during the initial fuel loading of PHWR in order to achieve full power from the start of the reactor. Either depleted uranium or thorium bundles can be used for this purpose in PHWRs. In India, reactor core with 35 thoria bundles was configured in certain reactor units for the purpose of flux flattening during the initial fuel loading. Locations of these thorium bundles in the core[17] were selected such that full power can be

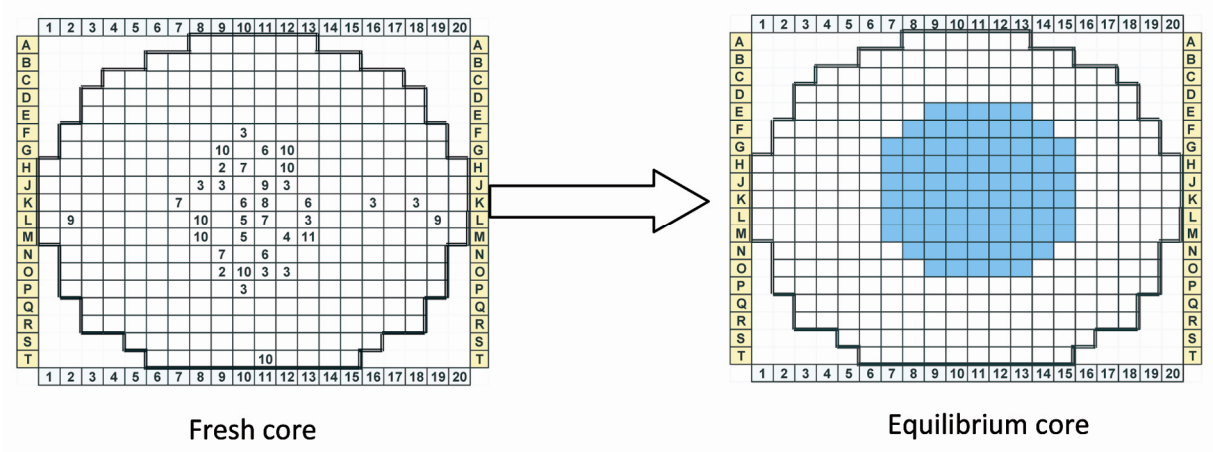

Figure 7. Typical fresh (with thorium bundles, their positions being indicated by the number) and equilibrium core (central region having high burn-up fuel compared to outer region) of 220 MWe PHWR.

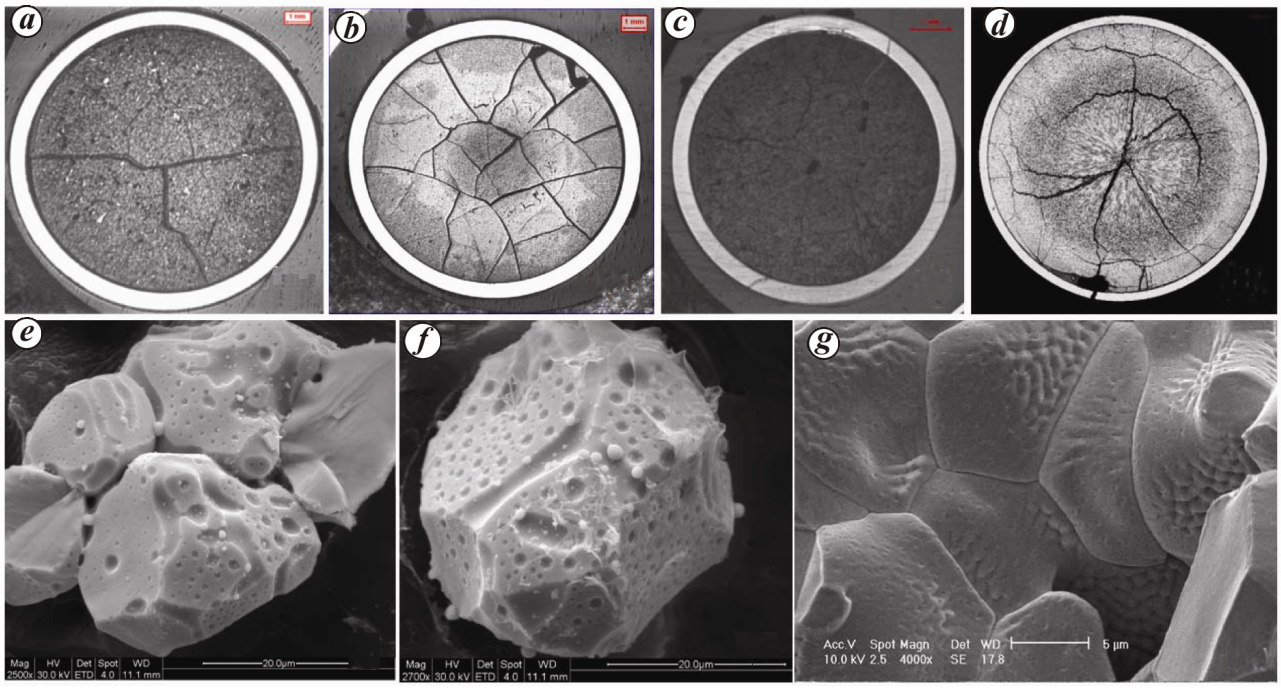

Figure 8. Microstructure of cross section of irradiated fuel pins: ***a***, ThO_2 – 4%PuO_2; ***b***, UO_2 – 4%PuO_2; ***c***, ThO_2 – 4%PuO_2 fuel showing clad failure; ***d***, UO_2 fuel showing clad failure. ***e***, ***f***, High number density fission gas bubbles and channels on fractured grain surface of UO_2 fuel at 4400 and 15000 MWd/t burn-up levels. ***g***, Lower number density of fission gas bubbles on fuel grain faces of ThO_2 + 4%PuO_2 fuel with burn-up of 18,400 MWd/t.

achieved without reducing the reactivity worth of primary and secondary shutdown systems. Figure 7 shows the typical fresh core of 220 MWe PHWRs containing thorium bundles and equilibrium core containing highly exposed fuel bundles in the central region of core. Although in this configuration a limited amount of thorium is used, it has generated considerable amount of data on the irradiation behaviour of thoria fuel. Post irradiation examination (PIE) of some of the irradiated fuel bundles[18] in research and power reactors has revealed that ThO_2 – 4% PuO_2 fuel is more tolerant to clad failure than UO_2 – 4% PuO_2 fuel (Figure 8 *a–d*). Microstructural examination and fission gas release data of irradiated fuel have shown that ThO_2-based fuel has a higher capacity of retention of fission gases within its lattice (Figure 8 *e–g*). This is not unexpected as diffusion of fission products is significantly slower in ThO_2 matrix.

The aforementioned saving of uranium is in the equilibrium condition. Therefore the net saving of uranium will depend upon the time taken for reaching the equilibrium condition relative to the life time of the reactor. Since the uranium requirement is more for thorium cycle

compared to that of uranium cycle to reach the equilibrium condition from fresh reactor startup, the saving of the overall uranium requirement will be lower than 30% as mentioned earlier.

Let us consider the case where plutonium is used as the driver fuel along with thorium to operate a PHWR applicable to situations where plutonium (both reactor and weapon grades) inventory needs to be deeply burnt. For example, if 3% reprocessed U^{233} is put with thorium oxide in the inner 19 fuel pins of the fuel bundle and outer 18 fuel pins are made up of thorium oxide with 1.5 wt% of plutonium oxide, discharge burn-up of the order of 21,900 MWd/t can be achieved. This cycle will require annually nearly 416 kg of plutonium and 856 kg U^{233} per Gwe. The annual discharge fuel will contain nearly 916 kg of U^{233}/Gwe while plutonium will be burnt out nearly completely. Figure 9 shows that as far as U^{233} is concerned, there is a slight increase in the fissile inventory (916 kg at the discharge level compared to 856 kg initially), while Pu^{239} is nearly completely burnt out at a rate of 416 kg annually.

Thorium utilization using the existing PHWRs will also need the following changes to be considered:

- In order to extract energy from thorium, it is necessary to allow conversion of thorium into fissile U^{233} *in situ* and this requires a relatively high burn-up (typically about 50,000 MWd/t). Therefore, the fuel cladding should be able to withstand such burn-up. A thicker cladding or using a new cladding material in the form of free-standing cladding capable of withstanding higher burn-up needs to be adopted.
- Power peaking should be within specified limits. Otherwise, the reactor has to be de-rated. Fuel bundles with different fuel pin radii can alleviate this problem to a great extent. Fuel pins in the outer most rings of the bundles near the moderator can have smaller radius.

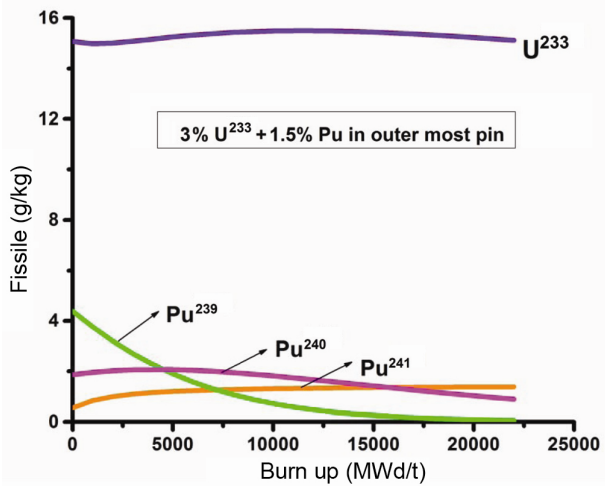

Figure 9. Variation of fissile material (g/kg of heavy metal) versus burn up (Pu burner case).

- The worth of control and shut off rods should be sufficient to control and to safe shutdown the reactor. Gadolinium mixed with fuel in few pins in a bundle and/or addition of poison in the moderator can enhance the control capability.
- Channel power peaking with high reactive bundles will be more compared to natural uranium fuel bundles. This will reduce the margin to dry out. To reduce its effect, two-bundle shift may be required instead of eight-bundle shift required for natural uranium fuel.

Such requirements have also been taken care of by design of the advanced heavy water reactor (AHWR) as discussed later.

Advanced heavy water reactor

The excellent neutron economy and the requirement of online fuelling due to limited excess reactivity are important features of PHWRs. Safety and operational features of PHWRs can be further improved by the following:

- Coolant void coefficient of reactivity can be made negative.
- Refuelling interval can be increased to the advantage of the operators.
- Reduction in heavy water loss and background tritium activity by changing over from heavy water to light water coolant in the high-pressure circuit.
- Introduction of passive safety features.

All these have been achieved in the Indian design of AHWR[19–23] which is a heavy water moderated, vertical pressure tube type reactor with natural circulation of boiling light water coolant. It is designed to produce 920 MWt (300 MWe) power. The reactor is designed for 100 years life. Heat from the fuel rods is removed by natural circulation of boiling coolant. The coolant void reactivity and other reactivity coefficients are negative. The neutron spectrum of this reactor is harder than that of PHWRs. Many passive safety features have been introduced. AHWR can be configured to accept a range of fuel types including enriched U, (U–Pu) MOX, (Th–Pu) MOX, and (Th–U^{233}) MOX in full core. In the Indian context, when a sufficient U^{233} inventory is accumulated, it is intended that AHWR will run with (Th–U^{233}) MOX.

In the start-up core, all the fuel pins of the cluster will contain (Th–Pu/enriched U) MOX. Gadolinium is introduced in some fuel pins of the fuel cluster to control the initial excess reactivity. The discharged fuel clusters will be reprocessed to recover U^{233} and plutonium separately. Refabricated fuel cluster will contain (Th–Pu/enriched U) MOX in the outer fuel pins and (Th–U^{233}) MOX in the inner fuel pins. This cluster will keep replacing the discharged fuel clusters from the core at appropriate locations. This process will be repeated and finally the

Table 2. Comparison of different cases

	OTC – All natural uranium	OTC – 3% enriched uranium + Th	Recycled U + Th + external fissile feed 4 g/kg	AHWR –LEU	PWR
Burn-up (MWd/t)	7200	60,000	25,000	60,000	40,000
Annual natural uranium requirement (tonnes) per Gwe	170	120	39	155	220
Fuel to be reprocessed tonnes/year	170	20.3	48	18.4	27.6
Annual fissile material in discharge fuel (kg) per Gwe	600 Plutonium	210 U^{233}		210 U^{233}	330 U^{233}

OTC, Once through cycle; Thermodynamic efficiency, AHWR and PWR 33%, For PHWR: 30%; Tail enrichment 0.2%; Load factor: 100%.

core will reach the equilibrium condition after certain years.

One of the recent designs of AHWR is based on mix oxide (MOX) fuel of low enriched uranium (LEU) with thorium. The equilibrium core of AHWR–LEU will be loaded with Th–LEU fuel clusters. The first core of AHWR–LEU is designed to be loaded with Th–LEU fuel clusters with a low average LEU content. With refuelling time of nearly one year, it will take nine years to reach the equilibrium condition. The average discharge burn up is around 60000 MWd/t. Power produced by U^{233} is nearly 38% at the core average burn up. Annual discharge fuel contains 63 kg of U^{233} with significant amount of U^{232} which makes it proliferation resistance.

Use of thorium in LWRs

Thorium can be used in PWRs either by distributing thorium homogeneously in all fuel pins with enriched uranium or in a heterogeneous seed and blanket arrangement. In one of the homogeneous distribution schemes, 80 wt% thorium oxide and 20% enriched uranium oxide (containing 20% U^{235}) are mixed uniformly in all the fuel pins. For a 1000 MWe reactor, 1080 kg/GWe of U^{235} is needed annually[16], while the discharged fuel will contain 330, 120 and 70 kg of U^{233}, U^{235} and Pu^{239} respectively. Once through cycle leads to marginal saving of natural uranium at high enrichment (~10% fissile enrichment). However, with the closed fuel cycle utilizing reprocessed U^{233} leads to a natural uranium saving[24] of up to nearly 40% can be achieved.

Radkowsky[25] considered a heterogeneous seed-blanket concept using thorium in the once through fuel cycle. Metallic uranium–zirconium fuel in the seed and enriched uranium oxide along with thorium oxide in the blanket are considered. This configuration avoids power peaking problem in the beginning. The moderator to fuel ratios in seed and blanket are different which will require modification of the standard fuel assembly design. The enriched uranium in the seed has to be replaced more often (3 years) compared to the thorium in the outer blanket (10 years) so that more *in situ* U^{233} can be produced. In this scheme, 875 kg /GWe of U^{233} gets accumulated in the blanket after 10 cycles. The annual discharge rate of plutonium is 33 kg/Gwe.

To avoid any modification in the reactor design for using thorium, a case is considered where the moderator to fuel ratios in seed and blanket are same. Thorium pins act as blanket while all inner fuel pins containing 6% enriched uranium oxide act as seed. Entire core in this scheme is replaced in three cycles. The one third of seed has to be replaced by fresh fuel after each cycle while blanket will be removed after three cycles. U^{235} needed[16] 1150 kg/GWeY and U^{233} produced after 3 cycles 623 kg in the discharged fuel.

Table 2 shows the comparison of PHWR, AHWR and PWR fuel cycles with thorium. It can be seen that PHWRs with flexibility in fuel management have certain advantages over LWRs in deploying thorium. While making a comparison between different fuel cycle options, one needs to take into account all the points, namely, annual requirement of mined natural uranium, effect of enrichment and reprocessing, the quantum of fissile materials in the discharge fuel and the radio-toxicity burden (due to long-lived radioisotopes) in the waste.

Conclusions on deploying thorium in BWRs will be similar to those of PWRs.

Use of thorium in conventional fast breeder reactors

Very high fuel utilization can be achieved in fast reactors with fuel reprocessing depending upon fuel burn up and losses during reprocessing. By putting thorium in a blanket of Pu/U^{238}, fuelled fast reactors can produce U^{233} without much affecting the breeding ratio. The breeding ratio can come down by nearly 2% for oxide fuel[26,27]. However, putting thorium both in the core (Pu/Th) and blanket of these reactors significantly reduces the breeding ratio by nearly 17% for oxide fuel[26]. The breeding ratio of U^{233}/Th cycle in fast reactors is much smaller than that of Pu/U^{238} cycle. This is due to two reasons, namely (i) η of U^{233} is low compared to that of Pu^{239} in fast neutron spectrum and (ii) small fast fission probability in Th^{232} as compared to that in U^{238}. Therefore, it is preferable to use, thorium in the blankets of Pu/U^{238} cycle

to produce fissile U^{233} that can be used to start new reactors. The advantage of U^{233}/Th cycle is due to reduction in sodium void coefficient and in making the Doppler coefficient of reactivity more negative in U^{233}/Th cycle compared to Pu/U^{238} cycle.

Use of thorium in breed and burn reactors

Breed and burn (B&B) fast reactors are being designed to significantly increase the uranium utilization without fuel reprocessing[28]. Minimum required average burn-up for sustainability of B&B reactors is 19.4% fissions per initial metal atom (FIMA)[29]. These reactor cores need very hard neutron spectrum. Neutron-induced radiation damage of the clad material and internal stresses due to high fission gas pressure puts limitation on burn-up. Limited fuel 'reconditioning' is needed to remove the volatile fission products and replace the fuel clad before fuel is reused in the reactor. B&B reactors are fast reactors that are specially designed to breed plutonium from depleted uranium and fission significant fraction of bred plutonium without reprocessing. In contrast, in conventional fast reactor, seed and blanket fuels are reprocessed. Nearly, 15 to 20 cycles of reprocessing are needed for good fuel utilization. B&B reactors are based on the concept of a 'moving zone' of power production which provides excess neutrons to its neighbouring subcritical zones. As burn-up proceeds, the latter, mainly constituted of fertile material, gets converted into fissile and consequently gains reactivity. The power-producing zone gradually shifts towards the adjacent zone of freshly converted fissile material. This is illustrated in Figure 10 where the movement of the burning and breeding zones is schematically shown. From engineering considerations, such a design (travelling wave reactor)[30] poses difficulties in implementation. A similar concept is proposed in an alternative design[29] in which the burning zone is kept stationary which is fed with freshly converted fuel from the adjacent region as shown in Figure 11. As the highly burnt fuel is discharged from a part of the burning zone, they are replaced by freshly converted fuel, which has gained adequate reactivity. To start with, the fresh fertile material acts as neutron absorbers and gradually transforms into net neutron producers. This is illustrated in Figure 12 *a*. Figure 12 *a* and *b* shows the variation of excess neutrons and k'_∞ with burn-up respectively, within a cell which consists of homogenized depleted uranium fuel and associated coolant and structural materials. Excess neutrons[27] of a cell is the difference of total number of neutrons produced and absorbed, integrated over time. Between point A and B in Figure 12 *a*, the cell is net absorber of neutrons. Between point B and D, the cell is net producer of neutrons. After D, it becomes neutron absorber due to accumulation of fission products and decrease in the fissile content. Excess neutrons are related to k'_∞ of the cell. The point C indicates minimum burn up required in B&B reactors.

A B&B reactor can definitely breed more fissile material than it consumes and, therefore, can add to fissile inventory necessary for the capacity growth of nuclear power. The doubling time, however, will be higher than that can be obtained in conventional fast reactors coupled with the closed fuel cycle. The concept of B&B can be adopted in conventional fast reactors with closed fuel cycle for the purpose of reducing the number of reprocessing cycles. However, operational convenience and safety issues need to be considered in detail before taking this path.

The B&B reactor concept discussed so far is in general terms of a fissile/fertile combination. If enriched U/Pu is used as fuel and depleted uranium as fertile material, no fissile addition is required in equilibrium condition, after reaching a discharge burn-up of 19.4% FIMA. Owing to a lower η value for U^{233} compared to Pu^{239} and a lower value of σ_f for Th^{232} compared to U^{238}, it may be difficult

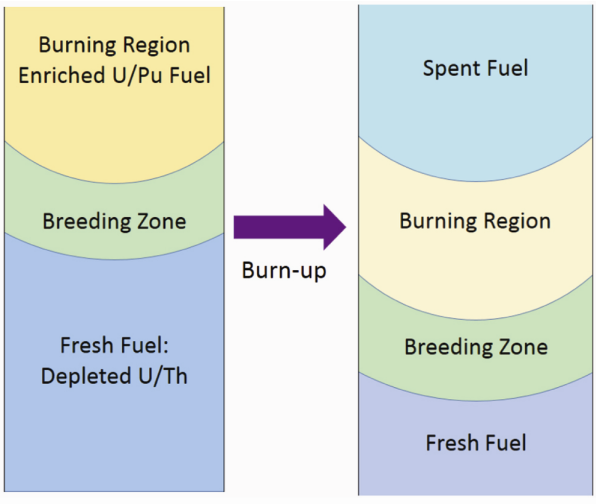

Figure 10. Schematics of candle reactor.

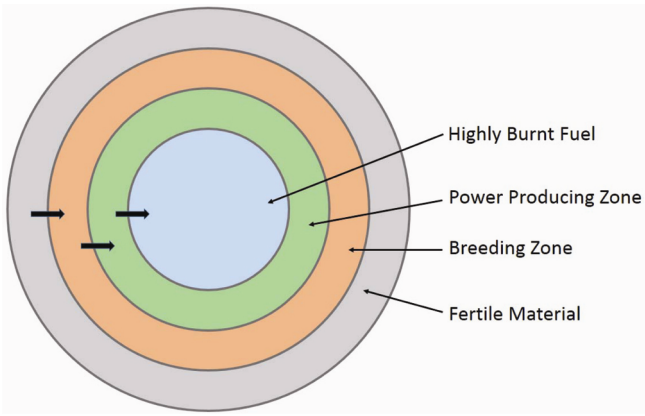

Figure 11. Schematics of standing wave reactor which shows regions corresponding to different burn-up levels. Fresh fuel will be loaded from the periphery and highly burnt fuel will be discharged from the central region periodically.

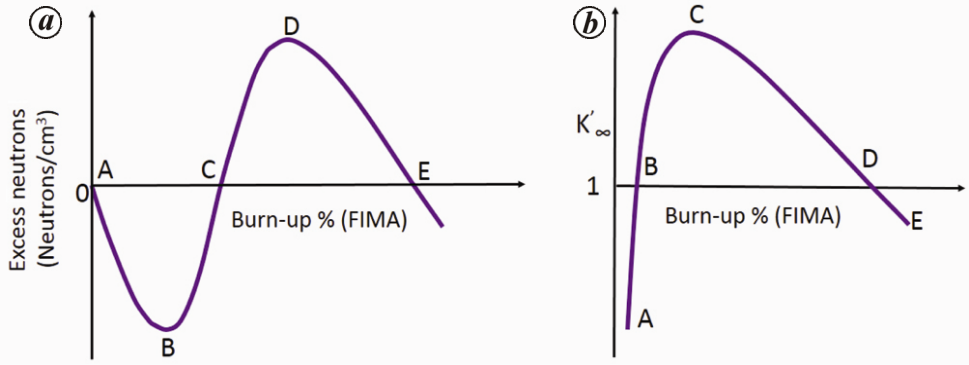

Figure 12. ***a***, Variation of excess neutrons with burn-up; ***b***, Variation of k_∞ with burn-up.

to achieve sustainability in U^{233}/Th cycle in a B&B reactor. However, with addition of fresh fissile material, B&B reactors can enhance fuel utilization significantly in U^{233}/Th cycle.

Use of thorium in MSBRs

The fuel in a molten salt reactor is in the form of molten salt of uranium tetra fluoride (UF_4) dissolved in the mixture of lithium fluoride (Li^7F), beryllium fluoride (BeF_2) and thorium fluoride (ThF_4). The melting point of this salt is nearly 500°C while the boiling point is 1400°C, thereby providing a large thermal safety margin. In this type of reactor, the fissile nuclides in the molten salt fuel undergoes fission in the core, which contains circulating molten salt and suitable moderators. Nuclear heat generated in the core is carried by the molten salt, which passes through the heat exchangers to transfer the heat to a secondary fluid.

Molten salt reactors have the distinct advantage of online removal of fission product poisons and addition of fresh fissile material. In addition, removal of protactinium from the core helps in reducing neutron loss. In solid fuelled reactor, it is not possible to remove the protactinium, which absorbs neutrons leading to a reduction in the fertile to fissile conversion. All these factors are responsible for making it possible for U^{233}/Th cycle to breed (breeding ratio exceeding unity) in thermal spectrum. U^{233}/Th cycle is therefore best suited for molten salt reactors, as far as breeding is concerned. It has been estimated that a breeding ratio of 1.05 can be achieved in molten salt reactors using U^{233}/Th cycle. The schematic of molten salt reactor is shown in Figure 13.

Most of the research on the development of these reactors took place at Oak Ridge National Laboratory (ORNL) in USA during 1954 to 1974. A 2 MWt ARE was conducted in 1954 while molten salt reactor experiment (MSRE) was operational during 1965 to 1969 at 8 MWt power. The operating experience was obtained in this experimental reactor using all the three fissile materials namely U^{235}, U^{233} and Pu^{239}.

As stated earlier, thermal reactors require smaller specific fissile inventory than fast reactors. This makes reactor doubling time achievable in molten salt thermal reactor closer to that of fast reactors even with smaller breeding gain. MSBR where BR can be of the order of 1.05 in thermal energy range therefore is an attractive option for fuel utilization considerations. ORNL had designed[31] two fluid cores for 1000 MWe (2250 MWt) where fuel salt consisted of LiF, BeF_2, $U^{233}F_4$ and fertile salt was LiF, BeF_2, ThF_4. Graphite was used as moderator. For this reactor, the breeding ratio was 1.07 and specific fissile inventory was 0.78 kg/MWe. This design was discarded due to the safety concern of introducing a large positive reactivity in case of loss of fertile salt and also because graphite cannot be relied upon as a sufficiently effective barrier to separate fertile and fissile fluids. ORNL later designed a single fluid[32] 1000 MWe (2250 MWt) reactor, where fuel salt was LiF, BeF_2, ThF_4 and UF_4 and graphite used as moderator. The breeding ratio was reduced to 1.04 while the specific fissile inventory increased to 1.0 kg/MWe. In Japan, studies were carried out for small reactors named as Fuji reactors[33] of 200 to 350 MWt with fissile salt as LiF, BeF_2, ThF_4 and PuF_3 (0.2 and 0.6 mol%) showing plutonium burning and U^{233} production. There was no online reprocessing considered in these Japanese studies, because of which the conversion ratio (CR) was around 0.93 for the U^{233}/Th cycle. In the recent design studies of MSBRs, zirconium hydride[34] is considered as the moderator material in place of graphite. MSBRs have many advantages compared to solid fuelled reactors. In these reactors, the fuel is in liquid state; therefore, fuel fabrication is not required which in particular is difficult for U^{233} fuels. Burn up limitations due to radiation damage of fuel and clad will not arise as fuel is in molten form. In solid fuelled reactor, the fuel is cooled for a long period of time after it is discharged from the reactor which is not needed in molten salt reactors. Online fuelling and reprocessing is possible. Thermal MSBRs possess many inherent safety features. It uses molten fuel; therefore fuel 'meltdown' is of no concern. The fuel is critical in the molten form in some

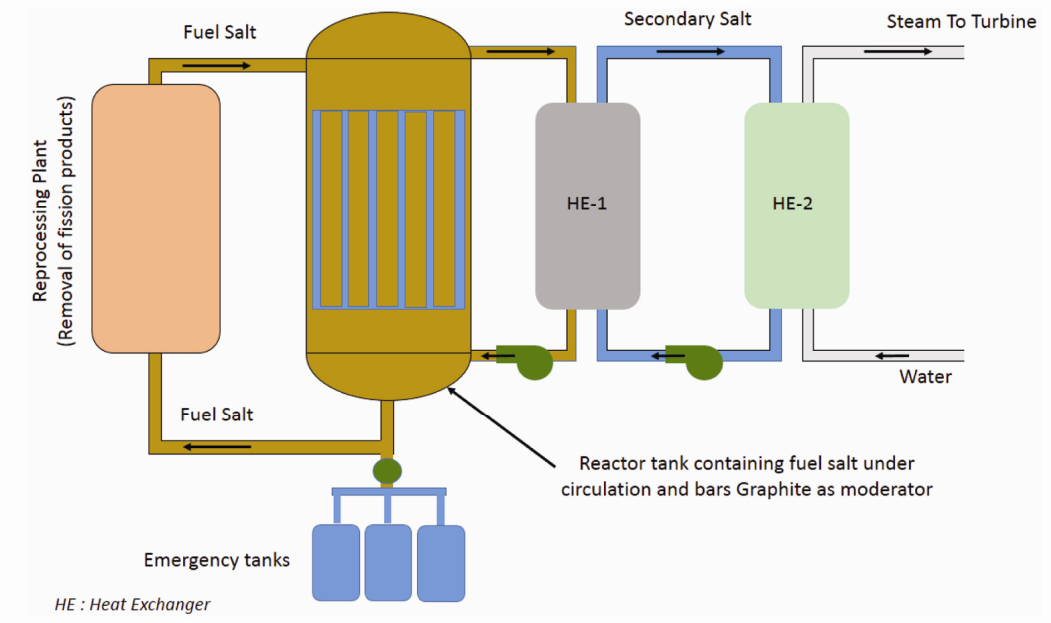

Figure 13. Schematics of molten salt reactor.

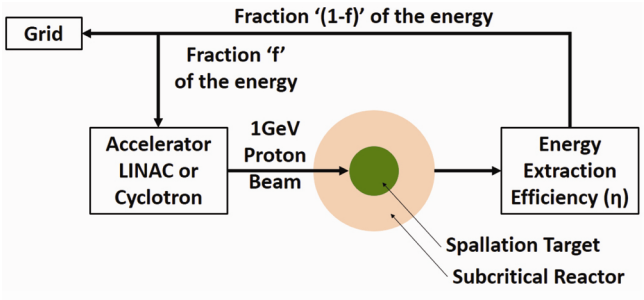

Figure 14. Schematics of accelerator driven sub-critical system.

optimal configuration in the core. Any escape from this configuration will not lead to re-criticality and it will become subcritical. Safety is enhanced due to large difference between melting point and boiling point (500°C and 1400°C) near ambient operating pressure, as there is small probability of dispersion of radioactivity. There is no requirement of high-pressure containment. High operating temperature also increases the thermodynamic efficiency. If due to some reasons, the fuel salt temperature increases, then the entire fuel salt can be dumped into the critically safe tanks. Online removal of poisons and fissile addition, results in gain in reactivity and neutrons while keeping source term for any accident low.

The MSBR needs further improvement with respect to the following[33,35].

- Dimensional changes in graphite takes place under irradiation; hence, it needs frequent replacement and adds to nuclear waste. Alternative materials are desirable. Zirconium hydride is proposed as a moderator in recent designs in place of graphite.
- The effective delayed neutron fraction β^{eff}, a safety-related parameter is low due to fuel circulation.
- Better structural materials need to be developed to hold molten fluorides to prevent radioactivity leaks.
- Radioactivity in primary circuit needs retention and cooling under all situations. Passive cooling will be desirable for this purpose.

Use of thorium in accelerator driven subcritical systems

In a critical reactor, neutrons produced by fission are exactly balanced by the number of neutrons lost by leakage and absorption. This balance maintains the reactor power at any desired level. In a sub-critical reactor, external supply of neutrons is needed to maintain a constant reactor power. Non-fissile neutrons generated through a spallation reaction can provide 20 to 40 neutrons per single event when a high-energy proton beam of 1 or 2 GeV coming from accelerator collides with a heavy atom nucleus such as lead[36]. Figure 14 shows the schematics of accelerator driven subcritical systems (ADSs).

Multiplication of neutrons, in subcritical reactor ($k_{\text{eff}} < 1$) where k_{eff} is the effective multiplication factor of the reactor, is given by

$$M = 1 + k_{\text{eff}} + k_{\text{eff}}^2 + \ldots = \frac{1}{1 - k_{\text{eff}}}.$$

Power in ADS is equal to proton beam power multiplied by gain $G = G_0/1 - k_{\text{eff}}$; where G_0 varies between 2.1 and 2.4. Thus, for $k_{\text{eff}} = 0.98$ and the beam power = 10 MW, the power in ADS will be 1050 MW (with $G_0 = 2.1$).

The gain increases as k_{eff} of the subcritical reactor increases, and therefore ADS power increases for the same beam power. In a nuclear reactor, the effective multiplication factor generally decreases in the high power operating conditions when the temperature of fuel, coolant and moderator and concentration of fission products such as xenon and samarium increases compared to that at low power. This drop in the effective multiplication factor is less in a fast reactor compared to that of a thermal reactor. Therefore, for the same k_{eff} at low power, subcriticality $(1 - k_{eff})$ is smaller in fast reactor compared to that of thermal reactor. Consequently, energy amplification is expected to be more in fast reactors for the same initial sub-criticality. Therefore, coupling of ADSs with fast reactors provides a better alternative to incinerate the TRUs. ADSs can also be used to utilize thorium along with small fissile materials such as uranium or plutonium oxides in subcritical thermal reactors (a PHWR or a molten salt reactor) by *in situ* generation of U^{233}. One-way coupled fast and thermal subcritical reactor concept with spallation neutron source in the centre is developed[37]. The inner core is a subcritical fast reactor with thermal neutron absorber liner surrounded by gap. The outer core is a subcritical thermal reactor; neutrons leaking from inner core can reach the outer zone where their number gets multiplied. Neutrons from thermal reactor cannot go to the inner core due to the absorber liner; that is why, it is one-way coupled. Figure 15 shows the schematics of one-way coupled system.

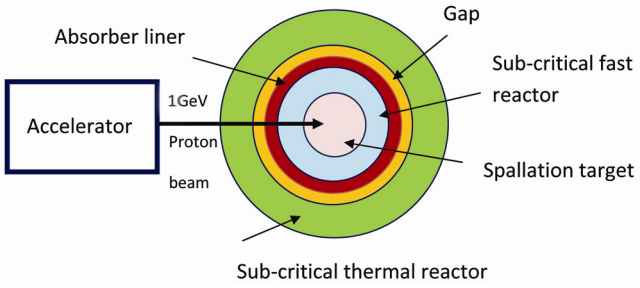

Figure 15. One-way coupled accelerator driven sub-critical system.

As a 'thought experiment', the lattice calculations (average k_∞) were carried out for a PHWR coupled with ADSs. Calculations indicated that a self-sustaining thorium cycle could be established in a subcritical PHWR. Thorium with 1.8% U^{235} MOX was put in outer two rings and ThO_2 was put in inner fuel pins in PHWR fuel bundle. The k_∞ is higher than 0.93 up to a burn-up of 57,500 MWd/t. Most of U^{235} is burnt out and U^{233} is produced as shown in Figure 16 *a*. Figure 16 *b* shows the variation of k_∞ with burn-up. The discharged fuel is reprocessed and loaded again uniformly in all the fuel pins of the fuel bundle. The k_∞ is higher than 0.95 up to burn-up of 52,500 MWd/t. Repeated recycling maintains the k_∞ and burn-up. No external fissile feed is needed except in the first cycle. All the remaining cycles will produce and burn U^{233} in a self-sustaining mode. Once neutrons from the spallation reaction can be supplied to a sub-critical core at an economically viable cost, thorium utilization can be achieved with only an initial supply of fissile nuclides.

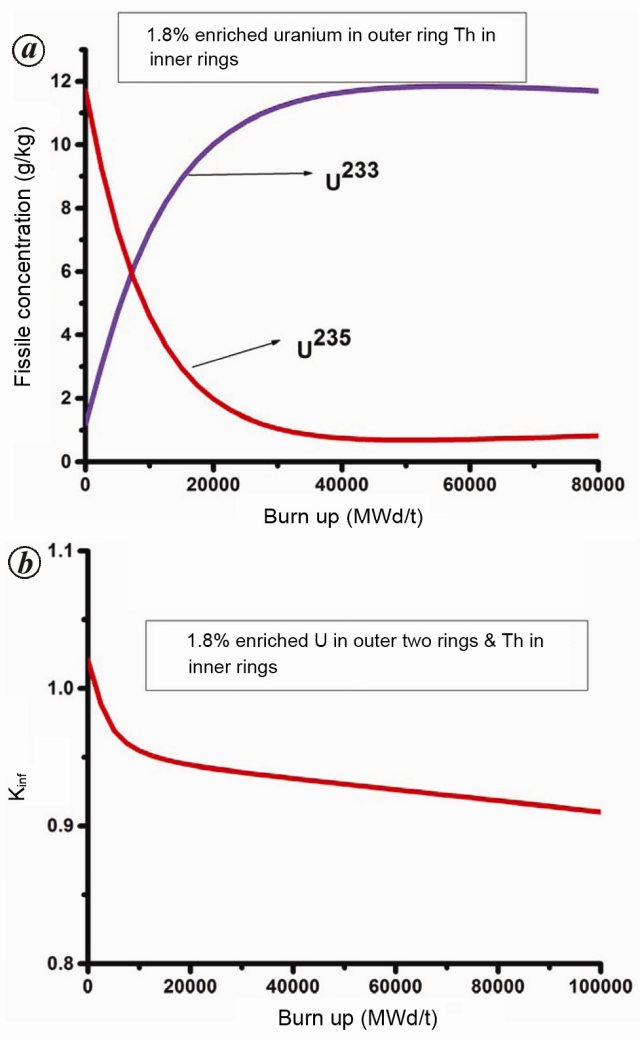

Figure 16. ***a***, variation of fissile concentration with burn-up; ***b***, Variation of k_∞ with burn-up.

Options under different scenarios

Various issues related to the generation of nuclear power from thorium have been discussed in the preceding sections. Based on these general observations, one can examine the possible ways energy from thorium can be tapped under different situations.

Scenario-A is where large inventory of ready to burn fissile material is available or accessible; present generation nuclear reactors are deployed for a large nuclear

generation capacity and there is no immediate need for a rapid growth of nuclear power generation capacity.

The ready to burn fissile material in this context means enriched uranium and/or plutonium. A large stock of these will necessarily imply a very large stock of depleted uranium as well. Continuation of the nuclear power programme at nearly the same capacity level without reprocessing the spent fuel will result in accumulation of transuranic actinides, which will remain a serious radiotoxic burden for over 200,000 years. Recoveries of fissionable isotopes, particularly Pu^{239} and burning them in nuclear reactors, will not only enhance supply of fissionable materials but will simultaneously reduce the radiotoxic life of nuclear waste. There are various options for burning plutonium in the present generation thermal and fast reactors and in B&B reactors with either depleted uranium or thorium as fertile matrix. Thorium offers a marked advantage over depleted uranium in terms of significantly reduced production of long-lived isotopes. Thorium oxide being a very stable compound can act as an excellent matrix material for immobilizing the long-lived isotopes in its lattice. A high burn-up attainable in thorium-based fuel will also allow extraction of a much higher energy output from the operation of a single fuel cycle. In case, spent fuel reprocessing is not adopted for technological or other reasons, early induction of thorium with the currently available seed will help in continuation of nuclear power generation programme further for several additional decades. B&B reactors and molten salt reactors with only volatile fission product removal will have distinct advantages over the current generation reactors in such a situation.

Scenario-B represents the situation in which the starting fissile inventory is modest but the reserve of thorium is very large.

In this situation, the early stage of nuclear programme necessarily aims at a rapid growth of fissile inventory. As explained earlier, adopting the closed U–Pu fuel cycle is essential for rapid breeding of fissile material, the high value of η for Pu^{239} in fast spectrum being the deciding factor. The three-stage Indian nuclear programme has evolved mainly based on this argument. In case, the demand of nuclear energy is growing fast, rapid growth of fissile material is absolutely necessary. Plutonium burning fast reactors, especially with metallic fuels, is known to have the highest breeding ratio and correspondingly low doubling time. Breeding is also possible in the thermal spectrum provided excellent neutron economy is achieved by avoiding parasitic absorption of neutrons as much as possible. Molten salt reactors offer such a possibility and may be deployed for fissile breeding and *in situ* burning of the long-lived actinides. Though with a lower breeding ratio, reactor doubling time for MSR will be closer to that of fast reactors because of a much lower total requirement of fissile material in the former. The growth in the fissile inventory, however, will not be as fast through MSRs as in the case of fast reactors with significantly higher breeding ratio.

Since MSR technology is yet to be developed as a commercial option, it is attractive to consider thorium deployment in present generation reactors. It has been argued earlier that amongst existing designs, PHWRs having flexibility in fuel management have certain advantages over LWRs in deploying thorium. This saves a considerable amount of natural uranium and will help building U^{233} inventory. It is important to note that with a stock of U^{233}, one can try out newer reactor designs such as AHWR and MSR which can demonstrate self-sustaining Th–U^{233} cycle.

The selection of any of the above-mentioned paths needs a careful examination of the fuel design and the suitable reprocessing schemes. From the consideration of easy reprocessing, one should be able to segregate ThO_2 pins from the spent fuel bundles for reprocessing them separately to produce U^{233}. The advantage of a breeding ratio of 1.05 in MSRs cannot be retained if U^{233} is mixed with U^{235} and higher actinides.

Scenario-C is related to the proliferation resistant nuclear energy programmes. For obvious reasons, the once through fuel cycle will be operational in such a case.

In order to maximize fuel utilization, one needs to achieve highest burn-up, which is possible in a fuel of mixed thorium, enriched uranium and/or plutonium. While fissioning of seed nuclides produces energy in the early stages, burning of *in situ* bred U^{233} predominantly contributes towards the energy production at later stages. The spent fuel in such a case is not easy to reprocess. Moreover, separated fissile material will contain γ-active nuclides, which will make it unsuitable for diversion in the weapon programme.

Scenario-D refers to situations where there is a serious concern over availability of suitable sites for setting up nuclear power stations.

The actual footprint of nuclear power stations, which includes the reactor building, the conventional island and some utilities, is quite small compared to that required for energy production by other means. This advantage, however, is offset by the fact that in several countries a mandatory exclusion zone of about 1 km radius and further a low population zone is a regulatory requirement. Designs of inherently safe reactors (with several passive safety features), a battery of small and medium size sealed reactors and reactors operating at near ambient pressures are opening up the possibilities of operating compact and safe reactor systems. They may obtain regulatory

clearance for reduced exclusion zone. Some of the design concepts described in this article, namely, AHWR and molten salt reactors, may eventually find acceptability under the condition of constrained space. High burn-up thorium-based fuel will have an edge over the present day uranium-bearing fuel in such reactors.

It may be concluded by stating that thorium has a role to play in each of the cases discussed. While in Scenario-A, the primary reason for thorium induction is the reduction of radio-toxicity in the waste; in scenario-B, it is the sustainability and growth of nuclear power. In Scenario-C, proliferation resistance is the reason; while in Scenario-D, inherent safety and reduced land requirement are the primary motives for thorium induction.

Way forward

Nuclear energy is a proven option of an environmentally benign resource to meet bulk energy needs of mankind for centuries to come. In its initial short period of existence so far, where U^{235} has been the basic fissile material used, it has demonstrated its capability as a sound commercial option for supplying reliable base load power. The availability of nuclear power can be extended significantly by converting fertile U^{238} and Th^{232} to fissile materials and recycling them for a power production for few centuries.

Use of thorium, which does not have any fissile component of its own, needs initial fissile material support from the present U–Pu fuel cycle. Once U^{233}–Th fuel cycle becomes operational, it will achieve sustainability in energy production for a long time and will provide the opportunity for annihilating almost all long-lived radioactive waste. However, in order to operationalize the standalone Th–U^{233} fuel cycle, a long transition phase will be needed for generating the U^{233} inventory in existing or new design reactors.

To fulfil the long cherished dream of sustaining nuclear power by thorium utilization, there is need for the following steps.

(a) Introduction of thorium in current power producing nuclear reactors.
(b) Separation of U^{233} from the spent fuel and gradually building a large inventory of U^{233}.

The step (b) will involve either a three stream reprocessing scheme for mixed U/Pu/Th based fuel or adoption of two different reprocessing streams; one involving U/Pu and the other for pure thorium fuel pins. The second option can be adopted only when pure Th fuel pins are not mixed with U/Pu spent fuel.

The Thorex process for reprocessing spent fuel-containing U^{233} and Th has been developed in a somewhat small scale in a few countries. No industrial scale plant has yet been set up. The need of the hour is to take up reprocessing activities of a reasonable size in order to gain experience, making the technology robust and for assessing their economics.

(c) There is an urgent need for the development of nuclear reactors specially suitable for thorium fuel. The efforts of the design and development of AHWR in India and current thrust in MSBR in several countries will create a way forward for the sustained use of thorium.

1. Boczar, P. G. *et al.*, Thorium fuel cycle studies for CANDU reactors, IAEA-TECDOC-2002, 1319, p. 25.
2. Sivasubramanin, S., Lee, S. M. and Bhardwaj, S. A., Current status and future possibilities of thorium utilization in PHWRs and FBRs, Annual Conference of Indian Nuclear Society (INSAC-2000) on Power from Thorium status, strategies and direction, 1–2 June 2000, Mumbai, India.
3. Introduction of thorium in nuclear fuel cycle, Short to long term considerations, NEA no. 7224, 2015, Nuclear Science, OECD.
4. Gupta, H. P., Menon, S. V. G. and Banerjee, S., Advance fuel cycles for use in PHWRs. *J. Nucl. Mater.*, 2008, **383**, 54–62.
5. Wattal, P. K., Recycling challenges of thorium based fuel, International Thorium Energy Conference, (IThEC-13), Cern, Geneva, 27–31 October 2013.
6. Rajora, A. *et al.*, Generating one group cross–section for isotope depletion and generation calculation for thorium. A paper accepted for presentation in International Thorium Energy Conference (ThEC-2015), Anushakti Nagar, Mumbai, India, 12–15 October 2015.
7. Olson, G. L. *et al.*, Fuel summary report – Shippingport light water breeder reactor, INEEL/EXT-98-00799, Rev.-2, September 2002.
8. Lewis, W. B., The super-converter or value breeder, A near breeder uranium-thorium nuclear fuel cycle, AECL-3081, 1968.
9. Briant, R. C. and Weinberg, A. M., Molten fluorides as power reactor fuels. *Nucl. Sci. Eng.*, 1958, **2**, 797–803.
10. Lewis, W. B., The intense neutron generator and future factory type ion accelerators, AECL-3190, 1968.
11. Rubbia, Carlo *et al.*, Conceptual design of a fast neutron operated high power amplifier, CERN/AT/95-44(ET), 1995.
12. Boczar, P. G. *et al.*, Recent advances in thorium fuel cycles in CANDU reactors, IAEA-TECDOC-1319, 2002, p. 104.
13. Krishnani, P. D. and Srinivasan, K. R., A method for solving integral transport equation for PHWR cluster geometry. *Nucl. Sci. Eng.*, 1981, **78**, 97.
14. Askew, J. R., Fayers, F. J. and Kemshell, P. B., A general description of the lattice code WIMS. *J. Br. Nucl. Energy Soc.*, 1966, **5**, 564.
15. Gupta, H. P., Use of thorium in PHWRs, National Conference on Power from Thorium: Present status and future directions, Mumbai, 22–24 December 2014.
16. Gupta, H. P., Yadav, R. D. S., Menon, S. V. G. and Banerjee, S., Utilization of Thorium in Different Reactors, German Annual Meeting of Nuclear Technology, 2009 Dresden, 12–14 May 2009.
17. Balakrishnan, K., Optimization of initial fuel loading of the Indian PHWR with thorium bundles for achieving full power. *Ann. Nucl. Energy*, 1994, **21**(1), p. 1–9.
18. Banerjee, S. and Govindan Kutty, T. R., Materials Challenges for Advanced Nuclear Fuel Cycle, A paper presented in Global 2013: International Nuclear Fuel Cycle Conference, Salt Lake City, Utah, USA, 29 September–3 October 2013.

19. Balakrishnan, K., Majumdar, S., Ramanujam, A. and Kakodkar, A., The Indian Perspective on Thorium Fuel Cycles, IAEA-TECDOC-1319, 2002, p. 257.
20. Arvind Kumar, Srivenkatesan, R. and Sinha, R. K., On the physics design of advanced heavy water reactor (AHWR), IAEA-CN-164-3S03, 2010, International Conference on Opportunities and Challenges for water cooled reactors in 21st century, Vienna, 27–30 October 2009.
21. Sinha, R. K. and Kakodkar, A., Design and development of AHWR – The Indian thorium fuelled innovative nuclear reactor. *Nucl. Eng. Design*, 2006, **236**, 683–700.
22. Balakrishnan, K. and Kakodkar, A., Preliminary physics design of advanced heavy water reactor (AHWR), IAEA-TECDOC-638, March 1990.
23. Thakur, Amit *et al.*, Fuel cycle flexibility in advanced heavy water reactor (AHWR) with the use of Th–LEU fuel, International Conference on Future of HWRs, Ottawa, Canada, 2–5 October 2011.
24. Christopher Grove, Comparison of thorium and uranium fuel cycles, National Nuclear Laboratory, UK, 2012, **NNL**(11), 11593(5), 9.
25. Radkowsky, A competitive thorium fuel cycle for pressurised water reactors of current technology, IAEA-TECDOC-1319, 2002, p. 25.
26. Ramanna, R. and Lee, S. M., The thorium cycle for fast breeder reactors. *Pramana – J. Phys*., 1986, **27**(1&2), 129–157.
27. Mohapatra, D. K. *et al.*, Physics aspects of metal fuelled fast reactors with thorium in blanket. *Nucl. Eng. Design*, 2013, **265**, 1232–1237.
28. Greenspan, E., A phased development of breed and burn reactors for enhanced nuclear energy sustainability. *Sustainability*, 2012, **4**, 2745–2764; doi:10.3390/su4102745
29. Petrosky, R. C., General analysis of breed and burn reactors and limited separations fuel cycles, Ph D thesis, MIT, 2011.
30. Sekimoto, H. and Ryu, K., Demonstrating the feasibility of the CANDLE burnup scheme for fast reactors. *Trans. Am. Nucl. Soc.*, 2000, **83**, 45.
31. Kasten, P. R. *et al.*, Summary of molten salt breeder reactor design studies, ORNL-TM-1467, March 1966.
32. Roberson, R. C., Conceptual design study of single-fluid molten-salt breeder reactor, ORNL–4541, June 1971.
33. Hirakawa, N. *et al.*, Molten salt reactor benchmark problem to constrain plutonium, IAEA-TECDOC-1319, 2002, p. 25.
34. Trans Atomic Power, Molten Salt Reactors, Technical White Paper, V 1.0.1, March 2014.
35. Gupta, H. P., Preliminary design of molten salt converter reactor, Conference on Molten Salt Nuclear Technology (CMSNT), BARC, Mumbai, India, January 2013.
36. Kapoor, S. S., Accelerator-driven sub-critical reactor system (ADS) for nuclear energy generation. *Pramana – J. Phys*., 2002, **59**(6), 941–950.
37. Degweker, S. B. *et al.*, Accelerator driven subcritical systems with enhanced neutron multiplication. *Ann. Nucl. Energy*, 1999, **26**, 123.

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