Energy from thorium—An Indian perspective

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Abstract. Nuclear energy as a sustainable resource in India has been very clearly formulated in the three stage nuclear programme. The role of thorium as a potential fuel in the third stage of this programme has also been elucidated. With this aim there have been pioneering research efforts in all aspects of the thorium fuel cycle. Thorium being fertile and not accompanied by the fissile species requires the use of a fissile topping. There have been several studies in India on the use of thorium in different reactor systems from thermal to intermediate and fast spectrum, molten salt reactors, high temperature reactors, compact nuclear power packs and even Subcritical systems. In this paper, we present some of the research studies on use of thorium in thermal reactor systems. We give an overview of the neutronic properties of thorium and the bred fissile material and then proceed to show the performance potential in different reactor systems. We also present the innovative Indian reactor designs which utilize thorium, namely the Advanced Heavy Water Reactor (AHWR) and the Indian High Temperature Reactors.

Keywords. Thorium utilisation; AHWR-LEU; Th; fuel cycle; IHTR.

1. Introduction

The use of Thorium has been successfully demonstrated in India from irradiation in research facilities to use thorium bundles for flux-flattening in the Pressurised Heavy water Reactor (PHWR) from KAMINI to design of an Advanced Heavy water Reactor. In the rest of the world too, research reactors like the HTGR in Peach bottom, USA and power reactors like Fort St. Vrain reactor in USA to the HTGRs of Germany thorium fuelled reactors have been operated successfully. However, efficient conversion into the fissile ^{233}U , separation of this ^{233}U in the spent fuel, its subsequent fabrication, the radiological problems of handling irradiated thorium fuel are technologically challenging. Reactors world over have been governed by the availability of uranium. For India, since our indigenous uranium sources are limited, we have to go for thorium cycle at the earliest for energy security and sustainability (Sinha 2006). The superior neutronic characteristics of ^{233}U in thermal spectrum render it as a viable option even as a

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potential breeder. We have been developing the thorium technology over the years, with a number of R&D programs and commercial application in our PHWRs (Kakodkar 2000).

2. Neutronic properties of the isotopes of the thorium fuel cycle

The thermal neutron absorption cross-section of ^{232}Th (7.34 barns) is about three times higher than that of ^{238}U (2.68 barns). Thorium (^{232}Th) having a higher absorption cross section than ^{238}U offers the possibility of absorbing the extra neutron thereby aiding a better conversion (Lung 1997). An important aspect of the Thorium fuel is that because of the larger absorption in Thorium itself, the fissile inputs would have to be higher than that of a comparable uranium cycle. The neutronic characteristics for some of the isotopes of importance in the Th and U cycles are given in table 1. In the thermal spectrum, ^{233}U has best neutronic properties among the fissile isotopes (table 1). The neutronic parameters in the table have been generated from the PREPRO code system (Cullen 2000).

An important nuclear characteristic favouring thorium cycle is the η value of bred fissile material ^{233}U . ^{233}U has an η value greater than 2.0 for fission caused by thermal neutrons, which remains constant over a wide energy range, in thermal as well as epithermal regions, unlike ^{235}U and ^{239}Pu , which makes the thorium fuel cycle less sensitive to the type of reactor.

The value of η which is a measure of the excess neutron available for fertile-to-fissile conversion is plotted in figure 1 for ^{233}U , ^{235}U and ^{239}Pu . The value of η for ^{233}U remains constant below 2 eV however it has a sharp dip around 2–3 eV (figure 1, table 1). From a design standpoint the higher thermal absorption cross section in ^{232}Th as compared to ^{238}U would lead to a harder neutron spectrum in thorium based reactors (Umasankari *et al* 2009). This means that the effective thermal flux would be lower in such systems. Because of the *in situ* generation of ^{233}U or better conversion, the reactivity loss in thorium-based systems is lower.

Thorium being lower in the periodic table generates less long-lived Minor Actinides (MAs), which is mainly ^{231}Pa . Hence the radio-toxicities associated with actinides of spent fuel in thorium cycle are an order of magnitude less than that of spent fuel from U-Pu cycles.

The fissile ^{233}U is produced via ^{233}Pa , which is the intermediate isotope in the conversion chain of thorium to ^{233}U . It has a longer beta decay half life of about 27 days. As a result, ^{233}Pa builds up to a significantly high level in equilibrium and a portion of it gets lost by neutron absorption before it decays to ^{233}U . Some reactor concepts have circumvented this problem by on-line fuel reprocessing thereby making the probability of decay to fissile nuclide relatively higher.

Neutron capture in ^{233}U leads ^{234}U , which has a significant absorption cross section and acts as a load in thorium-fuelled systems. Recycle of the bred ^{233}U would have to account for the presence of ^{234}U in the spent fuel (Umasankari & Krishnani 2003).

2.1 Radiological concerns of the thorium fuel cycle

The thorium fuel cycle also has associated radiological problem in ^{232}U , whose daughter products are hard gamma emitters. ^{232}U is formed via (n,2n) reactions, from ^{232}Th , ^{233}Pa and ^{233}U . The half-life of ^{232}U is about 69 years. The daughter products of ^{232}U are hard gamma emitter like ^{208}Tl (2.6 MeV) with very small half-lives (figure 2). As a result, the dose rate increases with time. But this tell-tale signature also makes the irradiated thorium fuel proliferation resistant.

Table 1. Neutronic properties of some fissile and fertile isotopes.

	²³² Th	²³³ U	²³⁴ U	²³⁵ U	²³⁶ U	²³⁸ U
Thermal data : Cross sections at (0.0253 eV)*						
(Maxwellian average are given in brackets) (temperature = 0.0253 eV; Range 1.0E-05 to 10 eV)						
σ_a barns	7.34	576.45	100.96	683.77	5.18	2.68
σ_c barns	7.34	45.24	100.89	98.69	5.13	2.68
	(3.446)	(35.18)	(115.88)	(50.60)	(39.14)	(18.02)
σ_f barns	0.00	531.21	0.067	585.09	.047	1.679E-05
	(0.00)	(308.674)	(.07304)	(269.1357)	(0.456)	(1.575E-05)
α^1	—	0.085	—	.169	—	—
ν^2 (at 0.0253 eV)	—	2.497	2.365	2.437	2.371	2.493
η^3	—	2.301	—	2.085	—	—
Maxwellian average (Range 1.0E-05 to 0.1 eV)						
σ_a barns	6.88	541.05	94.45	631.90	4.89	2.53
σ_c barns	6.88	43.22	94.39	92.27	4.84	2.53
σ_f barns	0.00	497.84	0.063	539.63	0.044	1.582E-05
Resonance Integral (0.5 eV–100 keV)						
Absorption barns	83.73	907.47	631.53	408.75	345.07	274.81
Capture barns	83.73	140.81	630.83	139.79	340.61	274.81
Fission barns	5.701E-06	766.67	0.7026	268.96	4.45	0.0023
α	—	.184	—	.520	—	—
First major low energy resonance eV	21.81	—	5.16	0.28	5.50	6.67
Cross Section at the first resonance barns	8151.06	—	51969.16	194.99	36195.30	21442.56
Fast fission threshold MeV	.004	—	—	—	—	—
Fast energy data (Fission spectrum averaged one group cross section)						
(Range 1.0 keV to 20 MeV)						
Fission barns	0.731	2.026	1.15	1.339	0.56	.295
Capture barns	0.135	0.102	0.149	0.157	0.151	.106
ν	2.4	2.89	2.79	2.82	2.74	2.81
η	1.91	2.86	2.64	2.74	2.65	2.69

¹ α is the capture-to-fission ratio² ν is the average number of neutrons produced per fission³ η is defined as the average number of neutrons produced per thermal neutron absorbed

Figure 1. The neutron regeneration factor η for different fissile isotopes.

Figure 2. Radioactive decay chain of thorium.

Another concern is the actinides in the spent fuel which contribute to long term radio-toxicity. The ^{233}U decay chain has ^{229}Th as one of its decay product which is important because it is generated due to the α -decay of ^{233}U and is lost by its own α -decay. So in the spent thorium fuel its concentration first increases with time and then decreases (the concentration of

Figure 3. Comparison of neutron spectra in different types of reactor.

^{229}Th peaks at about 10^5 years). ^{229}Th has a high *toxicity potential*¹ because of its 0.19 MeV gamma. However, its daughter product ^{213}Bi has very important medical application for cancer therapy.

The radio toxicities are defined as the amount of water or air required to dilute the toxicity of the nuclide to the level of maximum permissible concentration. Due the fact that the spent thorium fuel has higher activity and toxicity, there is a requirement to recycle ^{233}U repeatedly to reduce the long term effects vis-à-vis its storage.

2.2 Thorium utilization and neutron spectrum

It would be prudent to design systems, which exploits all the neutronic properties of thorium to our advantage. With thorium as the fertile species, one can afford to use light water in thermal reactors. On the other hand, heavy water offers a large volume to load thorium fuel. This would supply adequate neutron for efficient conversion to ^{233}U .

In figure 3, the neutron spectra in typical thermal and fast reactors are compared. This has to be seen in relation with the basic energy dependent cross sections which effectively translate into reaction rates. In thermal reactor systems, about 95% of the fission takes place below 0.625 eV.

2.3 Thorium vs Uranium as fertile host

One of the important parameter that characterises a fuel is its burnup or the amount of energy produced per unit mass of the fuel. A high discharge burnup would mean good fuel utilisation. A study based on the performance of ^{232}Th and ^{238}U as a fertile host in a thermal spectrum like

¹Toxicity potential is defined as the absorbed dose in sievert / Te HM equivalent to the biological effect of radioactivity.

Figure 4. Performance potential fissile topping in thermal spectrum.

the one in PHWR shows that LEU (Low enriched Uranium) gives a maximum burnup of about 25 GWd/T and the bred fissile material, Pu saturates at a low 0.6 % (Balakrishnan 1999). In a typical PWR spectrum, the plutonium contents are about 1.0% at burnups of 33 GWd/T. In contrast, the bred fissile material in thorium systems reaches a maximum of 1.5% in a thermal spectrum. But for such fissile performance one would have to go for high burnups i.e., about 45 GWd/T or above (Balakrishnan 1999). It can be seen from figure 4, that initially ^{235}U enrichment in thorium gives lower burnups as compared to use of ^{235}U in ^{238}U . This is due to the parasitic absorption in ^{232}Th . But for enrichments of the order of 3.0% or above, thorium gives a distinct advantage in terms of burnup. Thus the choice would be to use fissile contents of the order of 3.0% in an ambient heavy water spectrum.

3. Studies on use of thorium in different reactor systems

Before going to the thorium reactors being designed in India, we present a few studies done for use of thorium in different reactor systems. Using Plutonium along with thorium would increase the fissile base i.e., natural uranium can be conserved; the plutonium produced from thermal systems could be recycled into the same reactor. This would then achieve a two-fold advantage of burning the plutonium to reduce the long-term toxicities and also to produce another fissile material, ^{233}U . India has been pursuing a closed cycle option as spelt out in the three stage document. Studies have shown that plutonium can be recycled with thorium in almost all types of reactor systems (IAEA TECDOC-1319, 2002).

3.1 Thorium in thermal reactor systems

Pu discharged from PHWRs used with thorium in the typical 220 MW(e) reactor design gives a discharge burnup of about 12000 MWd/T (Balakrishnan 1999). This option was a retrofit

Table 2. Performance potential for RG–Pu with Thorium in different reactor systems for a target burnup of 60 GWD/T.

Parameter	PHWR 220 MW(e)	BWR Laguna Verde ¹	VVER Kalinin ²	AHWR 300 MW(e)
Fissile requirement/unit energy g/MWD	0.66425	1.2109	1.247	0.9329
Total Pu consumed/unit energy g/MWD	0.71776	0.98464	0.4698	0.85376
Fissile Pu consumed	0.62003	0.8448	0.4503	0.7415
Total Pu in Spent fuel/unit energy g/MWD	0.27139	0.8284	1.437	0.62464
Fissile Pu in spent fuel/unit energy g/MWD	0.04384	0.3157	0.7899	0.20392
MA/unit energy g/MWD	44.446	84.89	53.26	99.19
% Fissions from ²³³ U at equilibrium cycle	32.5	30.2	20.0	18.7
Power coefficient of reactivity mk/% F.P	−0.0572	−0.207	−0.1626	−0.1697

¹Laguna Verde BWR (IAEA TECDOC 849) Power level 2027 MWth/658 MWe²Kalinin VVER (IAEA TECDOC 847)Power level 3000 MWth/1000 MWe

into the existing operating reactor, which implies that control and safety features were unaltered. Table 2 throws light on a comparative study made on the use of Reactor Grade² Plutonium (RG–Pu) discharged from a typical Light Water Reactor (LWR), in different type of reactor systems. The reactor systems compared are two heavy water reactors, PHWR and AHWR and two light water reactor systems, a modern BWR (Laguna Verde) and VVER (IAEA GC, 2008). The target burnup chosen was 60 GWD/T. Fuel cycle parameters compared initial fissile content required, Plutonium consumes, Fissile plutonium consumed and minor actinides in the spent fuel in units of g/MWD. Thorium utilization was quantified on the basis of energy produced from ²³³U. The operational characteristics were compared by means of power coefficient of reactivity, which is a feedback parameter governing a temperature or a power rise. It is found that the PHWRs require the least initial Pu content as the conversion ratio is best in this system. In the BWR case, Pu consumed is maximum. Depending on the aim of Pu incineration or ²³³U generation, reactor system is chosen. Among the cases studied, heavy water moderated lattices offer best characteristics for utilisation of thorium. In particular, the use of RG–Pu and thorium in 220 MWe PHWRs offers an attractive option for near-term deployment (IAEA GC, 2008).

3.2 Thorium utilisation in molten salt breeder reactor (MSBR)

The molten salt breeder is based on circulating fluid fuel. The fuel is mostly in the form of molten fluorides of uranium and thorium. The fuel salt passes through graphite blocks. A major advantage of the MSBR is the on-line removal of fission products, implying that a constant reactivity can be maintained. Also, the circulating fuel salt spends considerable time outside the reactor in low neutron flux region which aids the conversion of ²³³Pa to ²³³U. Control can also

²Pu composition : ²³⁸Pu/²³⁹Pu/²⁴⁰Pu/²⁴¹Pu/²⁴²Pu :: 3.5 / 51.87 / 23.81 / 12.91 / 7.91.

be achieved by the circulating fluid and no extra control material is required. Some preliminary calculations done with liquid thorium- ^{235}U fluoride salt flowing in holes in a graphite matrix shows that a breeding ratio up to 1.07 can be obtained.

3.3 Conversion and breeding in thorium fuelled systems

Thorium has high thermal absorption cross section and does not have any intrinsic fissile element. By feeding the neutron to thorium, not only is the excess reactivity suppressed but also ensures production of ^{233}U , which is used as fuel within the same reactor. The production of fissile ^{233}U in an ambient thermal spectrum from a pure thoria fuel is illustrated in figure 5. It can be seen that at higher thermal fluxes, the faster is the rate of ^{233}U generation and also the ^{233}U saturates or reaches an asymptotic level. The ambient thermal flux in the vicinity of thoria rods can be increased by design. This feature is exploited in the conceptual ATBR design (Jagannathan *et al* 2001).

3.4 A conceptual thorium breeder reactor design

A Thermal Breeder Reactor (ATBR) is a reactor concept being studied for large scale utilization of thorium in thermal spectrum. The ATBR core is a 600 MWe or 1875 MW thermal power reactor concept. It uses vertical pressure tubes arranged in hexagonal lattice. This reactor design utilises the neutronic properties of ^{232}Th and ^{233}U . The fuel cycle is based on increasing the fertile captures and minimising the fissile depletion. This is essentially a seed and blanket concept, where one time irradiated thorium fuel is reconstituted with the seed fuel. The fundamental objective of the design is to control the excess reactivity by large scale loading of pure thorium without mixing it with any fissile material (Jagannathan *et al* 2001). Thorium would initially act like a pure neutron absorber but eventually become an effective fuel after acquiring sufficient amounts of the *in situ* bred fissile ^{233}U atoms. The innovation has been in reducing the wasteful

Figure 5. ^{233}U production as a function of thermal neutron flux.

neutron capture by external absorbers like control rods, boron and gadolinium during reactor operation at full power. A reactor configuration has been arrived at with very little reactivity variation for a cycle length of 750 days. This is achieved by intrinsic fertile to fissile conversion. Studies have shown that this reactor can use Pu from discharged fuel of LWR or PHWRs as the seed input.

4. Use of thorium in the Indian reactors

India has a steady programme of thorium fuel cycle demonstration defined by the objectives of the three stage programme. To this end research activities have been initiated in every aspect of the thorium fuel technology (IAEA TECDOC-1319, 2002).

- Thoria rods (J-rods) have been continuously irradiated in CIRUS the reflector region (between the inner and outer graphite reflectors) and ^{233}U has been accumulated. These rods have then been reprocessed. These J-rods have about $\sim 2\text{--}3$ ppm of ^{232}U in uranium (being irradiated in the low flux reflector region) (Basu & Srinivasan 1990).
- PURNIMA-II (1984–1986): India had initiated thorium studies as early as seventies and the first research reactor built with ^{233}U fuel generated in J-rods in the CIRUS was PURNIMA-II. Experiments were performed in PURNIMA-II facility with uranyl nitrate solution containing U^{233} reflected by BeO blocks (Srinivasan *et al* 1986). Variation of critical mass with different H/ U^{233} ratio was measured. Measurements of various reactivity parameters and neutron lifetime were also performed. These experiments also gave confidence in handling U-233.
- PURNIMA-III (1990–93): Experiments were performed with ^{233}U -Al dispersion fuel in the form of plates in PURNIMA-III (Pasupathy *et al* 1987). BeO canned with zircaloy was used as reflector. Void coefficient and temperature coefficient of reactivity were also measured. These measurements helped in finalising the core of KAMINI reactor. They also helped in validating the cross sections of ^{233}U and Be.

4.1 KAMINI

A reactor based on U^{233} fuel in the form of plates of U-Al alloy was commissioned in IGCAR at Kalpakkam in 1996. It is the **only operating reactor in the world with ^{233}U** as fuel (Pasupathy *et al* 1990; Basu & Srinivasan 1990). The reactor power is 30 KWth and has very high flux to power ratio. It has three beam holes used for activation analysis and facility for neutron radiography of Fast Breeder Test Reactor (FBTR) fuel.

4.2 Thorium for flux flattening in PHWR

One major step in thorium utilisation in India was achieving initial flux flattening with thoria bundles. The maximum power achievable in the fresh core of 220 MWe PHWRs is only 70% Full Power (FP) if it is loaded with all Natural Uranium fuel bundles, and this full power can be achieved only after about 100 Full Power Days (FPD) of operation. However, the maximum operating power can be improved by flattening of the flux. In earlier reactors like RAPS, MAPS and NAPS, depleted uranium was used for this. As an alternative thorium has been used in the fresh core of 220 MW(e) PHWRs. 35 lattice locations in the 220 MW(e) PHWR core were optimised for loading one thoria (ThO_2) bundle in each channel such that it provides the

required flux flattening without any loss of burnup in UO_2 fuel and without compromising the worth of Shut Down Systems (Balakrishnan & Kakodkar 1993). For the first time such a loading was experimented in 1991 at KAPS-1. The same loading pattern was followed in the subsequent reactors like KAPS-2, KGS 1 and 2 and RAPP-3 and 4. This represents a unique way of utilizing thorium and about 210 thorium bundles have been irradiated. The Post irradiation examination (PIE) of this irradiated thorium fuel was also done to estimate the uranium composition (Ganesan 2004).

However, in RAPS types of reactors, because of the absence of mechanical shut off rods and the secondary shut down system, flux flattening in the fresh core was achieved by using a lower number of thorium assemblies i.e., 18 thorium bundles in symmetrical positions.

5. Advanced heavy water reactor

India is engaged in designing an innovative reactor called **Advanced Heavy Water Reactor (AHWR)**, which will not only be a demonstration of thorium based technologies, but would also be a significant improvement in the safety features of the existing PHWRs (Sinha & Kakodkar 2006). The AHWR is a 920 MWth, vertical pressure tube type thorium-based reactor cooled by boiling light water and moderated by heavy water designed to maximise power production from thorium. The AHWR has many attractive features like negative coolant void reactivity, heat removal through natural circulation and other passive safety features (Sinha & Kakodkar 2006). The core configuration of the AHWR core is shown in figure 6. The physics design has several challenges namely, achieving safety characteristic of negative void reactivity with a boiling light water coolant in a pressure tube type of system and achieving near self-sufficiency in ^{233}U . While AHWR will also need a small external feed of plutonium or uranium, most of its energy output will come from thorium. The strong coupling between the neutronics and thermal hydraulics puts an extra requirement of flat radial power characteristics (Srivenkatesan *et al* 2000). Two different AHWR designs are being developed with thorium–plutonium and thorium–uranium cycles, in both closed and open cycle options. The plutonium topped version which is the reference AHWR design, AHWR-Ref and the LEU topped variant is called as the AHWR-LEU.

The reactor core of AHWR-Ref design consists of 513 lattice locations in a square lattice pitch of 225 mm. Of these, fuel assemblies occupy 452 locations and 61 locations are reserved for the reactivity control devices and shut down system-1 (Kumar *et al* 2009). Among the 61 locations for the reactivity devices, 37 locations are used for locating Shutoff Rods (SORs). The remaining 24 are used for locating control rods (CRs) for short-term reactivity compensation and power maneuvering during normal operation. The control element is boron carbide (B_4C) packed in SS tubes placed between SS shells both for control rods and shut off rods (Sinha & Kakodkar 2006). The main core design features are tabulated in table 3.

The main heat transport system is designed to transport heat from the fuel through natural circulation of boiling light water coolant. The active fuel length is 3500 mm. The reactor has two independent, functionally diverse fast acting shut down systems, namely, shutdown system-1 (SDS-1) consisting of mechanical shut off rods and shutdown system-2 (SDS-2) based on liquid poison injection technology. The SORs drop inside a perforated zircaloy guide tube.

Thorium facilitates light water to be used as coolant.

- Boiling light water enhances thermal utilization.
- The depletion of ^{233}U in the inner pins is nearly matched with the ^{233}U production in the outer pins (Umasankari & Srivenkatesan 2007).

Figure 6. Layout of AHWR equilibrium core.

Table 3. Important design parameters of AHWR.

Reactor Power	300 MWe (920 MWth)
Fuel	54 pin cluster (Th, ²³³ U)MOX+(Th,Pu) MOX or (Th,LEU fuel)
Coolant	Boiling light water
Moderator	Heavy water
Total no. of channels	513
No. of fuel channels	452 (AHWR-Ref core) 444 (AHWR-LEU core)
Lattice Pitch	225 mm
Primary shut down system	37 shut off rods (B ₄ C absorber) for the AHWR-Ref design 45 shut off rods (B ₄ C absorber) for the AHWR-LEU design
Secondary shut down system	Liquid poison injection in moderator (gadolinium nitrate injected into moderator)
No. of control rods	24
Passive poison injection	Poison injection through a passive valve due to increase in steam pressure

- Use of plutonium to achieve negative void coefficient and depletion of plutonium has to be compensated by mild neutron absorbers for achieving negative coolant void reactivity at higher burnups.
- In the AHWR-LEU design on the other hand, there would be a build-up of plutonium.

5.1 Fuel cluster for AHWR-Ref core

The fuel cluster for the equilibrium core called as the composite cluster, consists of 54 pins arranged in an array of 12, 18 and 24 pins, where the inner 30 pins have (Th,²³³U)MOX fuel and the outer 24 pins (Th,Pu)MOX fuel. The first array of 12 pins has a ²³³U content of 3.0% by weight and the middle 18 pins have 3.75% ²³³U. The 24 (Th,Pu)MOX pins in the third array have an axial gradation of 4.0% Pu in the lower half of the active fuel and 2.5% Pu in the upper part (Kumar 2000). The plutonium used is the one discharged from PHWRs at an average of 6700 MWD/T. The isotopic vector is ²³⁹Pu 68.79% ²⁴⁰Pu 24.6% ²⁴¹Pu 5.26% and ²⁴²Pu 1.35%. The cross section of the fuel cluster is shown in figure 7. The current design has a multi functional central displacer, which consists of Stainless Steel (SS) in the top half and Zircaloy-2 in the bottom half of the reactor. The top half of the core contains lower plutonium content and this required an absorber in the form of SS to have negative coolant void reactivity throughout the burnup.

5.1a Pu as fissile topping: The equilibrium fuel cycle is based on the conversion of naturally available thorium into fissile ²³³U driven by plutonium as external fissile feed. The uranium in the spent fuel will be reprocessed and recycled back into the reactor. The basic fuel cycle is based on the fact that the AHWR core should be self-sustaining in ²³³U. The ²³³U required is to be bred *in situ*, which implies that the initial core will be based solely on thorium–plutonium fuel. There will be gradual transition from the initial core, which will not contain any ²³³U to an equilibrium core, which will have composite clusters. During this transition phase a relatively

Figure 7. Cross section of the AHWR fuel cluster.

Figure 8. ^{233}U and Uranium build-up in the AHWR equilibrium core cluster with Pu topping.

large quantity of plutonium is required. ^{233}U is then recycled into the same core operating in a closed fuel cycle. The fuel cycle is thus involved where a three way reprocessing for the outer (Th,Pu)MOX pins. The initial content of the fuel in the composite cluster is about 34.5 g/kg of ^{233}U fuel. With burnup, the loss of fissile ^{233}U in the (Th, ^{233}U)MOX pins has to be compensated by the conversion of fertile ^{232}Th in the (Th,Pu)MOX to fissile ^{233}U . This can be seen in figure 8.

5.1b LEU-Th fuel cycle in AHWR: The physics design of AHWR has been focussed on the Th-LEU fuel in an open cycle configuration. With ^{235}U content of about 4.2% discharge burnup of about 60 GWd/T has been achieved. The equilibrium fuel cluster is loaded with 30% LEU in ThO_2 in the inner 12 pins, 24% LEU in the middle ring of 18 pins and an average of 16% LEU in the outermost 24 pins (Thakur *et al* 2011). Gadolinium (Gd) is used as burnable poison to suppress power ripples in two pins on the inner ring. The cross section of the fuel cluster is shown in figure 9. The AHWR-LEU core, by design has enhanced safety and proliferation resistance characteristics. The spent uranium fuel is denatured by the presence of ^{238}U . Also, the presence of ^{232}U in discharged U and ^{238}Pu in discharged Pu enhances its proliferation resistance characteristics (Neelima *et al* 2009). The power from Th- ^{233}U would be about 39%.

The AHWR-LEU fuel management has posed several challenges from managing a flat power/flux profile across the core as the Gd burns to managing the in-core peaking at the empty reactivity device locations. Several fuel management schemes through mini batch-fuelling are being studied as the core advances from initial to equilibrium phases. Presently feasibility to remove in-core ECCS injection is being studied. Also the core is being re-optimised in order to get better shut down margins.

5.2 Safety features of AHWR

It is well-known that in an operating reactor, the neutron power is maintained. The reactor is generally said to be in a dynamic equilibrium, where the neutron flux is essentially constant in

Figure 9. AHWR equilibrium Core Cluster with Th-LEU fuel.

time. The departure from this state is generally termed as transients. This departure is quantified in terms of reactivity.³ The reactor regulating and protection systems are designed to take care of any such transient behaviour and a multi-level protection ensures complete safe shutdown capabilities. The physics of safety features as envisaged by the AHWR design is listed below (Kumar *et al* 2009).

- (1) Negative fuel temperature coefficient
- (2) Negative power coefficient (beyond hot-standby condition)
- (3) Negative core-averaged coolant void reactivity coefficient
- (4) Low excess reactivity
- (5) Low speed of control rod withdrawal
- (6) Low power density
- (7) Two independent and functionally diverse fast acting shutdown systems
- (8) In addition, the reactor regulating system continuously monitors and controls local and global power.

The reactor control ensures a redundancy in core monitoring. The reactor is designed to have several inherent safety features and also several on line parameters render prompt action for a safe operation.

All the reactivity coefficients have been made negative by design, which means that during any transient negative feedbacks, are introduced, making the reactor power to reduce. The low

³Fractional departure from criticality is defined as reactivity.

power density implies smaller power produced from unit volume of the core, thereby reducing the severity of the consequences of any transient.

A major passive safety feature is core decay heat removal through isolation condensers (Sinha & Kakodkar 2006) and sufficient core submergence for prolonged shutdown. The coolant is boiling light water and it results in a strongly coupled neutronics and thermal hydraulics. The important feature of the AHWR core power distribution is that it has flat radial power characteristics (figure 10a) and a bottom peaked axial power distribution to get better thermal hydraulic margins (figure 10b). These are some of the physics solutions arrived by design to enhance safety.

Figure 10. (a) Thermal flux distribution in AHWR. (b) Axial flux distribution for a typical loading in AHWR after thermal hydraulic iterations.

6. High temperature reactor

BARC is designing high temperature reactor systems for different objectives. The designs being pursued for the high temperature reactor project is the Compact High Temperature Reactor (CHTR) and the 600 MWth Innovative High Temperature Reactor (IHTR). The CHTR is being designed as a demonstration facility for various technologies related to the desing of High Temperature Reactors. The IHTR utilizes the process heat to disassociate water into its constituent gases H_2 and O_2 .

CHTR is BeO moderated, Th- ^{233}U or HEU fuelled, Lead Bismuth Eutectic (LBE) cooled, vertical prismatic type reactor rated for 100 kWth power. It has 19 Fuel assemblies. Each fuel assembly of CHTR consists of 12 fuel bores filled with fuel compacts made of TRISO coated particles. Other salient design features are passive core heat removal by natural circulation of liquid heavy metal coolant, compact design to minimize weight, longer core-life (~ 15 years) (Dwivedi *et al* 2009).

In the current design of CHTR, the core has primary shutdown system based on six absorber rods to be inserted in inner coolant channels and liquid poison injection of Indium in 12 BeO reflector holes acting as secondary shut down system. The cross section of the CHTR core is shown in figure 11.

The major core design objectives are use of thorium based fuels with low fissile inventory, prolonged operation without refuelling and compact design to minimise weight of the reactor. The passive safety features include core heat removal by natural circulation of liquid heavy metal coolant, passive rejection of entire heat to the atmosphere under postulated accident condition.

The **IHTR-Core** is being designed with 150000 pebbles floating in the coolant in annular core of 150 cm thickness and graphite as moderator. The schematic of the pebble bed in IHTR core is shown in figure 12. The fuel particles, called TRISO coated particles has a kernel of 250 μm radius that contains the heavy metal ($^{233}UO_2 + ^{232}ThO_2$, where 7.6 wt% UO_2 of total heavy metal) and is surrounded by four carbon-based coating layers, forming a total outer radius of 450 μm (Dulera & Sinha 2008). The proposed coolant consists of mixture of fluoride salts (FLiBe) which has a freezing point of 400°C and a boiling point of $\sim 1400^\circ C$ at atmospheric pressure. Operating temperature of IHTR is about 1000°C.

Figure 11. Cross section of the CHTR core.

Figure 12. The pebble bed IHTR.

	1	2	3	4	5	6	7	8	9
A				●	●	●			
B		●	●	●	●	●	●	●	
C		●	●	◐	●	◐	●	●	◑
D	●	●	●	●	●	●	●	●	●
E	●	●	◐	●	●	●	◐	●	●
F	●	●	●	●	●	●	●	●	●
G		●	●	◐	●	◐	●	●	
H		●	●	●	●	●	●	●	
J				●	●	●			

Number Of Lattice Locations = 61

● 19-pin Natural-U Metal Fuel Cluster

◐ Shut-Off Rods, ◑ Absorber Rod

Lattice Pitch= 24.5 cm

Figure 13. Core layout of AHWR-critical facility.

Figure 14. Measured fine structure flux in NU and mixed pin cluster in AHWR-CF.

Figure 15. Mixed pin (Natural U-ThO₂) experimental cluster.

7. Experimental validation in the critical facility at BARC

A low power critical facility has been set up in BARC for validating the physics design codes to qualify the AHWR physics design (Raina *et al* 2006). This research reactor uses natural uranium metal fuel. It is low power facility rated at 100 W and the flux level is 1.0×10^8 n/cm²/s. This reactor went critical on 7th April, 2008. The observed critical height for the reference core

configuration was 226.7 cm which agreed well with the estimated value of 226.5 cm. The core layout of the Reference core of AHWR-CF is shown in figure 13.

Experiments performed so far included, measurement of critical height for various cores, calibration of reactivity devices, neutron spectrum measurement on central cluster, cadmium ratio measurement at infinite dilution, Fine structure flux measurement inside the central lattice of natural Uranium and thorium clusters. The measured fine structure flux across the 19 pin nat. uranium fuel cluster loaded in the AHWR-CF is presented in figure 14.

Recently, integral experiments with $\text{ThO}_2\text{--U}$ mixed pin cluster and $\text{ThO}_2\text{--PuO}_2$ cluster (figure 15) were performed by loading them in the central E-5 and several other locations.

8. Post Irradiation Examination (PIE) of thorium fuel

The amount of actinides and fission products in spent fuel constitute an useful information for fuel cycle analysts. Post irradiation examination gives not only an estimate of the concentration of irradiation levels, but also qualifies the structural integrity of the fuel. Post irradiation examination of thorium fuel irradiated in Indian PHWRs has been done. The thorium bundles used for flux flattening (section 4.2) have been irradiated up to 500 Full Power days (FPDs) and have accumulated a burnup of about 10000 MWd/T. The amount of ^{232}U content in the spent fuel has been estimated by chemical dissolution studies, gamma spectrometry and TIMS and spectroscopic methods (Ganesan 2000). Also advanced fuels of (Th,Pu) MOX have been irradiated time and again in research reactors and these pins will be taken up for PIE in the near future.

9. Summary

From an Indian perspective, early induction of thorium into the nuclear energy scene would be advantageous. To this end, research is going on in many fronts. There have been many significant indigenous efforts in thorium utilization. Thorium pins irradiated in CIRUS have been reprocessed and the bred ^{233}U been used as fuel in KAMINI reactor. Mechanical and metallurgical studies on thorium fuel have established the physical properties. Procedures for thorium reprocessing and manufacturing thorium-based fuels have been set-up. The post-irradiation examination of irradiated thorium fuel of PHWR has given experience in handling thorium reprocessing with high ^{232}U content of about 500 ppm. Advanced fuels with thorium have been fabricated and irradiated in the test loops of CIRUS. Thermo-physical properties of thorium have been rigorously examined experimentally.

Theoretical studies on the use of thorium with plutonium or uranium in almost all types of reactors have been done. Thorium fuel cycle studies have demonstrated the superior performance of the bred ^{233}U in thermal reactors.

The Advanced Heavy Water Reactor is being designed to serve as an integral technology demonstrator for the complete thorium fuel cycle. Experience will be gained in reprocessing to handling and fabricating to irradiation and finally waste management. The design of a critical facility for AHWR is a step closer to validation of design codes. The design of high temperature reactors is another significant effort in thorium utilization in a totally different reactor realm. A steady programme for thorium fuelled high temperature reactor is underway. The technological aspects of fabrication of TRISO particles and use of fluoride salts like FLiBe are being pursued. Based on the experience of using molten salts in HTR, some work has also initiated towards a design of a molten salt reactor, which uses fuel salts of thorium and lithium fluorides.

BARC is also embarking on an accelerator programme, where the spallation neutrons from a “non-fissile or fertile” nuclei such as lead, is incident on thorium based fuel in a sub-critical configuration. The ^{233}U can be bred in such systems.

It can be said that expertise has been gained in all aspects of the thorium fuel cycle for a near term deployment of thorium based thermal reactor system.

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