THE TWIN CHALLENGES OF ABUNDANANT NUCLEAR ENERGY SUPPLY AND PROLIFERATION RISK REDUCTION - A VIEW

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1. INTRODUCTION

Large scale growth in demand for energy world wide, consequent to spread of economic development to regions deprived so far of their developmental aspirations, has necessitated the inevitable emphasis on nuclear power, as a viable source of energy, for the following reasons:

a) Need to minimise further incremental Green House Gas (GHG) emission that poses the risk of global climate change

b) Need for alternate sustainable energy source, specially to keep hydrocarbon prices under check in spite of the rising demand

The development of nuclear technology over years, particularly the collective international efforts galvanised by bodies like International Atomic Energy Agency (IAEA), World Association of Nuclear Operators (WANO), World Nuclear Association (WNA), and others have helped nuclear power evolve as a safe and economical energy source that is capable of meeting rising global energy demands, when implemented in recycle mode.

The postulated growth of nuclear power throughout the world has, however, attracted serious attention to management of spent fuel in an eco-friendly and sustainable manner, with due attention to proliferation risks. Current nuclear fuel management practices, both open cycle and closed cycle - if not managed in a responsible manner, could present inherent proliferation risks, especially with respect to bred plutonium. Whereas, the diversion of highly radioactive fresh spent fuel in a 'once through' model is grave, decay of the fission products over time increases the accessibility and retrievability of plutonium after some time. Furthermore, while there are efforts being made to evolve proliferation resistant fuel cycles which could make diversion of separated plutonium much more difficult, sustainability of nuclear energy resources would demand this to be done in a manner that allows utilisation of available fertile materials fully. Ability to manage nuclear fuel supply in a manner that is consistent with eco-friendly waste management, full exploitation of energy potential and minimisation of proliferation risks is the key to success in making nuclear power play its legitimate global role encompassing all countries that need it.

The basic objective of this paper is to compare various alternative means to burn or recycle plutonium from thermal nuclear reactors and to explore the role of thorium in this context. Utilisation of plutonium is also crucial to deploying thorium based reactor systems in countries having large thorium reserves. The main indicators for comparison, used in this study, are - amount of plutonium remaining in the spent fuel, and production of minor actinides by reactor systems.

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2. CHARACTERISTICS OF THE THORIUM CYCLE

2.1 Nuclear properties

The main nuclides associated with the thorium fuel cycle have some distinctive nuclear characteristics as compared to those for the uranium based fuel cycle. These are highlighted below:

(a) If one compares thermal neutron absorption cross section of ^{232}Th vis-a-vis ^{238}U (7.4 barns vs. 2.7 barns), ^{232}Th offers greater competition for capture (Figure-1) of the neutrons and, as such, a lower proportion of the extra neutrons will be lost to structural and other parasitic materials. This also improves conversion of fertile material in case of thorium [1].

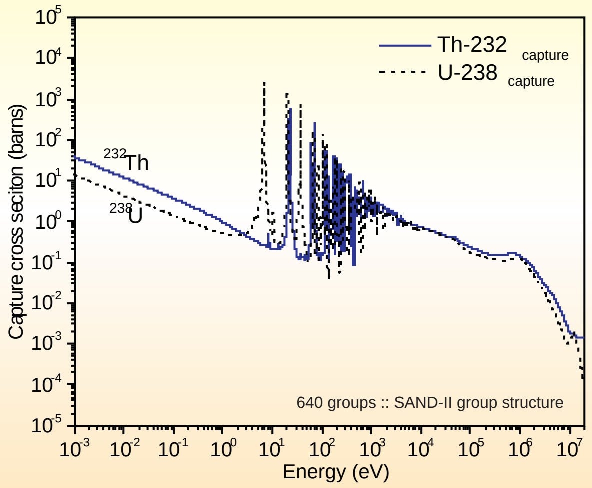

Figure-1: Comparison of capture cross section of ^{232}Th and ^{238}U

(b) ^{233}U has an eta () value greater than 2.0, which remains constant over a wide energy range, (Figure-2) in thermal as well as epithermal regions, unlike ^{235}U and ^{239}Pu. This makes the thorium fuel cycle less sensitive to the type of thermal reactor [1].

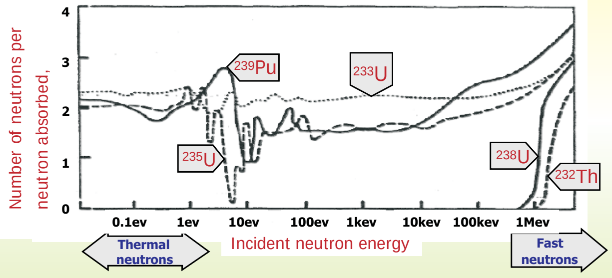

Figure-2: Variation of with respect to neutron energy

(c) The capture cross section of ^{233}U is much smaller (46 barns) than that for the other two fissile isotopes (101 barns for ^{235}U and 271 barns ^{239}Pu) for thermal neutrons, while fission cross sections (525 barns for ^{233}U, 577 barns for ^{235}U and 742 barns for ^{239}Pu) are in a comparable range. Thus, non-fissile absorption leading to higher isotopes with higher absorption cross sections (^{234}U/^{236}U and ^{240}Pu respectively) is much less probable. This makes recycle of ^{233}U less of a problem from reactivity point of view as compared to Pu [1].

(d) An important isotope that is of great relevance in the thorium cycle is ^{232}U. It 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 like ^{208}Tl (2.6 MeV) are hard gamma emitters with very small half-lives. As a result, the dose rate increases with time in the bred uranium isotopes. This radioactivity creates problems in the handling, reprocessing and recycling of bred ^{233}U and has to be properly factored in the design of process and the plant. However, it is this feature that makes thorium fuel cycle more immune to proliferation risks.

(e) Another parameter that deserves specific attention in the case of ^{232}Th is the relatively long half-life (27 days) of the intermediate product ^{233}Pa compared to the half-life (2.7 days) of the corresponding nuclide ^{239}Np in the case of ^{238}U. 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. Figure-3 shows the most prominent decay chains of these isotopes.

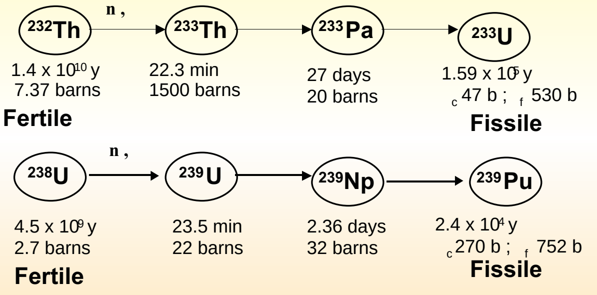

Figure-3: The most prominent decay chains of isotopes in thorium and natural uranium fuel cycle

(f) Long-lived minor actinides resulting from the burnup chain are in much less quantity for thorium fuel cycles, if the reactor operates purely in the ^{233}U -Th cycle. Actinides having masses beyond 237 are produced in negligible quantities.

2.2 Thorium as fertile host

From a fuel cycle analysis point of view, one can compare the characteristic of the two fertile hosts, viz., ^{238}U and ^{232}Th. In case of ^{235}U as the fissile feed in PHWRs, following conclusions [2,3] are arrived at:

(a) The amount of fissile content required to achieve a given burnup is higher with thorium when discharge burnups are lower. At higher discharge burnups, beyond about 50 GWd/t of Heavy Metal (HM), the initial fissile content required is lower. This is due to higher breeding with thorium system.

(b) In terms of energy extracted, i.e. fissile material utilisation, thorium as fertile host overtakes ^{238}U as host, at a burnup about 45 GWd/t of HM. This is because the bred plutonium saturates at about 0.6% at high burnups, while ^{233}U saturates to nearly 1.5%.

Similar conclusions would also be relevant for other fissile isotopes. The burn-up values of 45-50 GWd/t, though considered high in early days, are now well within the reach of current day water reactor fuel technology.

There are other advantages of using ^{232}Th that merit consideration. Variation of reactivity with burnup is smaller with thorium based fuel due to a relatively higher value of Fuel Inventory Ratio (FIR). Finally, with thorium based fuel, there is a more effective utilisation of external fissile fuel (^{235}U or Pu), added initially. This interesting feature has been relevant for thorium getting increasing attention for plutonium dispositioning.

With an operational perspective, multiple recycling of plutonium has adverse effects on reactivity coefficients in thermal reactors (LWRs), with even void reactivity becoming positive. For fuel dispositioning, employing inert matrix (IM) fuel (non-fertile metal alloys containing Pu) also leads to highly degraded reactor kinetics parameters [4]. As a result, only about one-third to one-fourth of the core can be loaded with such a fuel, bringing down the overall plutonium disposition rate via this route. Thorium, as a plutonium carrier, enables considerable improvement over both the cases.

Use of (Th, Pu) MOX in PWRs, improves plutonium burn-up, but the reactivity coefficients turn highly negative which might lead to strong feedback effects [4]. On the other hand, in a PHWR, full core can be loaded with (Th, Pu) MOX fuel without much deterioration in the safety characteristics of the core. Fraction of fissile plutonium burnt is also close to that in the case of inert matrix. This indicates superiority of heavy water moderated lattices with thorium for Pu dispositioning.

In the thorium fuel cycle, ^{232}U poses several technological challenges in the reprocessing and recycling of bred ^{233}U. On the other hand, because of the presence of this ^{232}U, the thorium is attractive from proliferation resistance point of view and is gaining worldwide attention for its use as a host for dispositioning of highly enriched weapon grade uranium and plutonium with maximum energy advantage. Another challenge is associated with the formation of ^{233}Pa, an intermediate in the formation of ^{233}U. It deteriorates the neutron balance and causes some difficulties in the reactor control. During a prolonged shutdown, there is a build up of fissile ^{233}U due to decay of ^{233}Pa, increasing the reactivity of the fuel.

3. SCOPE OF THIS STUDY

In the present study, in order to compare different options, a combination of various reactor systems (so that Pu produced in one stage is separated from the spent fuel from that stage and utilised in a subsequent stage) are analysed so as to compare these options from the point of view of:

a) **Residual fissile material including plutonium in the spent fuel of reactor in the last stage**

b) **Production of minor actinides**

The present study is based on the results derived from mainly the following two sources:

i) Data, related to options for efficient plutonium burning through the routes involving LWR-LWR(MOX), LWR-LWR(MOX)-FR(M), and LWR-FR(TRU) reactors, have been derived from reference [5]. Data for the route LWR-FR(L) have been inferred from the reference FR(M) data, as explained in Appendix-1.

ii) Data, for options highlighting the role of thorium in efficient plutonium burning, discussed for the routes comprising LWR-AHWR(L1), LWR-AHWR(L2), PHWR-AHWR(P), PHWR-PHWR(Th) and PHWR-PHWR(Th)-AHWR(LR) reactors, are based on Indian evaluations.

FIR is a ratio of the fissile inventory at a given time in the core life to the initial fuel inventory built in the core.

The designations of different reactors are explained in the next Section of this paper.

4. SPECIFIC REACTOR DESIGN CONCEPTS CONSIDERED

A brief overview of the reactors considered in the study is given below:

LWR: This reactor is based on French N4 PWR with enriched uranium oxide fuel (5 % isotopic enrichment). It has an electrical power rating of 1390 MWe. The fuel is subjected to a burnup of 50 GWd/t. This reactor is taken as a representative example of a modern LWR. [5]

LWR(MOX): This reactor is a French N4 PWR with 100% mixed oxide (MOX) fuel configuration, enriched to about 8.09% in Pu. LWR(MOX) utilises plutonium available from the spent fuel of first stage LWR reactor. Other parameters of this reactor are the same as those of above-mentioned LWR. [5]

PHWR: This reactor is the Indian Pressurised Heavy Water Reactor [6] rated at 220 MWe with natural uranium oxide fuel. The fuel of this reactor has a burnup of 6.7 GWd/t.

PHWR(Th): This reactor is based on Indian PHWR using (Th-Pu) oxide fuel. The reactor is a 220 MWe reactor using Th-Pu based MOX fuel. The fuel for this reactor has been assumed to achieve a burnup of 15 GWd/t.

A brief description of the reference design of PHWR is given in Appendix-2.

AHWR configurations: The reference design of the 300 MWe Advanced Heavy Water Reactor (AHWR) [7] caters to maximisation of power produced from thorium, using plutonium from PHWR, achieving near self-sustenance in ^{233}U at a core average discharge burn-up of 36 GWd/t (current design). A variant of this reference design, with a burn-up of 50 GWd/t and with capability to use plutonium quality (based on isotopic composition) discharged from LWR is termed AHWR(LR). This variant, based on the optimised reference case, has a deficit in the self-sufficiency in ^{233}U, as indicated in Table-1. This deficit can be met by several means. Out of these, the following two possibilities have been considered in this study:

(a) Obtaining the required ^{233}U make-up from a PHWR(Th) reactor.

(b) Enhancing the plutonium content in the fuel to achieve self-sufficiency in ^{233}U, though at a cost of some degree of reduction in power from thorium.

The scheme adopted to use the ^{233}U make up from PHWR(Th), showing the required mass flows (in kg), is given in Figure-4. In this scheme, Pu from PHWR(Th) is utilised for blending PHWR plutonium.

The AHWR design with self-sufficiency in ^{233}U, based on LWR discharged plutonium and having a core average discharge burnup of 50 GWd/t is designated as AHWR(L1). Another variant, using LWR quality Pu, and having a burnup of 36 GWd/t is designated AHWR(L2). Based on a similar approach, AHWR designs utilising PHWR discharged plutonium, while

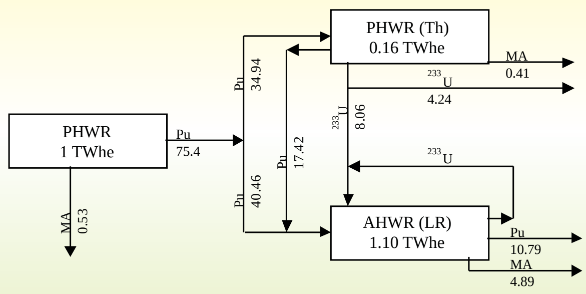

Figure-4: Synergistic scheme with utilisation of Pu from PHWR(Th) to provide make-up ^{233}U to AHWR(LR) being self-sufficient in ^{233}U, and having burnups of 50 GWd/t and 32 GWd/t are designated as AHWR(P1) and AHWR(P2) respectively. The important characteristics of material flows for these variants of AHWR, arrived at on the basis of several iterations, are included in Table-1. The residual mismatches (deficit in the AHWR(L1) case and surplus in the AHWR(L2), AHWR(P1) and AHWR(P2) cases) of ^{233}U are inconsequential for the purpose of this study, and have been ignored.

A brief description of the current reference design of AHWR is given in Appendix-3.

Fast Reactor (FR) configurations:

The Fast Reactor case is directly taken from reference [5], where the combination LWR-LWR(MOX)-Fast Reactor is described as a plutonium burner option. The specific case described in the reference is based on a fast reactor configuration that has full plutonium recycle together with top-up from LWR(MOX). In this study, this fast reactor is designated as FR(M).

This reactor is based on CAPRA type Pu burner fast reactor utilising MOX fuel. It has a low conversion ratio, and high burnup core. The plutonium is recycled. The plutonium to uranium ratio of the top-up fuel is adjusted to obtain keff=1 at EOEC (End of Equilibrium Cycle) [5]. This reactor is taken as a representative example of a fast reactor design for plutonium burning. It has a burnup of 185 GWd/t of heavy metal and uses MOX fuel (44.5% Pu).

There is no case described in [5] that corresponds to a Fast Reactor making use of LWR plutonium directly. Hence, making use of a concept of "Equivalent plutonium enrichment (EPE)", the plutonium material flows have been adjusted to yield an appropriate FR configuration, designated FR(L). The details of this approach are provided in Appendix-1.

Reference [5] also refers to another fast reactor design, as critical TRU burner. In this study, this type of fast reactor is designated FR(TRU). This is an ALMR type fast reactor with metallic fuel and capable of burning minor actinides. TRU feed consists of TRU discharged from LWR and uranium is obtained from depleted uranium discharged from LWR [5].

A summary of the important parameters of reactor concepts considered in the study is given in Table-1. The data pertaining to materials flow is normalised for 1 TWhe energy production.

Table-1: Data for reactors used in the present study (normalised to 1 TWh electricity production)

Reactor	Power, Thermal/ Electric (MWth/ MWe)	Enrichment %	Burnup GWd/t	Uranium kg		Plutonium kg		Thorium kg		Minor actinides kg
				In [Total (Fissile[1])]	Out [Total (Fissile[1])]	In	Out	In	Out	Out
LWR	4020/1390	5 (isotopic)[*]	50	2394[6] (119.7)	2238 (18.35)	0	28.75	-	-	3.57
LWR(MOX)	4020/1390	8.09 Pu[2,*]	50	2204 (6.61)	2126 (2.66)	194	132	-	-	17.74
AHWR(LR)	920/300	4.25[3] - ^{233}U 4.60[4] - Pu	50	60.3[7] (60.3)	56.1 (46.9) (45.3 ^{233}U)	52.5	9.79	2443	2354	4.44
AHWR(L1)	920/300	5.8 Pu (Inner) [9,10] 5.25 ^{233}U (Middle)[9,11] 5.8 Pu (Outer)[9,10]	50	44.6[7] (44.6)	52.9 (44.9) (43.5 ^{233}U)	99.3	25.1	2412	2325	7.86
AHWR(L2)	920/300	2.9 ^{233}U (Inner)[9,11] 2.9 ^{233}U (Middle)[9,11] 5.25 Pu (Outer)[9,10]	36	57.2[7] (57.2)	69.0 (60.7) (59.4 ^{233}U)	83.1	23.7	3409	3320	5.47
AHWR(P1)	920/300	4.75 Pu (Inner)[9,10] 5.0 ^{233}U (Middle)[9,11] 4.75 Pu (Outer)[9,10]	50	42.6[7] (42.6)	51.6 (43.6) (42.2 ^{233}U)	80.9	16.43	2432	2342	3.88
AHWR(P2)	920/300	See note 12 below	32	63.2[7] (63.2)	75.3 (66.6) (65.4 ^{233}U)	72.1	20.6	3858	3764	2.49
PHWR	750/220	0.72 (isotopic)[7]	6.7	20730[8] (149.4)	20512 (53.33)	0	75.4	-	-	0.53
PHWR(Th)	750/220	2.40 Pu[5]	15	0	83.6 (78.9) (78.5 ^{233}U)	223	111.2	9036	8922	2.59
FR(L)	3600/1450	44.5 Pu[2]	185	350.5 (0.5)	306.9 (0.41)	280.3	212.3	-	-	17.94
FR(M)	3600/1450	44.5 Pu[2]	185	309.5 (0.5)	259.5 (0.35)	248	180	-	-	15.17
FR(TRU)	1575/600	35.2(isotopic)[*]	139	523.1 (0.653)	468.9 (0.49)	227.7	178.7	-	-	30.7 (In) 23.9 (Out)

[*] Calculated using data provided in [5]

[1] Fissile U = ^{235}U + ^{233}U

[2] Enrichment = Pu/(Pu + U)

[3] Enrichment = ^{233}U/(^{233}U + Th) : Inner 30 Pins of AHWR fuel cluster

[4] Enrichment = Pu/(Pu + Th) : Outer 24 Pins of AHWR fuel cluster

ieemoare based on ratios of weights of fuels in oxide form

[5] Enrichment = Pu/(Pu + Th)

[6] Enriched uranium. Corresponding natural uranium requirement estimated as 20,513 kg. [5]

[7] ^{233}U

[8] Natural uranium

[9] Inner, Middle and Outer refers to the inner 12, middle 18 and the outer 24 pins of the AHWR fuel cluster

[10] Pu Enrichment = Pu/(Pu + Th)

[11] ^{233}U Enrichment= ^{233}U/(^{233}U + Th)

All enrichment estimatioare based on ratios of weights of fuels in oxide form

[12] Two different clusters are used. In cluster1, the enrichment values are: 3% U (Inner), 3.75% U (Middle) and 3.25% Pu (Outer). In cluster 2, these values are: 3.25% Pu Inner, 3.75% U Middle and 3.25% Pu Outer)

5. CASES OF DIFFERENT REACTOR COMBINATIONS FOR Pu DISPOSITION

Under this study, eleven cases of reactor combinations were evaluated as different options for Pu disposition. A definition of these cases is provided in Table-2. A pictorial depiction of the cases is given in Figure-5. In these cases, the Pu contained in the spent fuel of either LWR or PHWR is separated out and used in subsequent stages.

The minor actinides (^{231}Pa, ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, and ^{245}Cm) from the spent fuel are sent as waste for further processing and disposal for all the cases except case-VI, where they are recycled.

Table-2: Definition of the cases

Case No	Stage-1	Stage-2	Stage-3
I*	LWR	LWR (MOX)	-
II	LWR	AHWR (L1)	-
III	LWR	AHWR (L2)	-
IV*	LWR	LWR (MOX)	FR (M)
V*	LWR	FR (L)	-
VI*	LWR	FR (TRU)	-
VII	PHWR	PHWR (Th)	-
VIII	PHWR	PHWR (Th)	AHWR (LR)
IX	PHWR	AHWR (P1)	-
X	PHWR	AHWR (P2)	-
* Results of these cases were inferred from [5]			

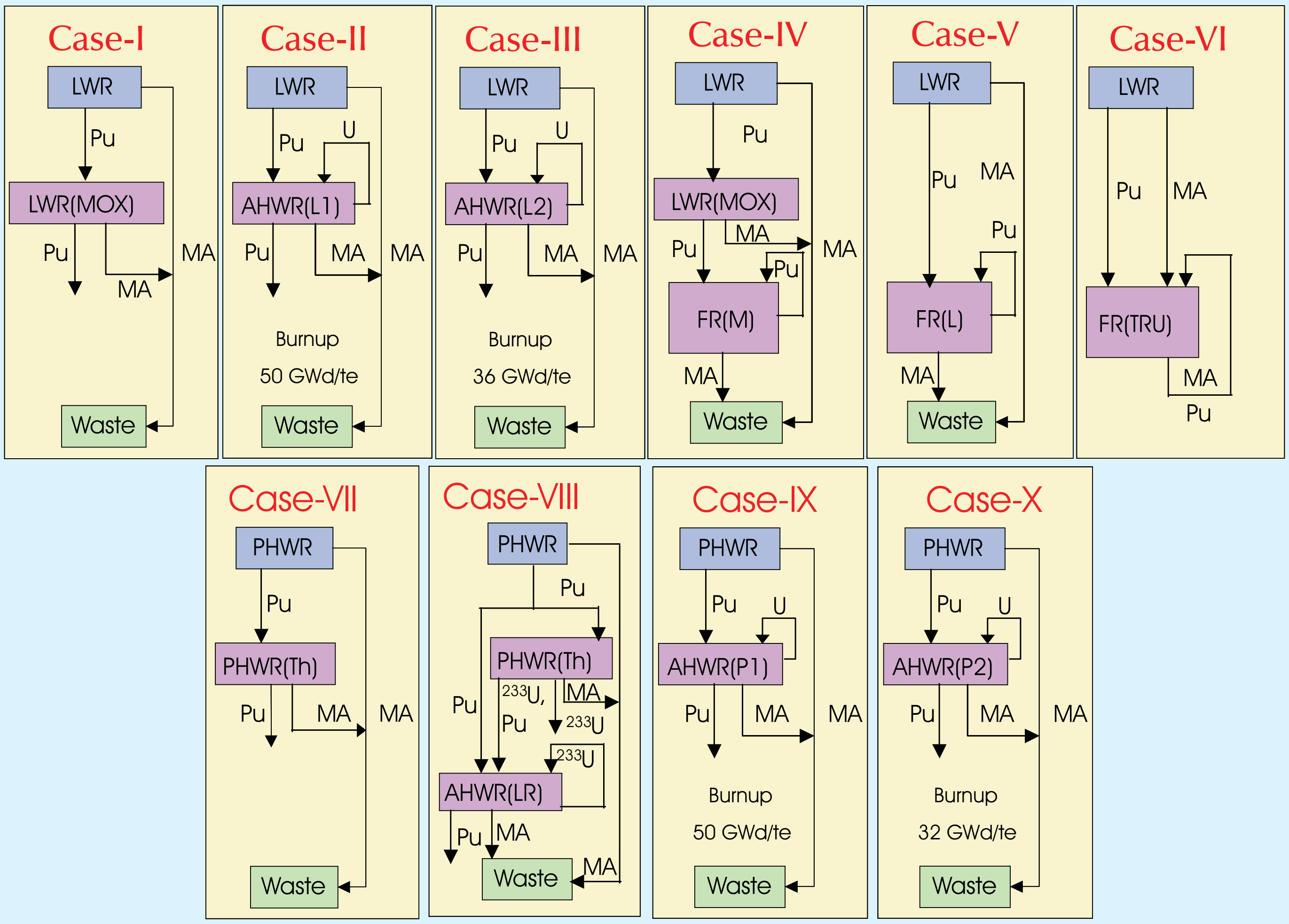

Figure-5: Schematic diagram of the cases analysed

6. THE ASSUMPTIONS AND THE METHODOLOGIES FOR ANALYSES

6.1 Assumptions

The following assumptions have been made while carrying out these studies;

a) No reprocessing losses in the entire fuel cycle have been considered. Some losses during reprocessing and fuel refabrication are expected, but are neglected in the present calculations for simplicity.

b) Changes in isotopic concentration of spent fuel during cooling have been neglected. However, two exceptions with respect to ^{233}Pa and ^{239}Np have been made as mentioned in paragraphs c) and d) below.

c) The amount of ^{233}U taken is the sum of the ^{233}U and ^{233}Pa contents. ^{233}Pa has a half life of 27 days. Hence, it will decay rapidly into ^{233}U.

d) The amount of ^{239}Pu taken is the sum of ^{239}Pu and ^{239}Np contents. ^{239}Np has a half life of 2.35 days. Hence, it will decay very rapidly into ^{239}Pu.

e) The plutonium in the fresh fuel of the fast reactor (FR(M)) is composed partly of the plutonium recycled from its spent fuel and the rest of the plutonium is obtained from the spent fuel of the previous stage LWR(MOX) reactor. Depleted uranium has been assumed to have 0.3% ^{235}U.

6.2 Methodology

In each of the cases listed in Table-2, the energy produced in Stage-1 reactor is taken as 1 TWhe. The Stage-2 reactor utilises plutonium output from the Stage-1 reactor. Plutonium output from Stage-1 is matched to the plutonium input requirement of the subsequent stage(s). Using the above-mentioned assumptions and normalised quantities given in Table-1 for 1 TWhe energy production, required parameters have been evaluated for comparison of options.

7. RESULTS

The material flows for each of the options listed in Table-2 are analysed and are provided in Table-3.

Table-3: Summary of results

Case no.	Case description	Energy produced by burning plutonium from first stage (TWhe)	Minor Actinides in spent fuel (kg)			Fissile amount in spent fuel of last stage				
						^{233}U (kg)	Total Pu		^{235}U (kg)	Fissile Total (kg)
			2nd Stage	3rd Stage	Total		Amount (kg)	% Fissile[5]		
I	LWR-LWR(MOX)	0.148	2.63	-	2.63	-	19.6	51.1	0.394	10.4
II	LWR-AHWR(L1)	0.290	2.28	-	2.28	12.6[1]	7.30	22.1	0.415	14.6
III	LWR-AHWR(L2)	0.503	2.75	-	2.75	29.9[2]	11.3	20.5	0.635	33.0
IV	LWR-LWR(MOX)-FR(M)	0.148/0.287 (Total: 0.435)	2.63	4.35	6.98	-	51.5[6]	34.1	0.101	17.7
V	LWR-FR(L)	0.423	7.58	-	7.58	-	89.8[6]	34.1	30.6	61.2
VI	LWR-FR(TRU)	0.582	14.1	-	14.1	-	104[6]	43.8	0.285	45.9[6]
VII	PHWR-PHWR(Th)	0.339	0.88	-	0.88	26.6	37.7	40.3	0.120	41.9
VIII	PHWR-PHWR(Th)-AHWR (LR)	0.16/1.1 (Total: 1.26)	0.41	4.89	5.30	49.9[6]	10.8	15.7	1.76	53.4
IX	PHWR-AHWR(P1)	0.932	3.62	-	3.62	39.3[3]	15.3	25.5	1.33	44.5
X	PHWR-AHWR(P2)	1.05	2.60	-	2.60	68.4[4]	21.6	31.2	1.32	76.4

[1] ^{233}U in fresh fuel = 12.9 kg
[2] ^{233}U in fresh fuel = 28.8 kg
[3] ^{233}U in fresh fuel = 39.7 kg
[4] ^{233}U in fresh fuel = 66.1 kg
[5] Fissile Pu = ^{239}Pu + ^{241}Pu
[6] Recycled

8. DISCUSSION

The results given in Table-3 are discussed in the following paragraphs.

8.1 Comparison of fast reactors and AHWR for incineration of LWR plutonium

A direct comparison of FR and AHWR (L) variants for plutonium incineration, on the basis of data given in Table-3, has been done.

Three fast reactor based cases, LWR(MOX)-FR(M), FR(L), and FR(TRU) have been compared with AHWR(L1) and AHWR(L2) cases. The comparison is based on plutonium present in the spent fuel of last stage reactor as well as minor actinides produced. The amount of plutonium present in the spent fuel of the last stage indicates the burden on reprocessing plutonium. This adds to cost, as well as proliferation potential. A lower value of this parameter is also an indicator of the effectiveness of the reactor(s) in efficiently using the fissile feed in fewer number of reprocessing cycles, if any. On the other hand, the amount of minor actinides produced indicates the burden on cost intensive routes of their incineration through a special fast reactor (including FR(TRU)) or ADS based incineration system. Figure-6 shows comparisons of the amount of total and fissile plutonium in the spent fuel of the last stage reactor. Figure-7 gives a comparison of amount of minor actinides produced.

Figure-6 : Comparison of plutonium in the spent fuel of last stage reactor for AHWR(L) and FR based options

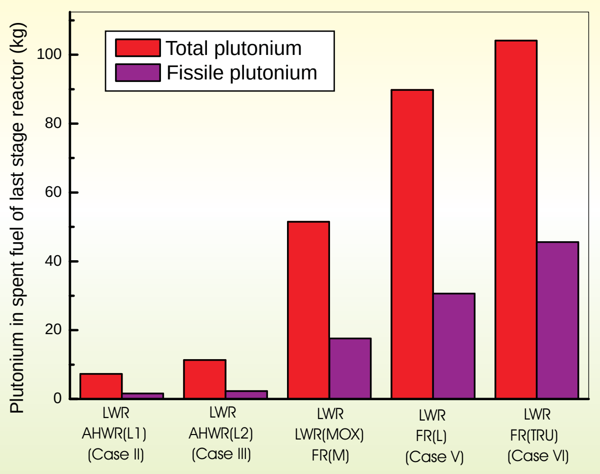

Figure-6: Comparison of plutonium in the spent fuel of last stage reactor for AHWR(L) and FR based options

Figure-7 : Comparison of MA produced Figure-6 and Table-3 show that AHWR(L) options produce about 7.3-11.3 kg total Pu (with about 20-22% fissile content) per TWhe of LWR Pu, as compared to 51-104 kg total Pu (with about 34%-44% fissile content) produced by fast reactor options. Therefore, incineration of plutonium in AHWRs is more effective in both quantity and quality. Additionally, AHWR route produces almost three times less MA as compared to a fast reactor option. Hence, in an overall sense, use of AHWRs offer an efficient and cost effective way for incinerating LWR plutonium.

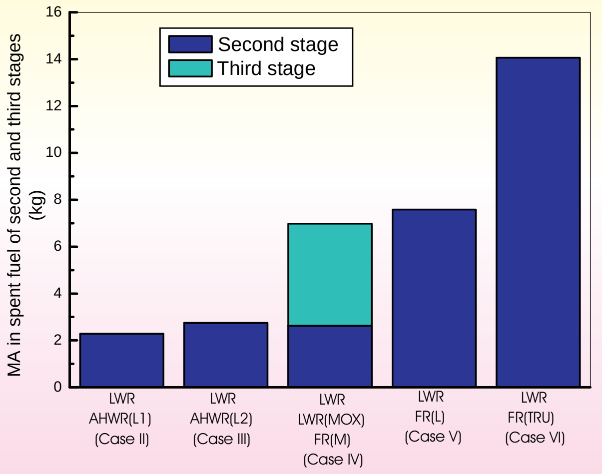

Figure-7: Comparison of MA produced

8.2 Comparison of options of incineration of PHWR plutonium

For incineration of plutonium obtained from PHWRs, four cases, PHWR (Th), AHWR(P1), AHWR(P2), including one case of PHWR(Th)-AHWR(LR) route, have been compared.

On comparison (Table-3 and Figure-8) of the plutonium left in the spent fuel of last stage reactor, it is clear that among the options for burning plutonium from the PHWR spent fuel, option PHWR(Th)-AHWR(LR) produces minimum (15.7% fissile Pu of total 10.8 kg Pu) left-over Pu in both quality and quantity. This is much lower as compared to 40.3% fissile Pu (of 37.7 kg total Pu) for the case with PHWR(Th). It can be seen that the fissile Pu content in the spent fuel of AHWR(P1) and AHWR(P2) is also

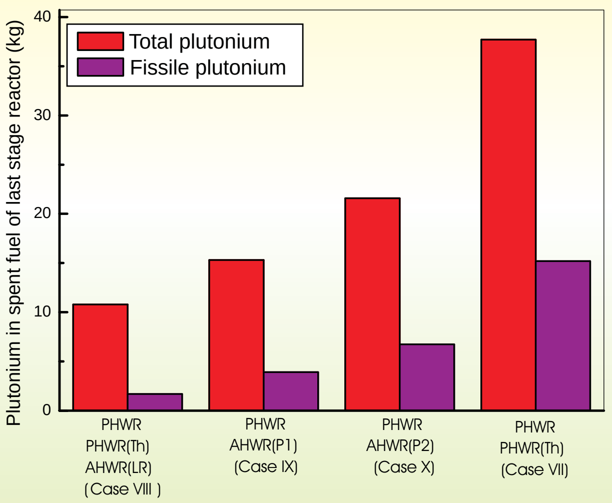

Figure-8: Total and fissile Pu produced

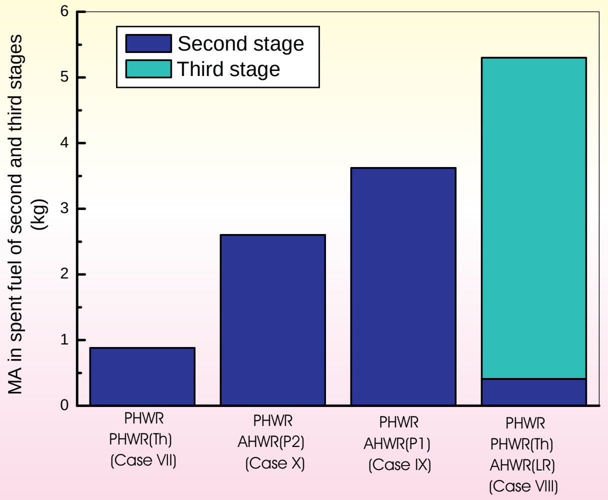

Figure-9: Comparison of MA produced

low and is about 25.5% (of 15.3 kg total Pu) and 31.2%(of 1.32 kg total Pu) respectively.

Figure-8 : Total and fissile Pu produced On comparing the MA produced by reactors subsequent to Stage-1 (Table-3 and Figure-9), it is seen that MA production is minimum for PHWR(Th) route and is almost 3-4 times less as that produced by AHWR(P1) and AHWR(P2) options. As is evident from Figure-9, use of PHWR(Th)-AHWR(LR) option will produce the maximum amount of minor actinides, almost six times that of PHWR(Th) route.

8.3 Consideration of ^{233}U in the spent fuel

From the above discussions, considering the total quantity of fissile Pu and minor actinides in the spent fuel of the last stage reactors, it is clear that the use of thorium based reactors offer the best results from proliferation resistance consideration and reduced minor actinide loads. With reference to Table-3, it can be seen that the total fissile content of the spent fuel for thorium based reactors is high, a large portion (80-95%) of which is composed of ^{233}U. The ^{233}U in the spent fuel of the different thorium fuelled options is associated with high radioactivity, mainly arising out of the daughter products of ^{232}U, as is indicated in Table-4.

The ^{232}U concentration values have been computed by BARC using the latest nuclear data, and have been validated through actual measurements on the thorium bundles irradiated in PHWRs. The results presented in Table-4 indicate that a near lethal radiation dose can be quickly received by anyone handling a significant quantity of unshielded ^{233}U. Indeed, the ^{232}U concentrations indicated above are based on only one cycle of operation. With multiple recycling of ^{233}U, these concentrations

Table-4: The gamma radiation exposure rate from 5 kg ^{233}U removed after one cycle

Reactor type	^{232}U concentration (ppm)	Gamma radiation exposure rate at 1 ft. distance (R/h)		
		After 1 year	After 10 years	After 100 years
AHWR (L1)	2368	355	1089	474
AHWR (L2)	1468	220	676	294
AHWR (P1)	2428	364	1116	485
AHWR (P2)	1289	193	593	258
AHWR (LR)	2107	316	970	422
PHWR (Th)	816	123	378	163

will increase further. There is, thus, enough scientific basis to discard a real possibility of diversion of ^{233}U available in the spent fuel of any of the thorium based reactors for building a weapon. In any case, such diversion would be extremely dangerous and unattractive, to say the least.

9. CONCLUSIONS

The conclusions arising out of this study clearly establish the vast superiority of a suitably optimised thorium fuelled reactor over a fast reactor, for an application aiming to incinerate plutonium contained in the spent fuel of LWRs. and PHWRs, effectively and efficiently. This superiority is further maintained when an additional parameter, viz., production of minor actinides is considered. The necessary multiple recycling in Fast Reactors is associated with a need for establishing associated reprocessing facilities, with attendant cost and proliferation risks. The higher minor actinide burden also involves setting up a larger number of advanced systems, in the future, for their incineration along with their higher associated costs, longer time frames of deployment, and attendant economical and technical risks of a relatively immature technology, as compared to conventional thermal reactors.

Thorium can be utilised in reactor designs that already exist, or new systems can be designed to utilise thorium using existing technologies. In any event, proliferation resistant designs using thorium are easily realisable. Thus, the implementation of thorium based options in thermal reactors of generally familiar designs is quite feasible, even in the near term, on account of the vast experience already available for the conventional thermal reactors. However, such choice does not exist for fast reactors.

Out of the current fleet of 443 nuclear power reactors operating in the world, less than half are under IAEA Safeguards, and only about ten percent of these safeguarded ones are located outside the OECD countries. Even in this scenario, and with a very slow growth of nuclear power in the last two decades, the volume of human and financial resources needed for the implementation of IAEA safeguards have constituted a large fraction of the resources available to the Agency. With the envisaged rapid growth in the demand for nuclear power, mainly in the developing countries, the ability to implement safeguards in the traditional manner could, itself, become a serious limiting factor, and perhaps a hindrance to such growth. It is, therefore, necessary to establish institutional as well as technological solutions that should enhance proliferation resistance along with an assured fuel supply, without adversely affecting long-term sustainability of nuclear fuel resources. Thorium offers a very important and attractive solution from this perspective.

Over the years, India has developed advanced capability in the utilisation of thorium, as a part of its strategy to enhance its nuclear capacity through a closed nuclear fuel cycle that would enable timely deployment of its large thorium reserves. Indeed, India is today the only country in the world operating a research reactor fuelled by Uranium-233. India has already completed the design and development of its mainly thorium fuelled Advanced Heavy Water Reactor, that has undergone peer reviews as well as pre-licencing safety appraisal by the Indian Atomic Energy Regulatory Board.

10. ACKNOWLEDGEMENT

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12. LIST OF ABBREVIATIONS USED IN THE PAPER

ADS	Accelerator Driven System
Adv.	Advanced
AERB	Atomic Energy Regulatory Board
AHWR	Advanced Heavy Water Reactor
ALMR	Advanced Liquid Metal Reactor
Am	Americium
CAPRA	Consommation Accrue de Plutonium en Réacteur rApide
Cm	Curium
EPE	Effective Plutonium Enrichment
FBR	Fast Breeder Reactor
FIR	Fuel Inventory Ratio
FR	Fast Reactor
GHG	Green House Gas
GWd/t	Giga-Watt-Days per Tonne
HM	Heavy Metal
IAEA	International Atomic Energy Agency
IM	Inert Matrix
LWR	Light Water Reactor
MA	Minor Actinides MeV
MeV	Million-electron-Volt
MOX	Mixed Oxide
MWe	Mega-Watt-electric
MWth	Mega-Watt-thermal
Np	Neptunium
NU	Natural Uranium
Pa	Protactinium
PHWR	Pressurised Heavy Water Reactor
Pu	Plutonium
PWR	Pressurised Water Reactor
Th	Thorium
Tl	Thallium
TRU	Transuranic
TWhe	Terra-Watt-hour-electric
U	Uranium
UOX	Uranium Oxide
WANO	World Association of Nuclear Operators
WNA	World Nuclear Association

CALCULATIONS FOR LWR-FR(L) USING EFFECTIVE PU ENRICHMENT AT INLET AND MOX ENRICHMENT EQUALISATION

The following assumptions are made for these calculations :

a) LWR Pu (contained in spent fuel) is isotopically richer than the Pu contained in LWR MOX spent fuel. Thus, lesser input Pu mass is required to achieve the same effective Pu enrichment (EPE). Using EPE at inlet of the FR as a constraint we can evaluate the required mass of LWR Pu. Table A1-1 below gives the percentage share of each isotope in the spent fuel of respective reactor.

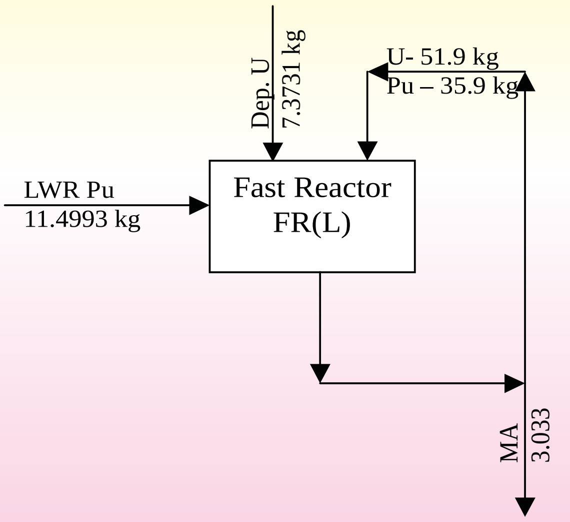
b)The EPE at the inlet to FR is maintained.

Table A1-1: Isotopic composition of spent fuel [5]

Isotope	LWR(MOX)	LWR	FR(M)
^{238}Pu	5.0466	3.4988	2.4485
^{239}Pu	37.9088	51.8728	27.9076
^{240}Pu	30.3142	23.8132	43.8230
^{241}Pu	13.2128	12.9065	6.1770
^{242}Pu	13.5175	7.9087	19.6438
Effective Pu Enrichment (EPE)*	64.2271	75.9595	43.7613

* Calculated using the formula [9]:
EPE= ^{238}Pu*0.58+^{239}Pu*1.0+ ^{240}Pu*0.1 + ^{241}Pu*1.5+ ^{242}Pu*0.04)

Figure-A1-1: Material flows for the Fast Reactor FR(L) case



However, the individual fractions of ^{239}Pu and ^{241}Pu (see table above) are not maintained. It is assumed that this difference does not affect the FR(L) characteristics sigificantly. Figure A1-1 depicts the mass flow diagram for this case.

c) The energy component is assumed to be proportional to the mass of Pu consumed in one cycle. The energy component of FR (L) reduces to 0.16911 TWhe. The mass flow calculations normalised for 1 TWhe energy production is given in table A1-2. Table A1-3 shows energy produced from 28.75 kg Pu in the spent fuel from LWR and associated MA and Pu discharges

Table A1-2: Material flow (normalised for 1 TWhe energy generated in FR(L))

Reactor	Power, Thermal/ Electric (MWth/ MWe)	Enrichment (%)	Burnup (GWd/t)	Uranium (kg)		Plutonium (kg)		Thorium (kg)		Minor actinides (kg)
				In	Out	In	Out	In	Out	Out
FR(L)	3600/1450	44.5 Pu[1]	185	350.505	306.906	280.291	212.291	-	-	17.935

[1] Enrichment = Pu/(Pu + U)

Table A1-3: Energy produced from 28.75 kg Pu from LWR and associated MA and Pu

Case no.	Case description	Energy produced by burning plutonium from first stage (TWhe)	Minor Actinides in spent fuel (kg)			Fissile amount in spent fuel of last stage			
							Total Pu		
			2nd Stage	3rd Stage	Total	^{233}U (kg)	Amount (kg)	% Fissile	Fissile Total (kg)
I	LWR-FR(L)	0.4228	7.583	-	7.583	-	89.757	34.1	30.607

Appendix - 2

PRESSURISED HEAVY WATER REACTOR

The design parameters of an Indian PHWR are given in Table-A2-1 and a schematic of primary heat transport system of the reactor is shown in Figure-A2-1.

Table-A2-1: Design features of Indian standard PHWR – 220 MWe (operating)

Rated power output (thermal)	756 MWt
Rated power output (electrical)	220 MWe
Type	Horizontal pressure tube
Fuel	Natural UO_2
Moderator and reflector	Heavy water
Coolant	Heavy water
Channel inlet/ outlet temperatures	249/293 C
Pressure in outlet header	8.5 MPa

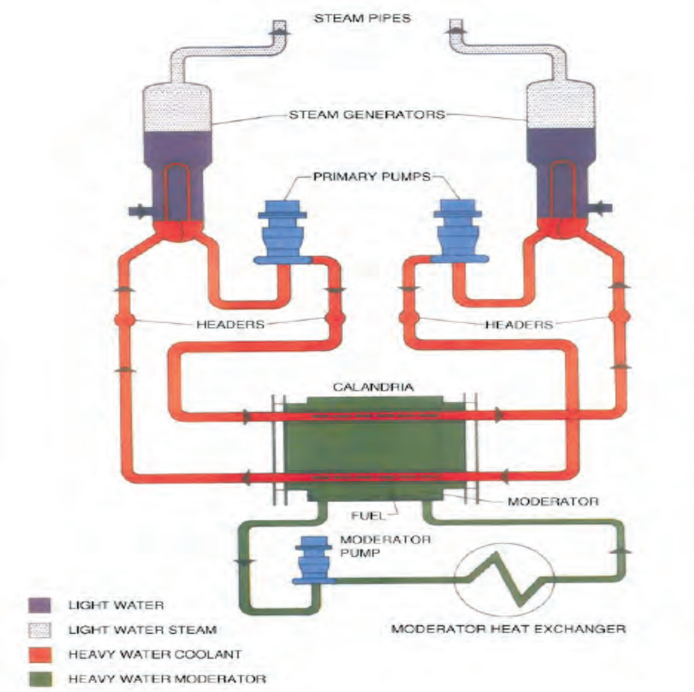


Figure-A2-1: Schematic of primary heat transport system of an Indian PHWR

The main features of PHWR are:

a) Use of natural uranium as fuel, which obviates the need for developing fuel enrichment facilities.

b) High neutron economy made possible by use of heavy water as moderator, which means low requirements of natural uranium both for initial core as well as for subsequent refueling. Further, fissile plutonium production (required for fast reactors in stage 2 of Indian nuclear power programme) is high, compared to Light Water Reactors.

c) Being a pressure-tube reactor, with no high-pressure reactor vessel, the required fabrication technologies are easier.

d) The technology for production of heavy water, required as moderator and coolant in PHWR, are available in the country.

More details on Indian PHWR can be found in reference [6]

ADVANCED HEAVY WATER REACTOR

The AHWR is a 300 MWe, vertical, pressure tube type, heavy water moderated, boiling light water in natural circulation cooled reactor. The fuel consists of (ThPu)O2 and (Th233U)O2 pins. The fuel cluster is designed to generate maximum energy out of 233U, which is bred in-situ from thorium and has a slightly negative void coefficient of reactivity. For the AHWR, the well-proven pressure tube technology has been adopted and many passive safety features, consistent with the international trend, have been incorporated. The design of the reactor is based on the feedbacks from the extensive analytical and experimental R&D. Important design parameters of AHWR are shown in Table-A3-1 and schematic of AHWR is shown in Figure-A3-1.

Table A3-1: Important design parameters of AHWR core

Reactor power	920 MWth, 300 MW_e
Core configuration	Vertical, pressure tube type design
Coolant	Boiling light water
Number of coolant channels	452
Pressure tube ID	120 mm
Lattice pitch	225 mm (square pitch)
No. of pins in fuel cluster	54 [$(Th\text{-}Pu)O_2$ - 24 pins, $(Th\text{-}^{233}U)O_2$ - 30 pins]
Active fuel length	3.5 m
Total core flow rate	2230 kg/s
Coolant inlet temperature	259 °C (nominal)
Feed water temperature	130 °C
Average steam quality	18.6 %
Steam generation rate	414.4 kg/s
Steam drum pressure	70 bar
MHT loop height	39 m
Primary shut down system	40 shut off rods
Secondary shut down system	Liquid poison injection in moderator
No. of control rods	13

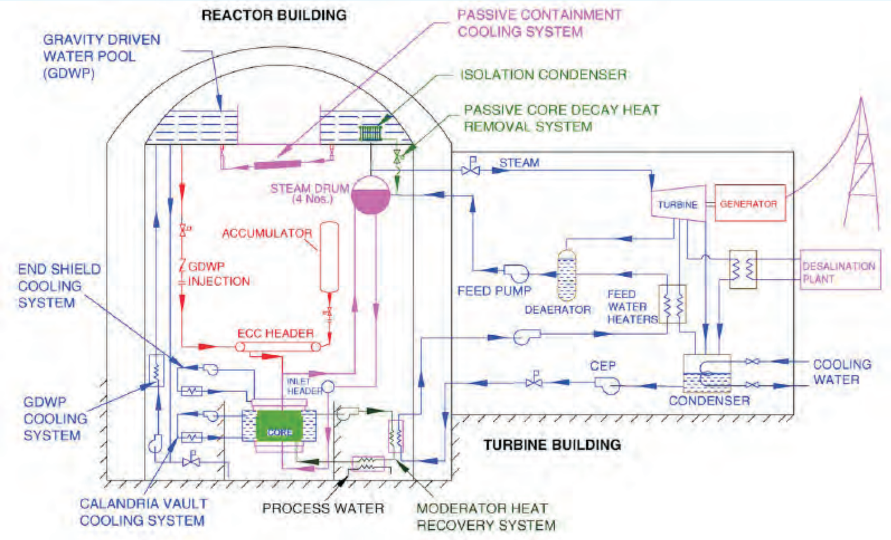


Figure-A3-1: Schematic of Advanced Heavy Water Reactor

The important design features of AHWR are;

(a) Elimination of high-pressure heavy water coolant resulting in reduction of heavy water leakage losses, and eliminating heavy water recovery system.
(b) Recovery of heat generated in the moderator for feed water heating.
(c) Elimination of major components and equipment such as primary coolant pumps and drive motors, associated control and power supply equipment and corresponding saving of electrical power required to run these pumps.
(d) Shop assembled coolant channels, with features to enable quick replacement of pressure tube alone, without affecting other installed channel components.
(e) Replacement of steam generators by simpler steam drums.
(f) Higher steam pressure than in PHWRs.
(g) Production of 500 m3/day of demineralised water in Multi Effect Desalination Plant by using steam from LP Turbine.
(h) 100 year design life of the reactor.

The AHWR has several passive safety systems for reactor normal operation, decay heat removal, emergency core cooling, confinement of radioactivity etc. These passive safety features are listed below:

a) Core heat removal by natural circulation of coolant during normal operation and shutdown conditions;
b) Direct injection of Emergency Core Cooling System (ECCS) water in the fuel cluster in passive mode during postulated accident conditions like Loss of Coolant Accident (LOCA);
c) Containment cooling by passive containment coolers;
d) Passive containment isolation by water seal, following a large break LOCA;
e) Availability of large inventory of water in Gravity Driven Water Pool (GDWP) at higher elevation inside the containment to facilitate sustenance of core decay heat removal, ECCS injection, containment cooling for at least 72 hours without invoking any active systems or operator action;
f) Passive shutdown by poison injection in the moderator, using the system pressure, in case of main heat transfer system high pressure due to failure of wired mechanical shutdown system and liquid poison injection system;
g) Passive moderator cooling system to minimise the pressurisation of calandria and release of tritium through cover gas during shutdown and station blackout;
h) Passive concrete cooling system for protection of the concrete structure in high-temperature zone.

Some more details of AHWR are given in reference [7]

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