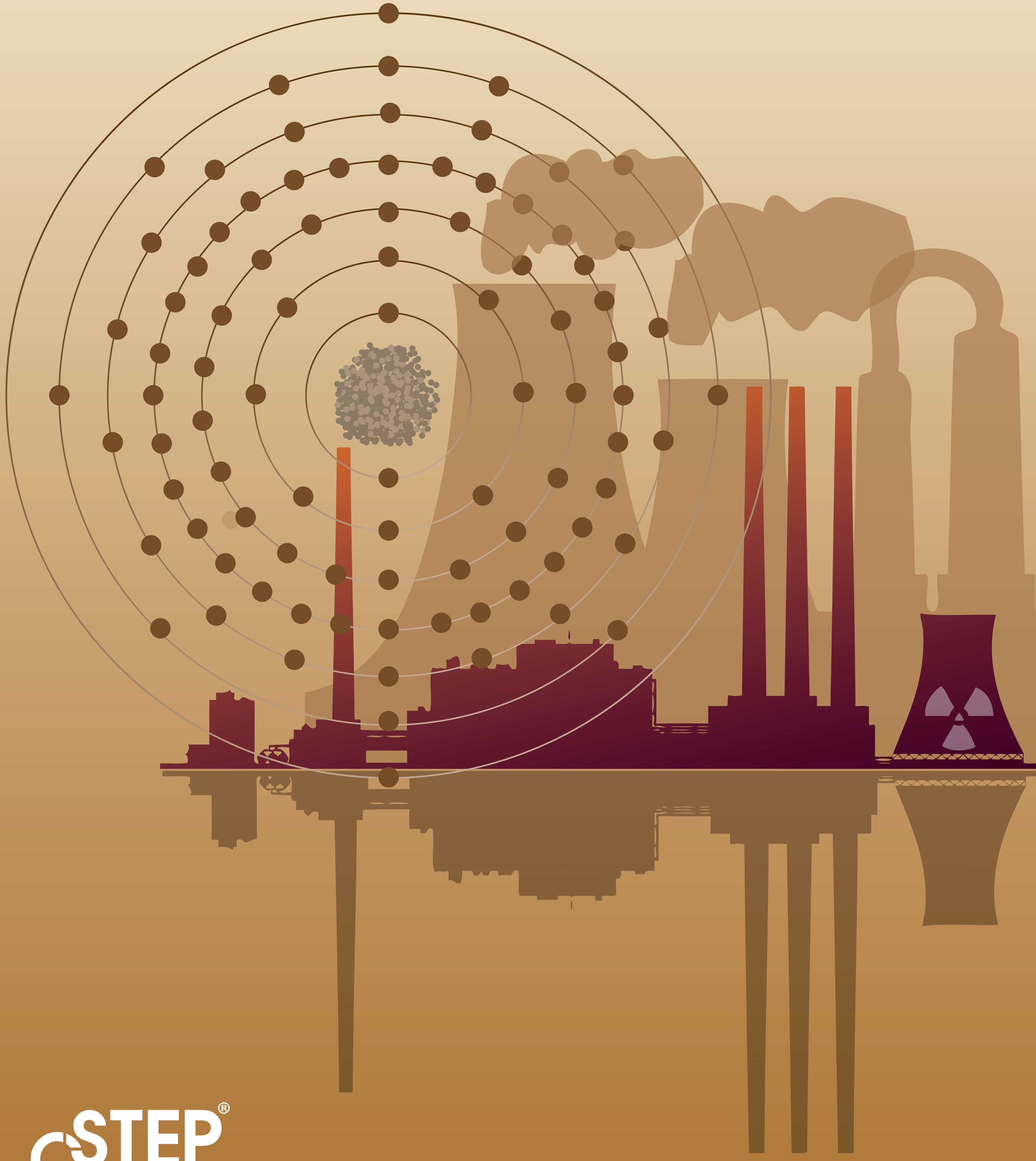
Thorium-Utilisation Pathways for India



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Thorium-Utilisation Pathways for India

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Executive Summary

Worldwide, the interest in thorium has revived in recent years, as a means to safer nuclear energy. For India, thorium-based nuclear power has been the endgame, and will form the final stage of the three-stage Indian nuclear programme for energy security developed by Dr Homi J. Bhabha in the 1950's. Despite consistent emphasis on this, there is a lack of clarity on how or by when thorium will be utilised on a large scale. In this context, we felt that it is relevant to assess strategies for thorium adoption. We explored alternate pathways and opportunities for early adoption of thorium in this report.

Thorium-based fuel cycle is adaptable to most reactor systems, and there are several options that are relevant to India. However, database and operational experience on thorium fuels and fuel cycles are limited, compared to uranium- and plutonium-based fuel cycles. An early introduction of thorium in currently operational reactors would be beneficial for establishing expertise in the various aspects of fuel cycle as well as data for validation of design codes.

Reprocessing spent fuel is a key to realising most options. For our analysis, we have considered only the plutonium build-up from the existing and planned reprocessing capacity. However, this capacity is insufficient to realise the plutonium worth from the continuously accumulating spent fuel from the operational and planned reactors. The development of spent-fuel reprocessing of thorium-based fuel—Thorex—is still in the early stages and needs extensive research and development (R&D) prior to achieving industrial status. Strengthening of the thorium fuels database is imperative for better development of industrial-scale fuel reprocessing.

The Department of Atomic Energy (DAE) has consistently emphasised that large-scale use of thorium will take place decades after the commercial establishment of the fast breeder reactor (FBR) stage. The main takeaway from our analysis is that it is possible to considerably advance the thorium stage if the spent fuel reprocessing of thorium-based fuel and a Th – U-233 breeder design are ready for deployment before we enter the FBR stage on a big scale. In such a situation, we feel that the requisite research and development related to thorium stage should be in parallel with, rather than consecutive to, the FBR stage.

India is presently at the forefront of global thorium-related research, and is also listed as a partner in most of the forums under the International Atomic Energy Agency (IAEA) umbrella. However, India is not a member of other multilateral collectives that emphasise thorium-based systems, such as the Generation IV International Forum (GIF). If India can leverage its R&D and non-proliferation credentials and join international forums, it will help place thorium-based systems and fuel cycle on a faster track.

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1. Introduction

The idea of thorium as a nuclear fuel is as old as nuclear power systems themselves. Th-232 is an abundant fertile isotope that transmutes to U-233—a non-naturally occurring fissile uranium isotope—on absorbing a neutron. U-233 has superior fission fuel properties in comparison to Pu-239 and U-235, making the Th – U-233 fuel cycle a versatile option, conducive to most nuclear reactor types. Initial production of U-233 requires a source of neutrons in reactors using U-235 or Pu-239 as fuel or from accelerators. The feasibility of thorium cycle has been demonstrated in a wide variety of reactors over the years. Thorium-based fuels are also expected to have better in-core performance[1] and inherent proliferation resistance. Thorium being relatively inert and insoluble, long-term storage and permanent disposal of spent fuels is comparatively simple, with no complications of oxidation or leakage. The main technical challenges for thorium are in the fuel fabrication and spent fuel reprocessing ends of the fuel cycle (International Atomic Energy Agency, 2005) (Annexure-B).

Worldwide, the interest in thorium–as a means to safer nuclear energy–has revived in recent years. Viewed as a game changer in decarboning the energy sector, thorium-based nuclear systems are being pursued by a number of private entities in USA and Europe. Further, two international initiatives on futuristic nuclear technology – IAEA's international project on innovative nuclear reactors and fuel cycles programme (INPRO), and the US-led generation IV reactors forum (GIF) - are also actively pursuing the thorium route.

For India, thorium-based systems have been the endgame for its nuclear power strategy since its inception. This is because of its unique position in terms of nuclear fuel availability, with only less than 1% of global uranium reserves (poor quality) and two-thirds of global thorium reserves. India's nuclear energy strategy is based on the three-stage programme envisioned by Dr Homi J. Bhabha in 1954, the main aim of which was to utilise the vast thorium reserves and provide energy security to the country (Bhabha & Prasad, 1959). Statements by the Department of Atomic Energy (DAE) emphasise that large-scale thorium energy deployment will take place decades after the commercialisation of fast breeder reactors (Singh, 2017a) (Singh, 2017b). Annexure-C provides details on the official thorium-utilisation plans and progress. There may be alternative strategies that could expedite the thorium plans for India, which need not wait for the establishment of fast breeder reactors. The Indo-US nuclear deal of 2008 and the subsequent Nuclear Suppliers Group (NSG) waiver have created more opportunities, both in terms of technology and fuel resources—opening up more avenues for thorium utilisation.

Although there is consistent emphasis on the three-stage programme (Annexure-C) as India's official strategy to expand nuclear energy, there is a lack of clarity on by when thorium could be utilised on a large scale. In this context, we felt that it is relevant to assess the strategies and timelines for thorium adoption. Towards this, the report has the following components:

1. Review of various thorium-utilisation options (design and fuel cycle) relevant to India and assessment of the dynamics of fissile material build-up and utilisation;
2. Timeline for large-scale thorium utilisation and strategies to expedite thorium utilisation

[1] Thorium fuels attain higher burn-ups and operate safely and reliably.

2. Methodology

We reviewed all the design options relevant to India for thorium utilisation –pressurised heavy water reactor (PHWR), pressurised water reactor (PWR), sodium-cooled fast breeder reactor (FBR), advanced heavy water reactor (AHWR), and molten salt breeder reactor (MSBR) – in terms of the fuel cycles, fissile-material flow, advantages, and challenges. We then examined various routes for reaching thorium stage depending on the fissile-material flows and estimated a timeline based on fuel uptake, spent-fuel reprocessing, reactor availability factor, and construction times. The relevant assumptions and further details are provided in the analysis section. All data used, was obtained from a detailed literature review.

3. Review of Options

In this section, we will discuss thorium-related options relevant to India—reactor designs as well as fuel cycles.

3.1. Pressurised Heavy Water Reactor (PHWR)

PHWR is the mainstay of the first stage of the three-stage nuclear power programme[2] (Figure 1). There are 18 PHWR units operational, with a total capacity of 4.46 GWe. The Indian nuclear establishment has mastered the PHWR technology with years of operational experience. The cabinet nod to ten more units in May 2017 will further consolidate the domestic supply chain for such reactors.

Thorium can be used in a PHWR design, with a driver fissile material[3] – Pu-239 (Figure 2), U-235 in the form of slightly enriched (Figure 3) or natural uranium (Figure 4), or U-233 (Figure 5). With plutonium (Nuttin et al., 2006), natural (Albright & Vergantini, 2015) and slightly enriched uranium (Lewis, 1968) as driver fuels, this can potentially serve as a mode of producing U-233, to facilitate large-scale thorium adoption in Th – U-233 breeders (Chauhan, 2015).

However, the spent fuel reprocessing will pose severe challenges for adopting PHWR directly for thorium-fuel cycles, due to the presence of plutonium and higher concentration of U-232 in the spent fuel[4]. If this is overcome, it is possible to achieve self-sustaining thorium fuel cycle with only PHWRs, provided there is sufficient amount of U-233. (Critoph et al., 1976).

[2] For further details on three-stage nuclear energy programme, please refer to Annexure-C
[3] Driver fissile material is the start-up material that serves as an ignition to start the chain reaction
[4] The presence of U-232, a high gamma emitter is inevitable in thorium cycle. It poses a major challenge in fuel fabrication and reprocessing stages in limiting radiation exposure of the operating personnel. The concentration of U-232 varies from few tens to hundreds of ppm. In this case, the U-232 concentration is on the higher side ~200 ppm.

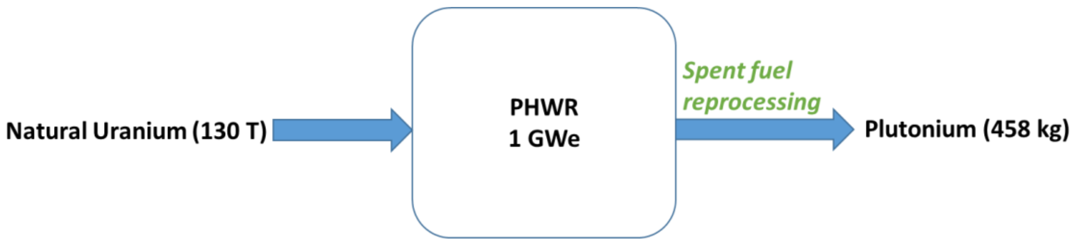

Figure 1: PHWR as the first stage of the three-stage programme

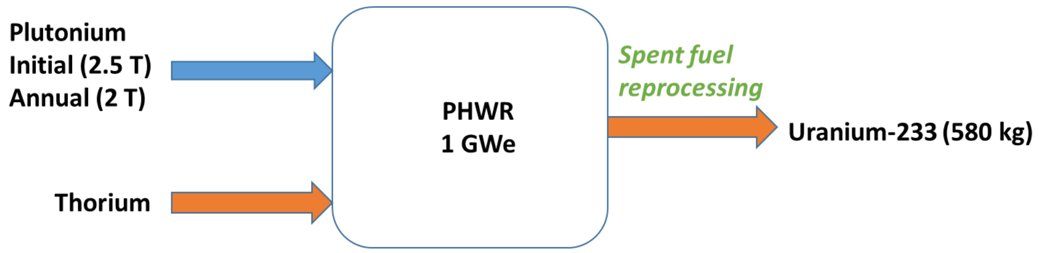

Figure 2: PHWR in thorium utilisation mode with Pu driver

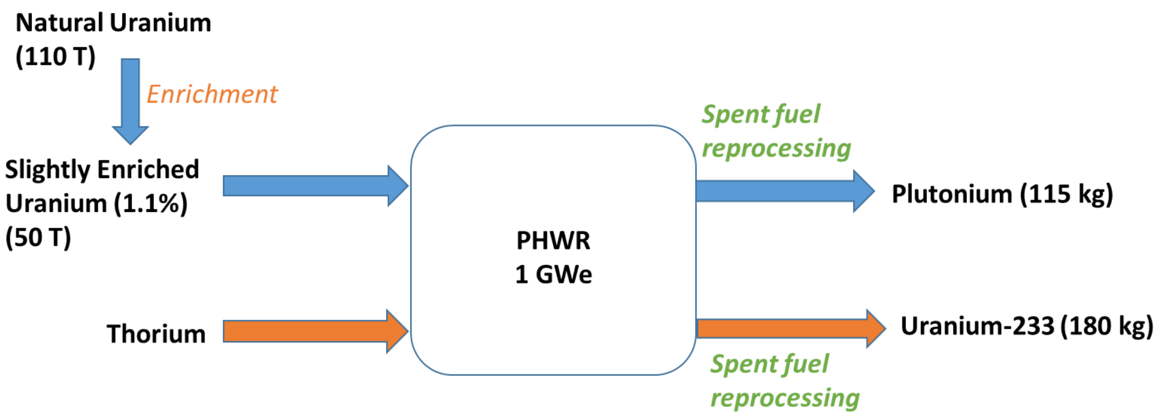

Figure 3: PHWR in thorium utilisation mode with SEU driver

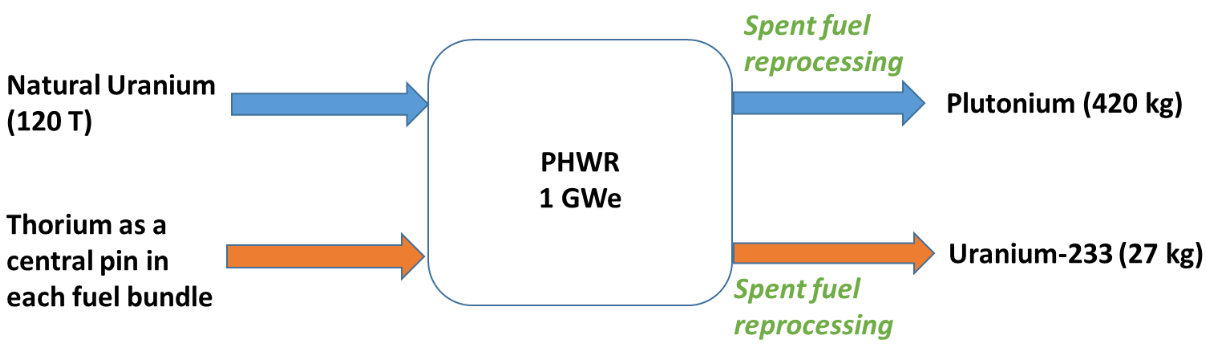

Figure 4: PHWR in thorium utilisation mode with natural uranium driver

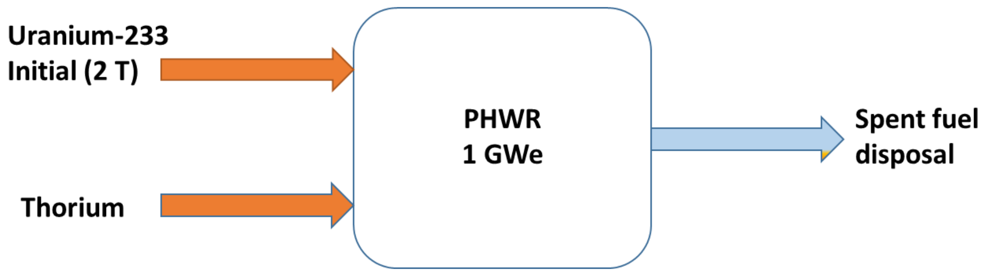

Figure 5: PHWR in thorium utilisation mode with U-233 driver

3.2. Pressurised Water Reactor (PWR)

Koodankulam 1 and 2, with a total capacity of 2 GWe, are the currently operational reactors of PWR design in India. India has signed agreements with Russia, France, and the US for importing Pressurised Water Reactors to support the nuclear power programme. Agreements are in place for 8*1000 MWe units with Rosatom, Russia; 6*1250 MWe with Westinghouse, US; and 6*1700 with Areva, France ("Nuclear Power in India | Indian Nuclear Energy - World Nuclear Association," n.d.).

PWRs conventionally operate on enriched uranium fuel cycle. Replacing the standard enriched uranium cycle with thorium fuel cycle is feasible in a PWR design (Figure 6) (Radkowsky, 1998). It was designed to be proliferation resistant and more economically feasible. The design runs on a "once-through" thorium fuel cycle, without having to reprocess U-233. This option can be quickly realised in any of the planned PWRs, subject to availability of low enriched uranium (LEU), which ideally should not be a limiting factor in light of the present possibility for purchase from external sources. This option provides a stand-alone mode for utilising thorium.

There are no technical challenges to this option, beyond redesign of the conventional PWR reactor vessel head.

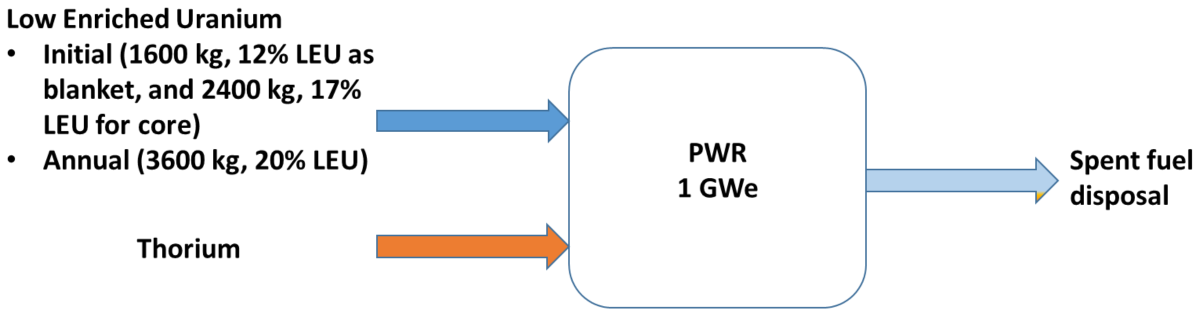

Figure 6: Thorium utilisation in PWR

3.3. Fast Breeder Reactor (FBR)

India has years of FBR operational experience with the Fast Breeder Test Reactor (13 MWe) and has completed the construction of the prototype 500 MWe unit (PFBR), both at Kalpakkam.

Owing to its fast-neutron economy, FBRs can breed fissile materials, i.e., convert fertile materials like U-238 and Th-232 to fissile materials, Pu-239 and U-233 respectively. The design of the reactor is such that the reactor core needs fissile material as fuel like the other reactors, and a 'blanket' around the core with fertile materials get converted to fissile material, breeding more fissile material than it consumes, as fuel.

Here, we consider two fuel cycles for the FBR – one with depleted uranium[5] (DU) in the blanket (Figure 7) which will breed plutonium; and one with thorium in the blanket (Figure 8), which will breed U-233. According to the three-stage plan, the plutonium reprocessed from PHWR will be used to fuel FBRs with DU blanket, which in turn will produce plutonium that can fuel FBR with thorium blanket. U-233, thus produced in the blanket, can be further used to fuel the final thorium-U233 stage.

The breeding could be better with carbide and metallic fuels, but they are in the research and development (R&D) phase. Most of the codes, and hence data available for fast breeder reactors, are based on plutonium and uranium mixed oxide-fuelled (MOX) core (Glaser & Ramana, 2007). The technique for reprocessing U-233 from thorium (Thorex) is in the R&D phase and needs to be demonstrated in industrial scale.

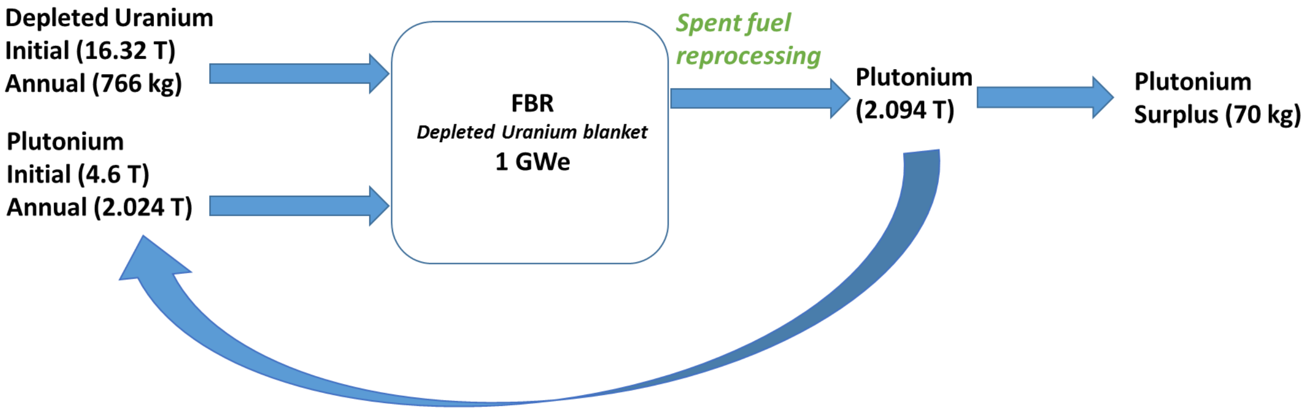

Figure 7: FBR with depleted uranium blanket

[5] Depleted uranium is uranium with lesser fissile isotope (U-235) composition than natural uranium. Natural uranium composition is 99.3% U-238 isotope and 0.7% U-235. Depleted uranium contains 0.07-0.08% U-235.

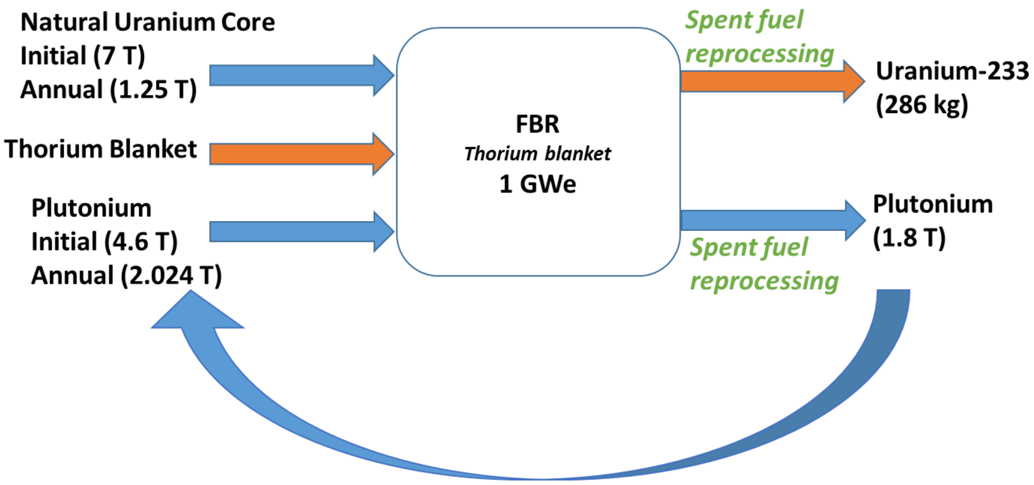

Figure 8: FBR with thorium blanket

3.4. Advanced Heavy Water Reactor (AHWR)

The design of AHWR, with either plutonium (Kumar, Srivenkatesan, & Sinha, 2009) (Figure 9) or LEU as driver fuel (*Advanced Heavy Water Reactor with LEU-Th MOX Fuel*, n.d.) (Figure 10), is in advanced stages and is expected to be ready for deployment in the near future.

The third stage reactors are envisaged to be Th–U-233 breeders, but they are still in the conceptual phase. AHWR was developed as an intermediate, time-bound plan for thorium fuel cycle demonstration. Although a start-up inventory of U-233 is required, it becomes self-sufficient in the Th - U-233 cycle. This design has important safety and economic advantages, but requires a continuous fissile feed as driver fuel. The U-233 produced in the AHWR from thorium will be burned in-situ, because of the high concentration of U-232 (~1000ppm), which renders the recovery of U-233 very complicated.

Figure 9: AHWR with Pu driver

AHWR with plutonium as driver fuel, however, takes a toll on the three-stage programme, specifically Stage 2, by drawing from the precious plutonium inventory meant to fuel FBRs. This further impacts the reprocessing requirements significantly. We have analysed this impact with three reprocessing capacity scenarios in Annexure-D.

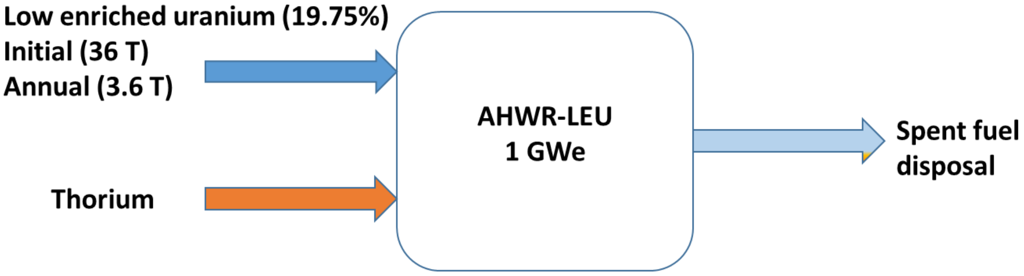

Figure 10: AHWR with LEU driver

3.5. Molten Salt Breeder Reactor (MSBR)

The ideal candidate for a stage 3 breeder is the Molten Salt Breeder Reactor (MSBR). The reactor design allows online or continuous reprocessing to achieve closed Th–U-233 cycle, so it does not require major reprocessing facilities. MSBR design has definite technological advantages over conventional reactor for U-233 breeding, owing to the fluid form of the fuel. Thorium-fuelled MSBRs meet many of the future goals of nuclear energy—improved sustainability, higher thermodynamic efficiency, inherent safety, stable coolant, low pressure operation that do not require expensive containment, ease to control, passive decay-heat cooling, and waste reduction(Elsheikh, 2013). Furthermore, MSBR can be operated in the load following mode. This will allow for flexible operation in a grid with a large share of intermittent renewable energy (MIT Energy Initiative, 2017). We have examined the MSBR option in detail in the next section.

The main challenge for this design is the requirement of appropriate structural materials—the molten salt fuel-coolant can rapidly degrade the structural materials through hotspots arising from deposition of fuel salt in cooler reactor parts, besides corrosion.

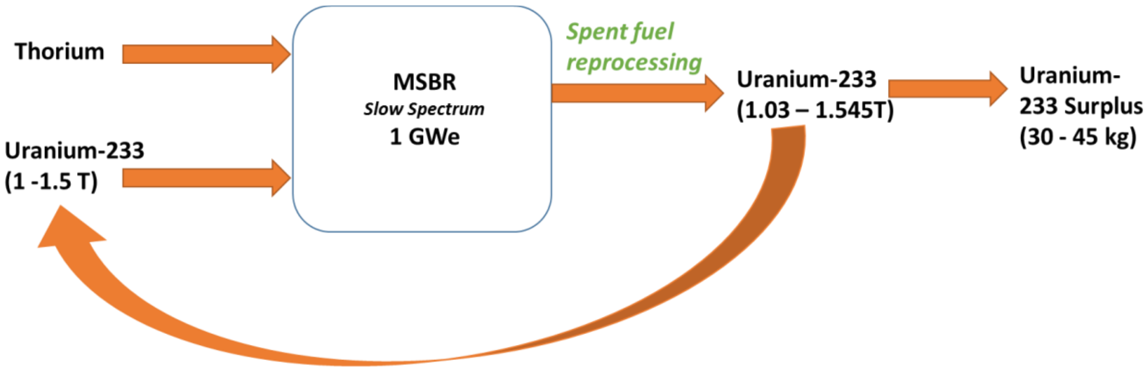

Figure 11: Slow-spectrum MSBR

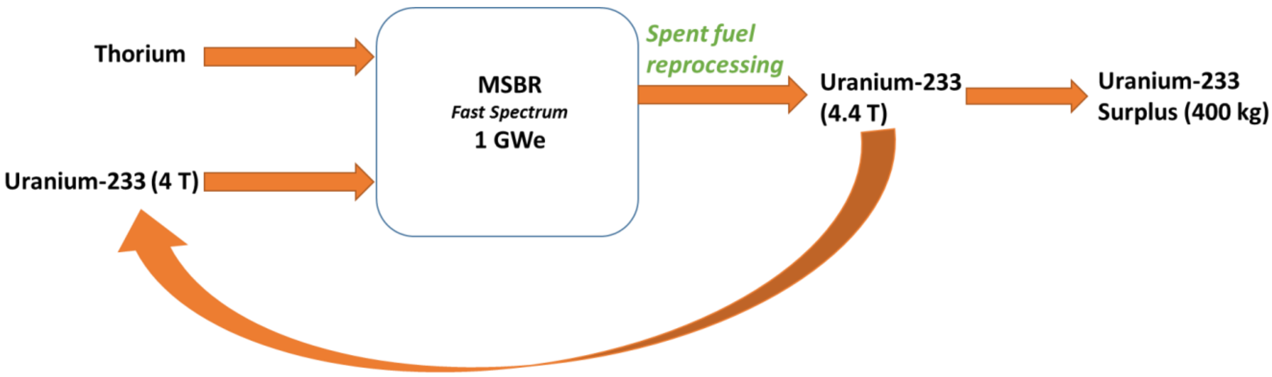

Figure 12: Fast-spectrum MSBR

MSBRs can operate in both slow- (Figure 11) and fast-spectrum (Figure 12) of neutron energy – i.e., with or without a moderator. The Indian MSBR is expected to be based on fast-spectrum (Vijayan et al., 2015).

4. Analysis and Results

In this section, we analyse the possible strategies of utilising thorium with the options reviewed and estimate a possible time frame for reaching the thorium stage. We will consider the quantity of plutonium that will build up till the year 2050 in each route (Annexure-D) as a common start point, and evaluate each of the routes in terms of corresponding reactor operation and the capacity addition required to get to the large-scale thorium stage. The reprocessing capacity requirement, as a key enabler, is also estimated.

The assumptions made for our analysis are provided in Table 1.

4.1. Energy Security

The main rationale for utilising thorium as per India's three-stage strategy is energy security/ independence. For energy independence through thorium-based systems, the key is to build a robust U-233 inventory. The quantity of U-233 required to set the ball rolling in the thorium-U-233 stage depends on the reactor system chosen. The most promising candidate for India and pursued worldwide as an inherently safe Gen IV design is the Molten Salt Breeder Reactors (Figure 11 and Figure 12).

U-233 requirement for MSBR operating on slow neutron spectrum is as low as 1-1.5 T/GWe and with a breeding ratio of 1.03, it will yield a surplus of 30 to 45 kg/GWe of U-233 annually.

For MSBR operating on fast-spectrum, U-233 inventory required is higher, ~4 T/GWe. However, MSBR in faster spectrum has a higher breeding ratio of 1.1, producing a surplus of 400 kg/GWe U-233 annually. It is also technologically less challenging than its thermal spectrum counterpart.

Once the thorium breeder stage starts, it can sustain itself. 1 GWe of fast-spectrum MSBR can produce sufficient U-233 to start another 1 GWe fast MSBR every 10 years. And 1 GWe of thermal spectrum MSBR can produce sufficient U-233 for another 1 GWe of thermal spectrum MSBR every 8-15 years. Table 2 provides a summary of the MSBR U-233 potential and timeline.

Table 1: Key assumptions

Fissile material flow: • **Thorium and natural uranium** are available **without any constraints** • **Reprocessing facilities** will function reasonably well with **90% recovering efficiency** and **~85% availability factor** • The analysis is based on the **plutonium build-up** by reprocessing the PHWR spent fuel from the existing and planned reprocessing plants (total 1000 T/year) **until 2050** as a common start point
Reactor parameters: • **Thermal efficiency** for all reactors is **30%** • **Average capacity factor,** based on their **historical generation,** has been assumed for the **currently operational reactors.** For the **reactors to be built**, **85%** capacity factor has been assumed • **Average discharge burn-up** for **PHWR 6500 MWd/T**; for **FBR over 100,000 MWd/T** • **Lifetime** of **50 years for PHWR**, **60 years for FBR and AHWR** reactor designs • For the **breeder** design reactors **civilian** mode of **operation** is assumed[6]. Only the surplus material after fuelling the reactor itself has been considered for further use.
Thorium – U233 reactor system: • **Molten Salt Breeder Reactor (fast-spectrum)** design chosen for our analysis.
Timeframe: • **2 years** for **cooling and reprocessing** of spent fuel • **6 years to build twin units of a reactor design** (assuming that technology and associated fuel cycle expertise are established and there is no siting issue) • **20 years gap after the first MSBR unit,** assuming the first as a prototype in every route • **No parallel projects** • **Back-to-back consecutive construction** every 6 years to estimate the earliest possible timeframe

Table 2: MSBR potential

Molten Salt Breeder Reactor	**U-233 per GWe (T)**	**Breeding Ratio**	**Annual Surplus U-233 per GWe (kg/GWe)**	**No of years to build sufficient U-233 for the next 1 GWe**
Thermal spectrum	1 - 1.5	1.03	30 – 45	34
Fast spectrum	4	1.1	400	10

[6] For the plutonium breeders assumed to be in civilian mode, there is no need or incentive to reprocess the plutonium in the blanket separately from the plutonium in the core. We assume that the entire stock of spent fuel discharged from the reactor is processed together, yielding a reactor grade-plutonium composition.

India's conceptual MSBR is of the fast-spectrum type, and we have assumed this design for our analysis. We have assessed the following four routes to build U-233 inventory to get to the MSBR stage.

Route A:

Route A is as per the three-stage programme. An aggressive expansion of the FBR stage is required to get to the MSBR stage in this route.

Maximum potential

The maximum potential in this route, considering plutonium build-up until 2050 is illustrated in Figure 13. We assess that a PHWR capacity of 4.4 GWe can lead to capacity addition of 31.2 GWe FBR – 26.4 GWe with depleted uranium (U-238) in the blanket to breed Pu and 4.8 GWe with thorium in the blanket to breed U-233 and result in a U-233 stock of maximum 11-12 T.

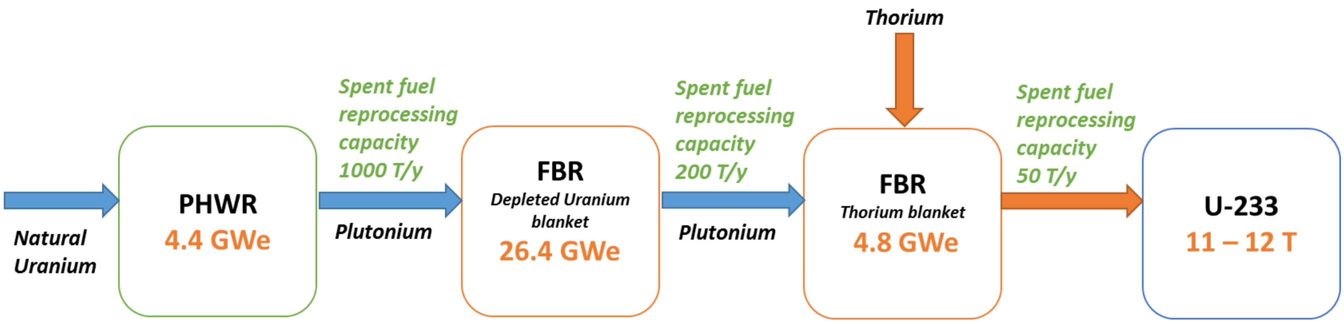

Figure 13: Route A – Maximum potential by 2050

To achieve this potential, this route requires reprocessing capacity as follows:

- Facilities to reprocess the PHWR spent fuel, both currently operational 400 T per year (Department of Atomic Energy, 2006a) and the planned 600 T per year capacity from 2021 onwards ("'Our policy is to reprocess all the fuel put into a nuclear reactor,'" 2012). This technology (PUREX) is well established.
- Additional facility (up to 200 T per year) of PUREX technology to reprocess plutonium from FBR blankets.
- Facility (up to 50 T per year) to reprocess U-233 from the thorium blanket and plutonium in the spent fuel from core of the next stage FBRs. U-233 reprocessing technology (THOREX) has not yet been demonstrated on a large scale.

Time Frame

For understanding the evolution of the route leading up to the thorium stage, the start point considered was the build-up of plutonium reprocessed from the PHWR spent fuel until 2050. We estimate the capacity and time frames for the FBR (depleted U blanket); FBR (thorium blanket); MSBR 1st generation (from U-233 build up from FBR with thorium blanket) and MSBR 2nd generation (from U-233 reprocessed from MSBR 1st gen).

We assumed six years for a twin unit of FBR with depleted uranium blanket to be constructed and ready for operation. Back-to-back construction every 6 years was considered to estimate the earliest possible time frame. For the subsequent stages – FBR with thorium blanket, MSBR 1st, and 2nd generation – maximum possible capacity as per plutonium and U-233 build-up has been considered.

Figure 14 represents the timeline of Route A.

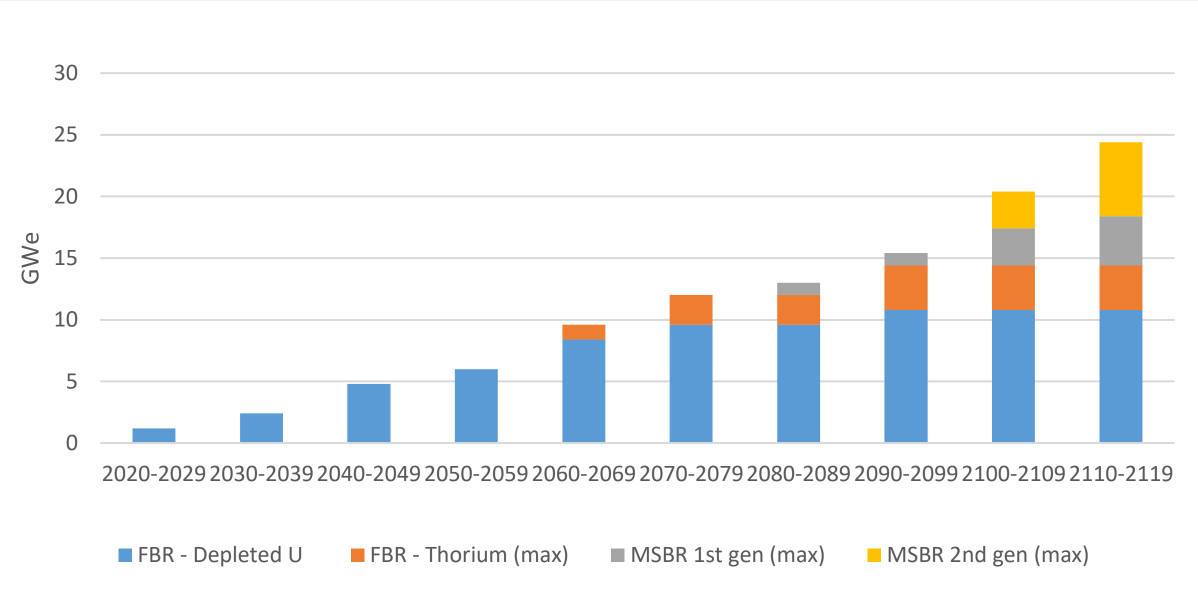

Figure 14: Route A - Timeline

Route B:

This route is a variation of (A) in that here, the plutonium from the PHWR spent fuel will be directly invested in U-233 breeding FBRs with thorium in the blanket.

Maximum Potential

The maximum potential in this route, considering plutonium build-up until 2050, is illustrated in Figure 15. A capacity addition of 13.2 GWe of such FBRs will result in a U-233 stock of about 73-74 T, which can be potentially used in U-233-based reactors.

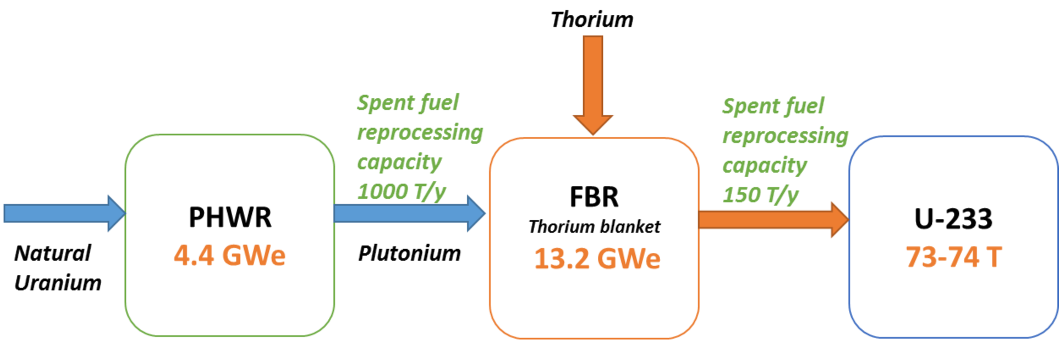

Figure 15: Route B - Maximum potential by 2050

In terms of reprocessing, to realise its maximum potential this route requires:

- Facilities to reprocess the PHWR spent fuel, both historical capacity and the planned 600 T per year capacity from 2021 onwards. This technology (PUREX) is well established.

- Facility (capacity up to 150 T per year) to reprocess U-233 from the thorium blanket of the next stage FBRs, and plutonium in the spent fuel from the core. U-233 reprocessing technology (THOREX) has not yet been demonstrated at a large scale.

Time Frame

The start point considered was the build-up of plutonium reprocessed from the PHWR spent fuel until 2050 here as well. We estimate the capacity and timelines for the FBR (Thorium blanket); MSBR 1st generation (from U-233 build up from FBR with thorium blanket) and MSBR 2nd generation (from U-233 reprocessed from MSBR 1st gen).

We assumed six years for a twin unit of FBR with thorium blanket to be constructed and ready for operation. Back-to-back construction was considered to estimate the earliest possible time frame. For MSBR 1st and 2nd generation, maximum possible capacity as per plutonium and U-233 build-up has been considered.

The timeline for Route B is represented in Figure 16.

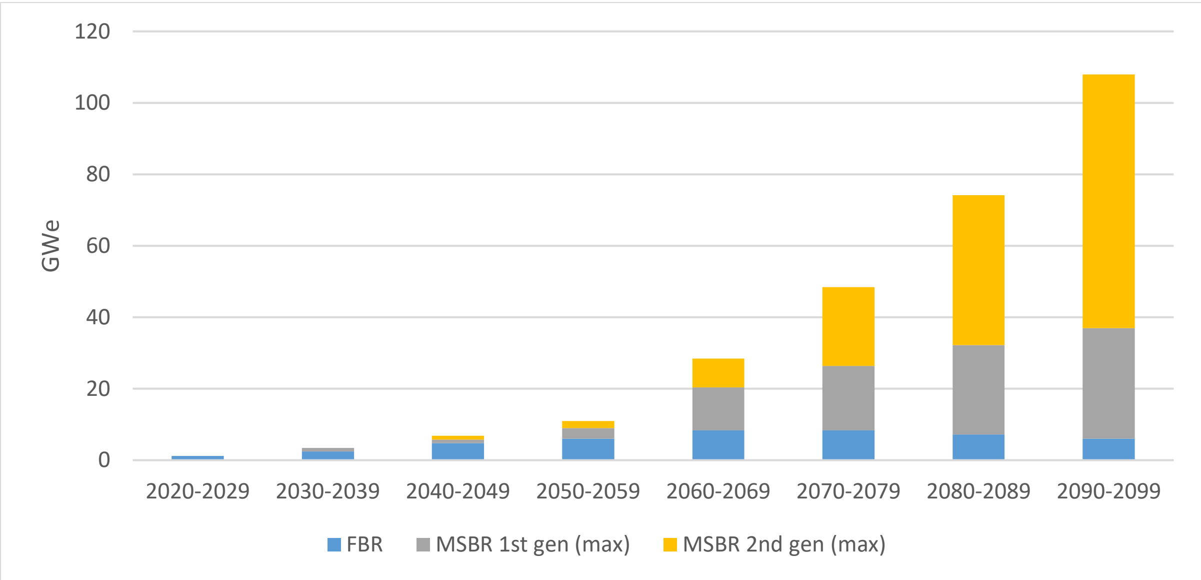

Figure 16: Route B - Timeline

Route C:

In this route, the plutonium from the PHWR (on natural uranium fuel cycle) spent fuel will be utilised in PHWR running on thorium-plutonium fuel cycle. U-233 can be reprocessed from the spent fuel.

Maximum Potential

This route can be easily adapted. Four PHWR units of 700 MWe are currently under construction in Rajasthan and Kakrapar. If they get commissioned by 2020 and two of those units were to run on thorium-plutonium fuel cycle, a stockpile of ~10 T can be achieved as early as 2035. If one more such unit switches to thorium-plutonium fuel cycle by 2034, the entire plutonium stock built up until 2050 will get invested as driver fuel for Th-Pu PHWRs. This will result in a U-233 inventory of 28-29 T.

A capacity addition of 2.1 GWe of thorium-plutonium fuelled PHWRs will lead to a U-233 stockpile of maximum 28 T, as illustrated in Figure 17.

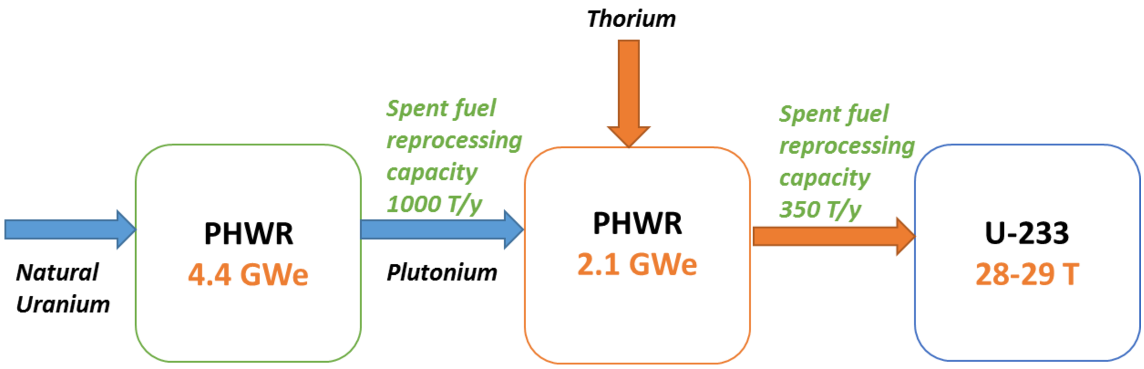

Figure 17: Route C – Maximum potential by 2050

In terms of reprocessing, this route requires:

- Facilities to reprocess the PHWR spent fuel, both historical capacity of 400 T per year and the planned 600 T per year capacity from 2021 onwards. This technology (PUREX) is well established.
- Facility (~350 T per year) to reprocess U-233 from the spent fuel from PHWR fuelled by thorium-plutonium. The presence of plutonium isotopes will render the process more complicated than THOREX process and this technology is yet to be demonstrated.

Time Frame

Figure 18 represents the timeline of MSBR 1st and 2nd generation maximum capacities possible with this route. We have considered plutonium build-up until 2050 from PHWR spent fuel as a common start point across routes. Such a plutonium build up is sufficient to fuel a maximum of 2.1 GWe of PHWR (thorium + plutonium) only until 2050. This is why, in the timeline, the PHWR stage is not seen after 2050. Now, if we were to consider plutonium build-up with the same reprocessing capacity beyond 2050, it would imply more years of operation of PHWR (Th + Pu), more U-233, which in turn translates to higher MSBR capacities – both generations.

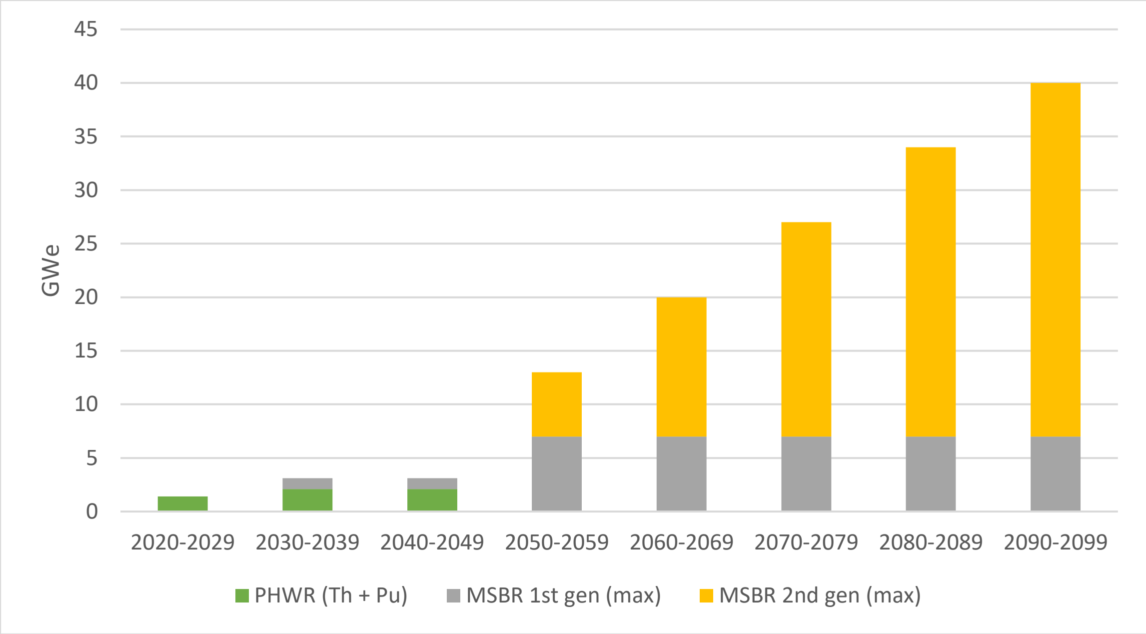

Figure 18: Route C - Timeline

Route D:

This route does not depend on plutonium and can produce U-233 with PHWR design.

Maximum potential

As of now, there is 4.46 GWe of PHWR power, and 2.8 GWe is under construction. If, say, all the PHWRs commissioned post 2000 (TAPS 3 and 4, Kaiga 3 and 4, RAPS 5 and 6) and the ones under construction, total 4.76 GWe, were to be running on the Th + nat U cycle from 2020 onwards, the resultant U-233 stockpile would be ~3.73 T by 2050. This is illustrated in Figure 19. In addition to U-233, the spent fuel also contains plutonium worth ~60 T, which can be used for fuelling other reactors.

In terms of reprocessing, this route requires:

- Facility (~750 T per year) to reprocess U-233 from the spent fuel from PHWR in thorium-natural uranium cycle. The presence of plutonium in the spent fuel will render the process more complicated and this technology is yet to be demonstrated.

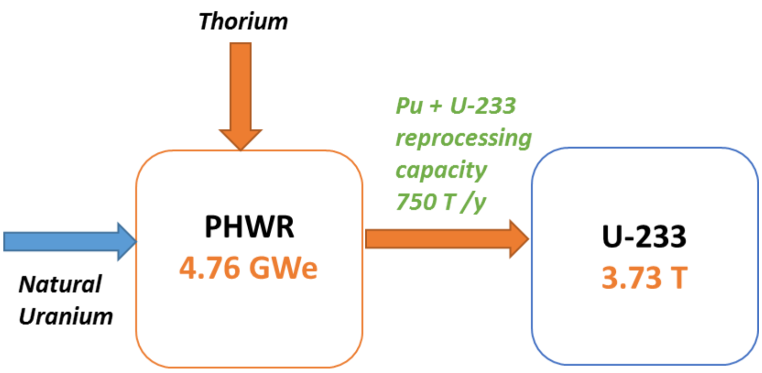

Figure 19: Route D - Maximum potential by 2050

Time Frame

Figure 20 represents the timeline of MSBR 1st and 2nd generation maximum capacities possible with this route. Here too, the reprocessing of PHWR fuel beyond 2050 is not considered to maintain consistency. Hence PHWRs are not seen beyond 2050.

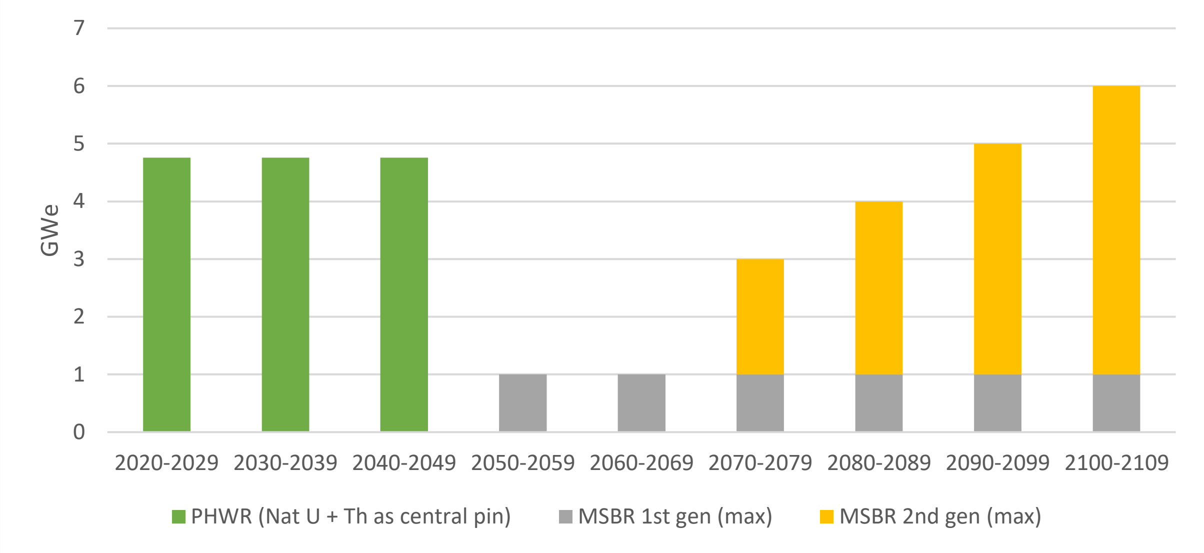

Figure 20: Route D - Timeline

Summary of the Routes

- In terms of U-233 inventory potential as well as timeline, Route (B) is the most promising. However, it is technologically tedious due to the FBR capacity requirement, as well as U-233 reprocessing capabilities.
- Routes (C) and (D) are the most easily adaptable, given that they are based on PHWR technology exclusively.
- Route (C) has comparatively good U-233 potential as well as an early timeline. Here, the main technological challenge will be the establishment of U-233 reprocessing capabilities.
- Route (D) is clearly the least effort route and even though the total capacity is very low, the MSBR 2nd generation numbers are comparable to Route A. It can serve as a supplementary route, given that Plutonium can also be reprocessed from the PHWR along with U-233.
- A comparison of the routes, with respect to maximum U-233 potential (with the plutonium build up until 2050), earliest possible timeframes, and corresponding additional capacity for 1 GWe MSBR is provided in Table 3 and Table 4.

Table 3: Routes summary – Time frame and additional capacity requirement

Route	Earliest time frame for 1 GWe MSBR (fast)	Corresponding additional capacity (GWe)
A	2082	12 (FBR)
B	2037	2.4 (FBR)
C	2029	1.4 (PHWR)

D	2053	4.76 (PHWR)

Table 4: Routes summary - Maximum U-233 potential and MSBR capacity

Route	Max U-233 potential (T) by 2050	Max 1st gen MSBR (fast) capacity (GWe) by 2100
A	10-11	1
B	73-74	31
C	28-29	7
D	3-4	1

4.2. Self-Sustaining Thorium Fuel Cycle (Without Breeder Reactors)

The four routes (A, B, C, D) rely on breeder reactors to realise thorium-U233 self-sustaining stage. It is also possible to achieve Th-U233 self-sustaining cycle by exclusively hinging on the PHWR design in principle.

For operating in self-sufficient mode, there is an initial requirement of U-233 of ~2T per GWe (Critoph et al., 1976). There are two possibilities to accumulate U-233 from PHWRs—with Pu as driver fuel (Figure 2) and with slightly enriched uranium (SEU) as driver fuel (Figure 3). The results are presented in Table 5.

Plutonium as driver fuel: Considering the build-up of plutonium, if two of the four PHWR (700 MWe) units that are under construction are commissioned by 2020; and are running on Th-Pu fuel cycle, 10 T of U-233 can be accumulated by 2035. This will be sufficient for 5 GWe of PHWR operating in self-sufficient mode. By the end of 2050, the total U-233 stock from Th-Pu PHWRs will be ~30 T, with a capacity of 2.1 GWe. This will be sufficient to sustain the lifetime requirements of the 5 GWe self-sustaining PHWR energy.

SEU as driver fuel: To accumulate U-233 at a comparable rate as plutonium driver fuel, the capacity in this case is 5 GWe by 2020 and 5 GWe more by 2035. Obtaining SEU does not seem to be a challenge in a scenario where the objective is self-sustaining nuclear energy based on thorium fuel addition cycle. There is no requirement of plutonium reprocessing facilities (Purex) at all for this route.

Table 5: Self-sustaining thorium cycle capacity- plutonium vs. SEU as driver fuel

YEAR	U-233 (T)	Pu as driver fuel	SEU as driver fuel
		Capacity (GWe)	Capacity (GWe)
2020	0	1.4	5
2035	10	2.1	10
2050	30	2.1	10

4.3. Thorium in Other Reactors:

Worldwide, the interest in thorium is for improving the credentials of nuclear energy because of its characteristic advantages such as proliferation resistance and easier spent-fuel disposal. There is a growing stock of thorium (800-1000 T annually) in India, as a side product of rare earth mining.[7]

The option of utilising thorium in operational reactors of PHWR and PWR designs, utilising lowenriched uranium (LEU) as a driver fuel can also be considered in the given context. Since India has already secured the NSG waiver and bilateral deals for fuel supply with Canada, Australia, and Kazakhstan and is pursuing membership in the NSG, securing LEU does not seem to be an obstacle for this option.

5. Discussion

An early introduction of thorium would be beneficial for establishing expertise in the various aspects of fuel cycle as well as data for validation of design codes. As of now, the candidate that is the most actively pursued by the DAE is the AHWR option, which is meant to be a technology demonstrator for the complete thorium fuel cycle. Given our limited indigenous uranium resources, growing stock of thorium, and feasibility of thorium fuel in existing reactor designs, it is worthwhile to consider the other early introduction possibilities. This will help in consolidating thorium-based fuel cycle technologies and facilities.

In the DAE replies to Lok Sabha questions on thorium, it has been consistently emphasised that large-scale thorium plans will be implemented after commercial establishment of the FBR stage(DEPARTMENT OF ATOMIC ENERGY, 2017, n.d.-b, n.d.-a). It becomes clear from our analysis that the reprocessing capacity is a key enabler for the full realisation of the three-stage programme. We also find that there are alternative strategies based on the PHWR design reactors (to build up U-233 inventory for breeders) that can advance large-scale thorium adoption by at least two decades. The main takeaway from our analysis is that if the spent fuel reprocessing of thorium-based fuel and Th-U-233 breeder design are ready for deployment before we enter the FBR stage on a big scale, it is possible to considerably advance the thorium-U233 stage[8]. In such a situation, we feel that the FBR-related R&D and thorium cycle related-R&D should be in parallel tracks, independent of each other.

India is at the forefront of global thorium-related research and is listed as a partner in most of the forums under the IAEA umbrella. However, India is not a member of other multilateral collectives that emphasise thorium-based systems, such as Generation IV International Forum (GIF). If India can leverage its R&D and non-proliferation credentials and join international forums like GIF, it may help in getting thorium-based systems and fuel cycle on a faster track.

[7] Indian Rare Earths Limited (IREL) has set up a 10,000 tons per annum (tpa) Monazite Processing Plant (MoPP) at OSCOM, Odisha. Monazite contains ~ 8 - 10% Thorium in its oxalate form, which means that ~ 800 - 1000 T of Thorium oxalate will be getting stockpiled annually as a by-product.

[8] Assuming we overcome the key technological challenges specific to thorium fuel cycle – Annexure-B

6. Annexure-A: Operational and Under-Construction Nuclear Reactors

Table 6: Operational and under-construction reactors

	Reactor	Design	Installed capacity (MWe)	Year / expected year of commencing of operations	Safeguards
			OPERATIONAL		
1	TAPS 1	BWR	160	1969	2009
2	TAPS2	BWR	160	1969	2009
3	RAPS 1	CANDU	100	1973 – 2004	2009
4	RAPS 2	PHWR	200	1981	2009
5	MAPS 1	PHWR	220	1984	Nil
6	MAPS 2	PHWR	220	1986	Nil
7	Narora 1	PHWR	220	1991	2014
8	Narora 2	PHWR	220	1992	2014
9	Kakrapar 1	PHWR	220	1993	2010
10	Kakrapar 2	PHWR	220	1995	2010
11	Kaiga 1	PHWR	220	1999	Nil
12	RAPS 3	PHWR	220	1999	2010
13	Kaiga 2	PHWR	220	2000	Nil
14	RAPS 4	PHWR	220	2000	2010
15	TAPS 4	PHWR	540	2005	Nil
16	TAPS 3	PHWR	540	2006	Nil
17	Kaiga 3	PHWR	220	2007	Nil
18	RAPS 5	PHWR	220	2010	2009
19	RAPS 6	PHWR	220	2010	2009
20	Kaiga 4	PHWR	220	2012	Nil
21	Koodankulam 1	PWR	1000	2014	2009
22	Koodankulam 2	PWR	1000	2017	2009
			AWAITING COMMISSIONING		
23	Prototype FBR	FBR	500	2018	Nil
			UNDER CONSTRUCTION		
23	Kakrapar 3	PHWR	700	2022	2017
24	Kakrapar 4	PHWR	700	2022	2017
25	RAPS 7	PHWR	700	2022	-
26	RAPS 8	PHWR	700	2022	-
27	Koodankulam 3	PWR	1000	2025	-

7. Annexure-B: Technological Challenges of the Thorium fuel cycle

Thorium fuel cycle poses unique challenges. Characteristics such as higher melting point, better radiation resistance, and lower co-efficient of thermal expansion, allow for it to have a higher burn up and ability to operate in much higher temperatures. However, the same characteristics cause challenges in structural material requirement – the cladding material and structural materials inside a reactor operating on thorium cycle require higher corrosion resistance to molten fluoride salts, higher radiation tolerance, better high-temperature strength, and good manufacturability.

The melting point sintering of ThO_2 (3350^0C) is higher than UO_2 (2800^0C). Hence, the temperature requirements for fabricating ThO_2 and ThO_2–based mixed oxide fuels is much higher compared to uranium fuels. ThO_2 and ThO_2–based mixed oxide fuels are relatively inert, which is a major advantage when it comes to fuel waste management and eventual disposal. However, they do not dissolve easily in concentrated nitric acid unlike uranium and uranium-plutonium mixed oxide fuels, complicating the spent fuel reprocessing significantly.

U-232 is produced in the Th –> U-233conversion, making both fuel fabrication and spent fuel reprocessing difficult. An increased presence of U-232 makes the process much more radioactive, due to the intense gamma radiation emitted by its decay, and necessitates remote operation. The presence of U-232 is inevitable in the thorium cycle, but the concentration varies in different reactor types according to the fuel type and burn up. This difficulty of handling U-233 with the presence of U-232 is also cited as the main reason why Th-U233 cycle is proliferation-resistant. In addition, it necessitates remote and automated handling in heavily-shielded cells, increasing the cost of fuel cycle activities.

8. Annexure-C: Official Plans and Progress

The Department of Atomic Energy (DAE) plans on thorium utilisation are based on the three-stage programme chalked out by Homi J Bhabha in 1958 (Bhabha & Prasad, 1959). The idea is that the spent fuel of one stage is used as a resource for the subsequent stages based on a closed fuel cycle. (Figure 21)

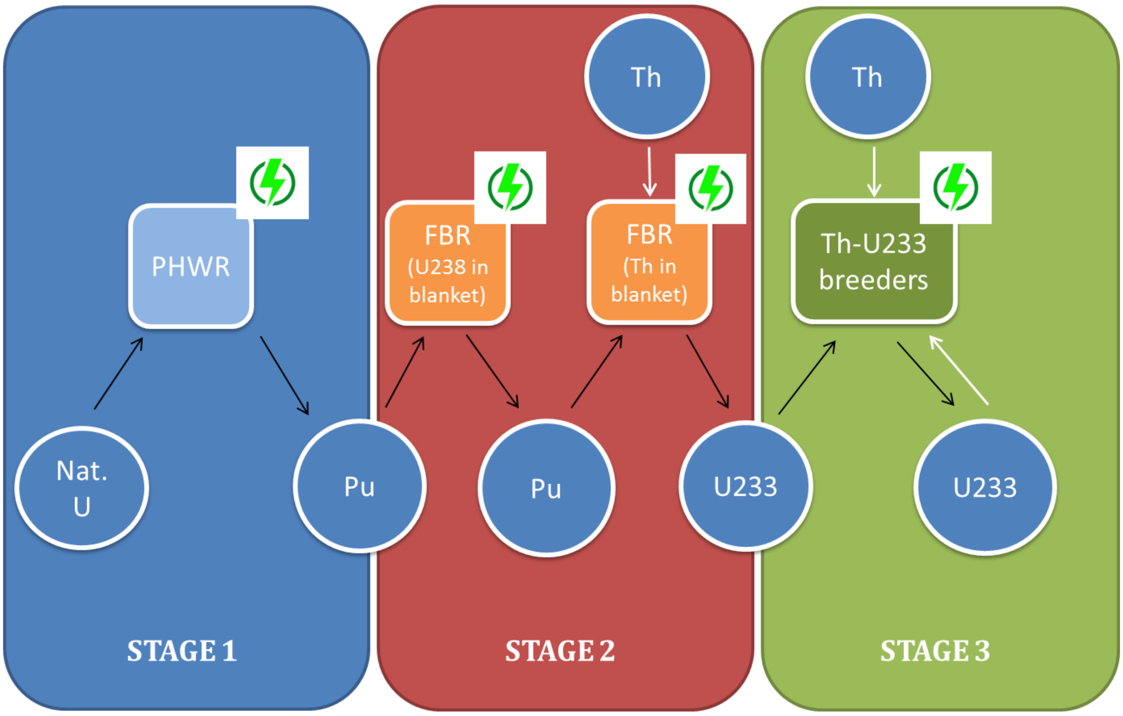

Figure 21: Three-stage nuclear energy programme

The spent fuel from Stage 1 reactors—PHWRs—contains plutonium, which can be used to fuel Stage 2 reactors—FBRs. Stages 2 and 3 were envisaged to progressively become independent of the previous stage(s), ultimately becoming self-sustaining. This is achieved by using breeder reactors. The plutonium from stage 1 is used to fuel FBRs designed to breed plutonium, to fuel more FBRs. Stage 2 will then become self-sustaining, with no dependency on plutonium from Stage 1. A portion of the stage 2 FBRs will breed U-233 from thorium, which will be used to fuel Stage 3 reactors. Stage 3 reactors will be Th - U-233 breeders that sustain themselves, thus becoming independent of U-233 input from Stage 2 ("About Us:ANUSHAKTI - Atomic Energy In India: Strategy for Nuclear Energy - BARC," n.d.).

8.1. Current Status

India is at the forefront of thorium research, among countries like the USA, Germany, Japan, and Russia. Considerable thorium irradiation data has been acquired in research reactors (CIRUS and DHRUVA) and PHWRs, but in a limited way. With sustained R&D over the years, there is experience over the entire thorium fuel cycle, but on a small scale. There is also some expertise in thorium-based fuel fabrication. The development of reprocessing technique called THOREX is still in the early stages and needs extensive modifications prior to achieving industrial status.

Most of the experience has come from the recovery of low amounts of U-233[9] that was bred in research reactors and some power reactors for non-fuel purposes[10]. The database needs significant augmentation to achieve industrial scale maturity for the process (Department of Atomic Energy, 2006a). Designs that are being pursued for large-scale thorium utilisation by the DAE are the following (Kakodkar, 2001):

1. **Advanced Heavy Water Reactor (AHWR):** AHWR is designed for timely development of thorium-based technologies. The design incorporates advanced safety features and is an overall improved version of the indigenous PHWR design. Pre-licensing design safety appraisal has been completed for the design by the Atomic Energy Regulatory Board (AERB) and is currently in advanced stage of detailed engineering (*Advanced Heavy Water Reactor*, 2008). An export version of the AHWR design, envisaged to be fuelled by Low Enriched Uranium (LEU) and Thorium, is also being pursued by the Bhabha Atomic Research Centre (*Advanced Heavy Water Reactor with LEU-Th MOX Fuel*, n.d.).
2. **Molten Salt Breeder Reactor (MSBR):** India is working on the development of technologies for the molten salt breeder reactor (MSBR), which can breed fissile material effectively, and hence can be useful to provide long-term sustainable energy using thorium (Vijayan et al., 2015).
3. **Accelerator-Driven Reactor System (ADS):** As discussed earlier, thorium has to be transmuted to U-233, for it to undergo fission and produce energy. In regular reactors in the critical mode, a driver fissile nuclei provides the neutrons required and excess reactivity is controlled by neutron absorber rods; whereas in an ADS, the core is kept subcritical and the neutrons are continuously supplied by an external accelerator system. Furthermore, there is no limit on the achievable burn up of the fuel since criticality need not be maintained. Therefore, ADS is expected to have superior breeding capabilities[11]. Studies are ongoing at BARC to evolve a suitable ADS design optimising the energy produced, fuel utilisation, and energy required for the accelerator system. Since this reactor idea is in a conceptual phase and involves additional development of high-power accelerator technologies, it is not included in the report (Department of Atomic Energy, 2006b).

[10] Thorium, being a fertile element, is used in small amounts for adjusting non-uniform neutron flux inside nuclear reactors. A small quantity of U-233 gets produced in the process. The U-232 concentration is usually <=500 ppm.

[11] The Accelerator Driven System (ADS) is a type of reactor which produces power even though it remains 'sub-critical' throughout. Conventional reactors are 'critical' - which means that the number of neutrons produced by fission is exactly balanced by the number lost by leakage and absorption by various materials in the reactor. This balance is responsible for the chain reaction and hence maintaining constant reactor power. In 'sub-critical' mode, reactors produce fewer neutrons by fission than are lost by absorption and leakage, and require an external supply of neutrons to sustain the chain reaction. Neutrons are supplemented externally by 'spallation' – neutrons produced by interaction of a high energy proton beam with a heavy atom nucleus such as lead. Such reactors are safe since the reactor can be controlled externally by the proton beam, rather than the use of control rods in the reactor core to adjust and maintain criticality. Higher burn ups can be achieved in these systems, because of external control of neutron supply.

9. Annexure-D: Plutonium From PHWR Spent Fuel

Throughout our analysis, we have used the plutonium build up from PHWR spent fuel till 2050, as represented in Table 7. Only the existing and planned reprocessing capacities have been considered.

Table 7: Plutonium from PHWR

YEAR	Reprocessed spent fuel quantity (Tonnes per year)	Plutonium content in the spent fuel	Plutonium content with 90% efficiency
1983	21.37	0.07	0.07
1984	23.89	0.08	0.08
1985	23.89	0.08	0.08
1986	23.89	0.08	0.08
1987	45.75	0.16	0.14
1988	45.75	0.16	0.14
1989	68.71	0.24	0.22
1990	68.71	0.24	0.22
1991	68.71	0.24	0.22
1992	68.71	0.24	0.22
1993	68.71	0.24	0.22
1994	93.23	0.33	0.29
1995	100	0.35	0.315
1996	100	0.35	0.315
1997	200	0.7	0.63
1998	100	0.35	0.315
1999	100	0.35	0.315
2000	200	0.7	0.63
2001	200	0.7	0.63
2002	200	0.7	0.63
2003	200	0.7	0.63
2004	200	0.7	0.63
2005	200	0.7	0.63
2006	200	0.7	0.63
2007	200	0.7	0.63
2008	200	0.7	0.63
2009	200	0.7	0.63
2010	200	0.7	0.63
2011	200	0.7	0.63
2012	200	0.7	0.63
2013	200	0.7	0.63
2014	400	1.4	1.26
2015	300	1.05	0.945

2016	300	1.05	0.945
2017	400	1.4	1.26
2018	400	1.4	1.26
2019	400	1.4	1.26
2020	400	1.4	1.26
2021	1000	3.5	3.15
2022	1000	3.5	3.15
2023	1000	3.5	3.15
2024	1000	3.5	3.15
2025	1000	3.5	3.15
2026	1000	3.5	3.15
2027	900	3.15	2.835
2028	900	3.15	2.835
2029	1000	3.5	3.15
2030	1000	3.5	3.15
2031	1000	3.5	3.15
2032	900	3.15	2.835
2033	900	3.15	2.835
2034	1000	3.5	3.15
2035	1000	3.5	3.15
2036	400	1.4	1.26
2037	400	1.4	1.26
2038	1000	3.5	3.15
2039	1000	3.5	3.15
2040	1000	3.5	3.15
2041	1000	3.5	3.15
2042	1000	3.5	3.15
2043	1000	3.5	3.15
2044	900	3.15	2.835
2045	900	3.15	2.835
2046	1000	3.5	3.15
2047	1000	3.5	3.15
2048	1000	3.5	3.15
2049	900	3.15	2.835
2050	900	3.15	2.835

A total reprocessing capacity of 1000 T per year is sufficient to process the spent fuel from the currently operational PHWRs (not including the four units under construction), but it will be insufficient if more PHWR units were to be commissioned.

Every reprocessing facility is assumed to shut down for 2 years after every 15 years of operation for maintenance (Availability factor = ~86%).

10. Annexure-E: AHWR as the Final Stage of Three-Stage Nuclear Plan

This section illustrates impact on the nuclear energy and reprocessing capacities and the overall timeline if the chosen third-stage reactor system requires plutonium, such as the AHWR. The projections are made following a fissile material approach for the three-stage programme, i.e., the capacity building of the subsequent stage is determined by the quantity of fissile material available as a consequence of the previous stage(s). In addition to the currently operational PHWRs, we have also considered 10 additional units. All Pressurised Heavy Water Reactors (PHWR) and the Prototype Fast Breeder Reactor (PFBR) are considered decommissioned after 50 years; Fast Breeder Reactors (FBR) after 60 years and the Advanced Heavy Water Reactors (AHWR) after 80 years. If the lifetimes are extended, the results will vary. We considered three reprocessing capacity scenarios:

1. **Present reprocessing capacity:** Three facilities, PREFRE and PREFRE 2 at Tarapur; and KARP at Kalpakkam, with a total capacity of 400 T per year continuing to operate throughout.
2. **INRP 2020:** In addition to the present 400 T per year, the officially planned Integrated Nuclear Recycling Plant (INRP) with a capacity of 600 T per year is assumed to come online by 2020.
3. **Projections:** Projections for reprocessing capacity thereafter is made such that the entire stockpile of PHWR spent fuel accumulating over the years is reprocessed. The projected capacity may be an over estimate, given that the current global reprocessing capacity is ~6000 T per year.

Table 8: Reprocessing capacity additions in 'Projections' scenario

Year	Capacity additions (T/y)
2020	600
2038	900
2055	1500
2072	3000

10.1. Scenario 1: Present Reprocessing Capacity:

This scenario is based on the assumption that the reprocessing capacity will remain 400 T/y, the present-day capacity. The independent stage 1 (PHWR) reaches a maximum capacity of 17.28 GWe by 2040. The fissile material build up is such that the earliest possible deployment of AHWR reactors is by 2040-44. Optimised by the plutonium requirement by both stage 2 and 3 reactors, FBR stage reaches its maximum possible capacity of 2.5 GWe by 2030 and AHWR reaches its maximum of 1.8 GWe by 2085 or so. The capacities are severely limited by the availability of plutonium because of the meagre 400 T/y reprocessing capacity throughout. Even though there is sufficient spent nuclear fuel accumulating from Stage 1 PHWRs, it fails to translate to plutonium fuel for the lack of reprocessing capacity in this scenario. The maximum total capacity of the three-stage plan in this case is 20 GWe achieved in the 2040-44 period.

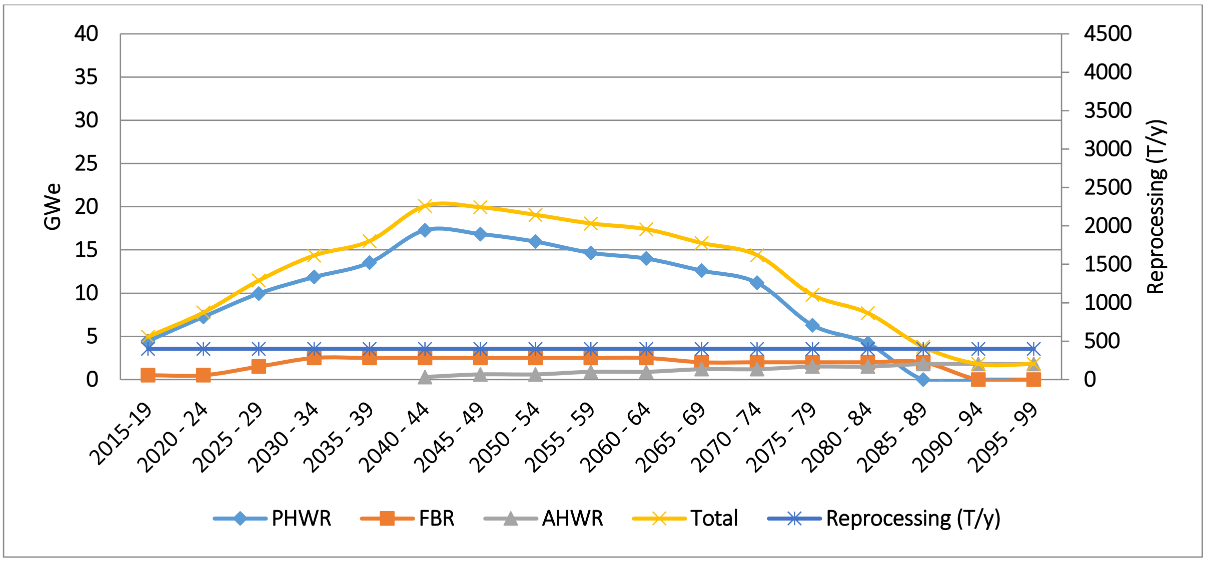

Figure 22: Present reprocessing capacity

10.2. Scenario 2: INRP 2020

In this scenario, it is assumed that the Integrated Nuclear Recycle Plant (INRP) with 600 T/y capacity will be operational by 2020 to augment the present day 400 T/y capacity. The independent stage 1 (PHWR) reaches a maximum capacity of 17.28 GWe by 2040. The fissile material build up is such that the earliest possible deployment of stage three (AHWR) reactors is by 2040-44. Optimised by the plutonium requirement by both stage two and three reactors, FBR stage reaches its maximum possible capacity of 6.5 GWe by 2040 and AHWR reaches its maximum of 4.8 GWe by 2095 or so. The plutonium situation has improved compared to the previous case, and is reflected in the increased capacities of stage two and three reactors. It is to be noted that even in this case, the reprocessing capacity is not sufficient to reprocess the entire stockpile of spent fuel accumulating from stage one reactor operations. The maximum total capacity of the three-stage plan in this case is 24.8 GWe, achieved in the 2040-44 period.

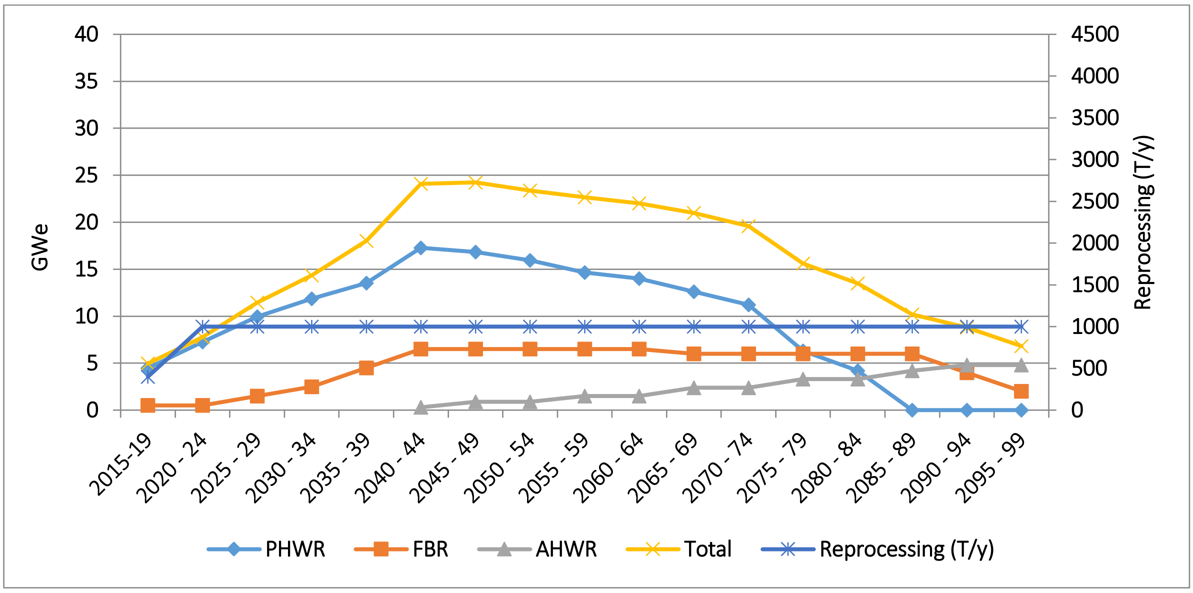

Figure 23: INRP 2020

10.3. Scenario 3: Projections

In this scenario, the reprocessing capacity is projected so that the entire stockpile of spent fuel accumulating from stage one is translated to its plutonium worth. The independent stage one (PHWR) reaches a maximum capacity of 17.28 GWe in the year 2040-44. Optimised by the plutonium requirement by both stage two and three reactors, FBR stage reaches its maximum possible capacity of 12.5 GWe by 2030 and AHWR reaches its maximum of 12 GWe by 2095 or so. The maximum total capacity of the three-stage plan in this case is 34 GWe, achieved in the 2040-44 period.

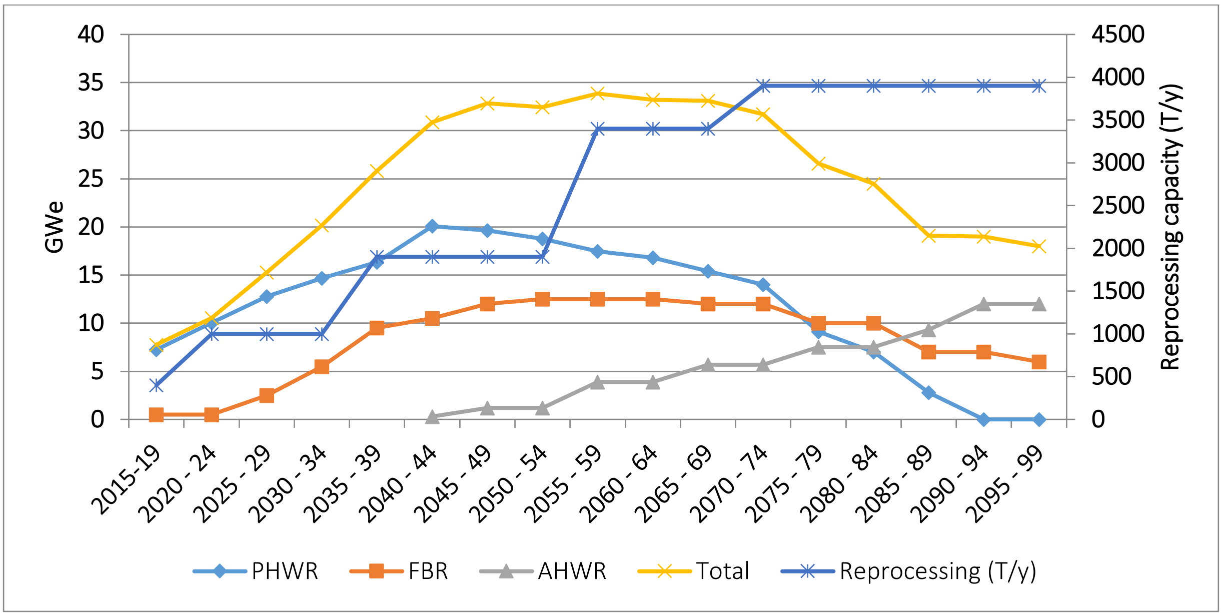

Figure 24: Reprocessing projections

The summary of the results is presented in Table 9, which illustrates the impact of reprocessing on the timeline and overall capacity, even though the starting point of quantity of spent nuclear fuel from the PHWR is common across all three scenarios.

Table 9: Reprocessing scenarios summary

Scenario	**FBR Maximum capacity**		**AHWR Maximum capacity**		**Total maximum capacity**	
	GWe	**Period**	**GWe**	**Period**	**GWe**	**Period**
Present capacity	2.5	2030-34	1.8	2085-89	20	2040-44
INRP 2020	6.5	2040-44	4.8	2095-99	24	2045-49
Projections	12.5	2050-54	12	2090-94	34	2055-59

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