India's nuclear breeders: Technology, viability, and options

Rahul Tongia and V. S. Arunachalam

India's nuclear power programme is based on indigenous materials and technology, with the potential for providing energy security for many centuries. This paper examines the technical viability of this plan, specifically, the role of breeder reactors for extending the domestic uranium supplies. Our study shows breeding is unlikely to occur at anywhere near the rates envisioned, leading to a very slow growth of fast breeder reactors. In addition, domestic uranium reserves restrict the growth of Pressurized Heavy Water Reactors, which are likely to be the main contributors to nuclear capacity in the short-term. The Th–^{233}U cycle in fast breeders does not appear attractive, and, for the ^{238}U–Pu cycle, only metallic fuel offers hope of a relatively fast doubling and reprocessing time. To increase the share of nuclear power in the coming decades, India should consider the construction of a number of large thermal reactors based on indigenous and imported uranium and also the design, development and validation of reactors that operate with Th–Pu fuels.

INDIA'S nuclear power programme began in 1948 with the establishment of the Atomic Energy Commission under the chairmanship of Homi Bhabha. The need for energy security based on domestic fuel availability was cited as a basic reason for going to nuclear power. The coal reserves were not considered adequate for the needs of the growing population of India. In the 1955 and 1958 UN conferences on Peaceful Uses of Atomic Energy, Bhabha argued that the economics of nuclear versus coal power depended on the assumptions such as the plant distance from the coal mine. Nuclear power was expected to be the only available long-term source of energy for India[1–3].

As India preferred a technology that did not require uranium enrichment, then considered expensive, it was decided to use Pressurized Heavy Water Reactors (PHWRs). These are of the Canadian Deuterium Uranium (CANDU) design, and use natural uranium for fuel and heavy water as moderator.

A long-term goal of India's programme is to use the vast reserves of thorium available in the country. As thorium is a fertile material and not a fissile one, it is necessary initially to use another fissile material for fuel and also to breed fissionable uranium 233 (^{233}U) from the thorium–^{233}U cycle. Plutonium (Pu), a by-product of the CANDU reactor, is the initial fuel for this stage. A method for producing more Pu is through breeding in Fast Breeder Reactors (FBRs). FBRs using plutonium fuel operate with no moderator (hence, the name fast reactor), and can be designed to produce more fissile material than they consume (hence, the term breeding). The fissile material produced at the end of this stage is the fuel for the third stage of the planned power cycle. The three-phase power programme as formulated by the Indian Atomic Energy Establishment is shown in Table 1 (refs. 4, 5).

Development and testing for the second phase is already underway. India has been operating a Fast Breeder Test Reactor (FBTR) since 1985, and has even started producing one megawatt of electric power from this reactor. However, the first Prototype Fast Breeder Reactor (PFBR) of 500 MWe (megawatt-electric) using U–Pu as fuel is still one or more decades away. Nuclear Power Corporation (NPC) is unlikely to begin commercial deployment of fast reactors until the planned 500 MWe PFBR or its successor becomes technically successful and commercially viable. Assessed from the present status and fissile materials availability, India, in the coming decade, is unlikely to commission the third stage power reactor that uses ^{233}U as fuel.

Despite operating Asia's first nuclear reactor, India has now fallen behind many countries pursuing nuclear power. Today's installed gross nuclear capacity of 2210 MWe is actually de-rated to or available at only about 1750 MWe, and operates at overall (lifetime) plant load factors (PLFs) of only around 50%. This compares unfavourably to the March 1996 world average lifetime PLFs of nearly 70% for Pressurized Water

Rahul Tongia and V. S. Arunachalam are in the Department of Engineering and Public Policy, Carnegie Mellon University, Pittsburgh, PA 15213, USA.

Table 1. India's planned three phase nuclear power programme. Based on the potential for PHWRs, this implies a domestic availability of approximately 60,000 tons natural uranium

	I	II	III
Reactor type	PHWR	LMFBR	LMFBR or HWR
Fuel	Natural uranium (once-through)	(U–Pu) oxide or carbide or metal	Thorium–^{233}U cycle
Potential	~340 GWe-yr	~16,000 GWe-yr	~168,000 GWe-yr
	Produces Pu in spent fuel	High breeding	Low breeding

Reactors (PWRs) and 66% for Pressurized Heavy Water Reactors (PHWRs) (ref. 6). Though the performance of the most recent year shows improvement in PLFs to 67% (ref. 7), these may not be adequate to compensate for a less than optimal performance in the past decades.

Concerns on performance aside, the importance of nuclear power lies in its potential as well as the energy security it might provide. While the current share of nuclear power is only about 2% by generation[8], it still remains as one of the important and available sources of power to meet India's growing energy demands. India now experiences about 14% average shortfall and 28% peak shortfall in electricity production[9]. In fact, power has now become such an important national priority that almost one-third of all development investments in India are related to this issue[10].

In this paper, we discuss the technology and viability of breeder reactors using different fuels in India's nuclear power programme. We evaluate these by using the concept of System Doubling Time (SDT) and the data on breeding and fuel doubling times available in open literature. The paper is divided into three parts. In the first part, we define and discuss the parameters used in our models and also their chosen values. The results of the modeling studies are presented in the second part. In addition to presenting results on breeding, we discuss the performance of PHWRs as these reactors are also expected to contribute to the fuel required for fast reactors. In the last section, based on the results obtained from our modeling, we discuss the performance of India's Nuclear Power Programme in meeting India's electricity needs and the original objective of energy security, and suggest options that may be technologically more relevant today than the plan formulated many decades ago.

Methodology

In this study, we build a parametric model for the growth of nuclear power reactors solely based on the constraints of fissile material availability. India's domestic reserves of uranium are limited and estimated to be 30–70 thousand tons[11,12]. While the current worldwide glut of uranium has depressed uranium prices and reduced incentives for more exploration, India's case is different, as India does not have access to uranium from outside. A recent article by Ramanna[13] suggests that India has only 35,000 tons of recoverable uranium at $80 or lower per kg. India's limited fissile material stock when used in PHWRs can provide only 58 GWe-yr of energy per 10,000 tons of uranium (at 6,700 MWt-days/ton rated burn-up), or only 407 GWe-yr based on 70,000 tons uranium.

Through fuel reprocessing and breeding in FBRs, the fuel supplies can be extended by a factor of 50. This is achieved by converting fertile ^{238}U (found in natural uranium as well as spent fuel) into fissile ^{239}Pu in a FBR. FBRs require significant quantities of fissile material to initiate criticality. If enriched uranium is not used, depending on the reactor size, 15 to 20% of the fuel in a FBR would have to be plutonium.

A sample of the fissile material flow is given below[14,15]:

1 ton nat. U used in PHWRs → 3.5 kg Pu in spent fuel

$$\frac{118 \text{ tons U / yr}}{\text{GWe PHWRs}} \{@0.685 \text{ PLF}\} \rightarrow 413 \text{ kg Pu/yr.}$$

Doubling time

Plutonium processed from the spent PHWR fuel can be loaded into a FBR for breeding more plutonium from ^{238}U. Upon burn-up in a reactor, the discharged fuel must undergo reprocessing before reuse. The term reprocessing is often used to describe the entire process before reuse, i.e. the combination of cooling of burnt-up fuel rods, processing to extract plutonium, and fuel fabrication. Most of the reprocessed plutonium is loaded back as fuel into the same reactor, and the excess plutonium is accumulated for starting another reactor. *Doubling time* characterizes this growth of fissile material. Depending on the operating conditions, the doubling times are defined as follows[16,17]:

Reactor doubling time

This is the core fissile requirement (in-pile inventory) divided by the fissile gain per year. Any excess fissile material produced over the Fissile Material at the Beginning of the Cycle (FBOC) is known as Fissile Gain (FG).

System doubling time

This measure accounts for losses both during reprocessing and by radioactive decay. These losses occur outside the reactor. SDT also accounts for the out-of-pile

Table 2. Reactor profiles as used in the study. These are based on refs 20, 23

		Oxide	Carbide-Lee	Metal	Carbide
^{238}U–Pu	Blanket material	dep. U	dep. U	dep. U	dep. U
cycle	In pile inventory (kg)	3158	1449	2248	2615
	Fissile gain (kg/yr) (before losses)	245	144	412	354
	Breeding ratio	1.325	1.406	1.582	1.479
	Burn-up (MWd/ton)	100,000	100,000	100,000	100,000
	Core reload fraction	1/3	1/3	1/3	1/3
	Nominal fuel residence (years)	2	2	2	2
	Reactor size (MWe)	500	500	500	500
Th–^{233}U	Blanket material	Th	Th	Th	Th
cycle	In pile inventory (kg)	3304	1482	3040	2903
	Fissile gain (kg/yr) (before losses)	43	36	56	58
	Breeding ratio	1.099	1.098	1.115	1.114
	Burn-up (MWd/ton)	100,000	100,000	100,000	100,000
	Reload fraction	1/3	1/3	1/3	1/3
	Nominal fuel residence (years)	2	2	2	2
	Reactor size (MWe)	500	500	500	500

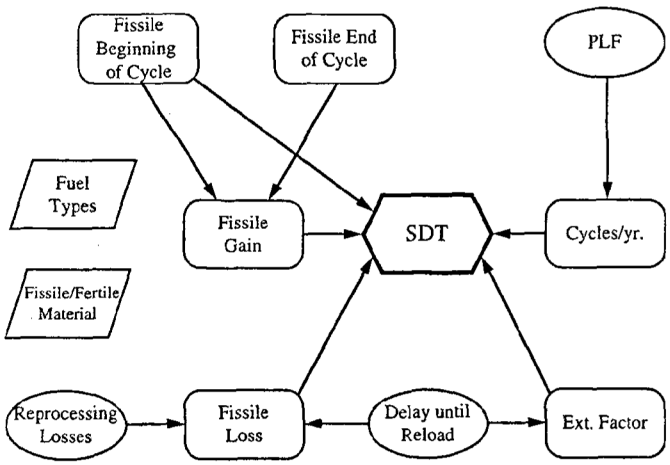


Figure 1. Influence diagram for system doubling time model. SDT is proportional to the total cycle fissile inventory (beginning of cycle plus out-of-pile) and inversely proportional to the net gain per cycle and the number of cycles per year. SDT calculations are performed across fuel cycles and fuel types.

inventory requirements due to the delay between the discharge of fuel after burn-up and its reload, time required for cooling, reprocessing and fabrication of the fuel. This extra inventory is measured by the term Ext Factor (EF), in eq. (2). SDT, by definition, is longer than reactor doubling time (RDT).

Compound system doubling time

This doubling time is for compounded growth of fissile material, and assumes a growing number of reactors. As soon as enough fuel for a new reactor accumulates, it begins operation, increasing the rate of fissile production. It equals SDT × $\log_e$ (2). A single reactor can only achieve SDT, but as the system grows to a number of reactors, the doubling time reduces from SDT to CSDT. It is generally accepted that as the reactor base grows to between 10 and 16 reactors, the doubling time approaches 0.7 × SDT, approximately $\log_e$ (2) × SDT. This compounded growth requires not only a continuous addition of new reactors as soon as the fuel is ready, but also proportional increases in reprocessing and fabrication facilities.

Major factors that influence doubling time and their values are listed in Table 2. Other factors that affect long term doubling times are the PLFs, the cooling, reprocessing and fabrication times, construction schedule, and other delays. While the last two points are important for relating practice to theory, for the study, construction of reactors or FBR fuel-handing facilities is not assumed to be a limiting constraint; the only constraint to growth in this study is fissile availability.

SDT calculations

Figure 1 shows a simplified influence diagram for the model used for SDT calculations.

The equations used to calculate SDT are given below:

$$\mathrm{SDT} = \frac{\mathrm{FBOC} \times \mathrm{EF}}{(\mathrm{FG} - \mathrm{FL}) \times \frac{\mathrm{cycles}}{\mathrm{yr}}}, \tag{1}$$

FBOC = Fissile beginning of cycle,

$$\mathrm{EF} = \mathrm{Ext-Factor} = \frac{\text{cycle fissile inventory}}{\text{core fissile inventory}}, \tag{2}$$

FG = Fissile gained per cycle (net), FL = Fissile lost per cycle (outside reactor),

$$FL = FPL + {}^{241}Pu \text{ decay}, \quad (3)$$

FPL = Fission processing loss.

Experience and data on fast reactors and breeding

While there is considerable worldwide experience with fast reactors[18], only India, Japan, Russia, and China appear interested in fast reactors for near term electric power. Even countries such as France and UK that advocate reprocessing fuel to extract plutonium have put their fast reactor development on hold[19].

The primary source for data on breeding is the 1980 International Nuclear Fuel Cycle Evaluation (INFCE) report, compiled with international collaboration[20]. As many of the reactor configurations still exist only on paper, this report has been the primary source of data and information. Our analysis also draws extensively on this report.

In Bhabha's 1958 papers on using thorium, he pictured a doubling time of only 5–6 years for ^{233}U in the Th–^{233}U cycle[3,21]. INFCE projects this to be at least 70 years. This is understandable as Bhabha assumed commissioning of LMFRs, Liquid Metal *Fuel* Reactors (molten fuel). These were not built due to technical difficulties.

Publications by DAE scientists, especially Ramanna and Lee, suggest an optimistic picture for the growth of fissile material through breeding[22–24]. Table 3 shows doubling times from these publications. The INFCE data are based on CSDT as simply SDT × $\log_e$ (2). Most Indian data also use INFCE calculations, except for the carbide fuel. Indian calculations (given as Carbide-Lee in Table 3) suggest a shorter doubling time for thorium carbide. The authors have not discussed the reasons for this difference. For calculating doubling times, the Indian authors have used futuristic reactor performance scenarios, and yet-to-be-built reactor designs. The reactors described in INFCE are 1,000 MWe, while the Indian reactors are based on 500 MWe designs. This parameter alone need not affect the results, *if construction is not a constraint.* Reactor profiles used in our study are given in Table 2 (refs 20, 23). These are based on the advanced reactors designs, or post (year) 2000 designs.

Model parameters for analysis

The cycles/year are a direct function of PLF. All publications assume a sustained, average PLF of 0.75, which is high for any large FBR. On average, Indian lifetime PLFs have been more modest for PHWRs, averaging between 30 and 60% (refs 25, 26). In our model, the PLF was varied from 0.4 to 0.75, either parametrically or as a uniform distribution. The SDT model assumes that as the PLF decreases, the fuel residence time increases until the target burn-up is reached (100,000 MWd/ton).

Table 3. Published doubling times. The times in parentheses are converted from the published times by multiplication or division by $\log_e(2)$. The oxide, carbide, and metal calculations are from ref. 20; Carbide-Lee is based on ref. 23

	U–^{238}Pu cycle		Th–^{233}U cycle	
	SDT	CSDT	SDT	CSDT
Oxide	(25.7)	17.8	(155.8)	108.0
Carbide-Lee	14	(10)	72	(50)
Metal	(12.3)	8.5	(108.3)	(75.1)
Carbide	(14.7)	10.2	(101.0)	70.0

(all times in years).

The delay for cooling, reprocessing, and fabrication is also important, as the out-of-pile inventory is proportional to this delay. Publications assume a total one-year time for all the reprocessing activities for advanced reactors to be built after 2000. This seems unlikely for a number of reasons discussed later in the paper. For this analysis, we have varied the out-of-pile time between 2 and 3 years. This is for the standard wet chemical reprocessing route for extracting Pu or ^{233}U (Purex or Thorex processes respectively). There is an alternative method envisioned for metallic fuel, known as pyroprocessing or dry reprocessing[27,28]. For our model, we have assumed 1–2 years delay time for dry reprocessing. The assumption of one year reprocessing delay is optimistic given the time for cooling itself would be about eight months (the nominal time between reloads). This model also assumes that the entire cycle (in-pile plus out-of-pile) inventory of fissile material is available for a new reactor before it begins operation.

While calculating the doubling times, we have assumed the out-of-pile inventory needs to be based on the nominal cycles per year. If the PLF is lower than the nominal values, one might consider reducing the out-of-pile inventory needs, based on the actual (lower) load factor. However, this would condemn the reactor to lower PLFs in subsequent years as well. All the Ext-Factor calculations for out-of-pile inventory in our paper are, therefore, for nominal PLFs only. It is important to note that these calculations do not account for a buffer reserve against reprocessing/fabrication disruptions. It would be advisable to have at least two years' output from the largest reprocessing/fabrication facilities available as a buffer stock.

The publications assume losses during reprocessing at 1%. In our model, these are varied between 1 and 3%. The higher value appears closer to the current level of losses[29]. Finally, our model accounts for decay of ^{241}Pu

during cooling, reprocessing, and fabrication (^{241}Pu has a half-life of 14.4 years). This is important when accounting for the different reactivity worths (an approximate measure of fissionability) of Pu isotopes, which many publications fail to do. The decay losses of ^{241}Pu are magnified by its reactivity worth of 1.5. In addition, ^{241}Pu decay leads to a build-up of Americium-241, a neutron poison. This reduces the actual fissile worth even further[30].

Based on the above considerations we have calculated SDT for different parameter assumptions. We have also modeled the transition from SDT to CSDT, calculating the number of reactors required for reaching CSDT. For this calculation we have assumed fissile material availability to be the only constraint, and have ignored the continual decay of ^{241}Pu while awaiting the build-up of sufficient reprocessed fuel for commissioning a new reactor.

Plutonium from PHWRs

It takes a number of breeder reactors to reach CSDT. To offset this, the Department of Atomic Energy (DAE) suggests the utilization of Pu from reprocessed spent PHWR fuel. It is therefore important to model PHWR growth as a source of plutonium. The limit to PHWR growth is set by the domestic uranium availability and, in the model, this is varied between 30 and 90 thousand tons. The availability of uranium from other countries is not included in these calculations as there are restrictions imposed by the Nuclear Suppliers Group on the supply of nuclear materials to India.

It is necessary to know the PLF and the life of the reactors for calculating the fissile material throughput. For this calculation, we have assumed a 40 year PHWR life and an average PLF varying between 40 and 75%. In reality, the PLF may vary from year to year depending on the reactor performance or need for repairs. In some years, such as during coolant channel replacement, the PLF may even be zero.

After the current construction of the two 220 MWe twin units (RAPS 3&4 and Kaiga 1&2) is complete in approximately 1–2 years, NPC will construct the first pair of 500 MWe PHWR reactors (TAPS 3&4). This may be joined by the simultaneous construction of Kaiga 3-6 (220 MWe each), after which NPC plans to construct only 500 MWe reactors.

Reprocessing facilities for handling spent fuel from PHWRs are also important for producing Pu. India's current reprocessing capacity is around 140 tons of metal/yr, to be augmented by a new 100 tons/yr plant (KARP) scheduled to go on stream by 1997 (refs 15, 31). However, the existing 140 tons/yr plants are near the end of their service life. It is estimated that reprocessing plants take about 10 years to construct[32]. They also have a shorter service life than reactors, and this is varied in the model between 20 and 25 years. Their overall PLFs, are likely to lie between 50 and 60%. A realistic view of PHWR fuel reprocessing capacity is given in Figure 2.

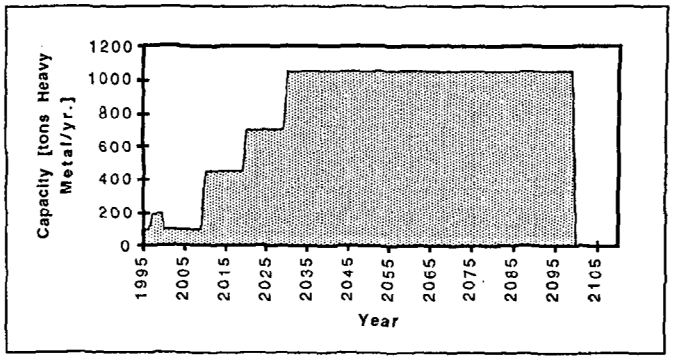


Figure 2. Possible PHWR spent fuel reprocessing capacity. This assumes a 10 year construction schedule, and 20–25 year plant life. Soon after KARP begins operation, the current plants go off-line. Subsequent constructions are all of 350 tons PHWR spent fuel/yr capacity. After a point, the only constructions are to replace decommissioned plants. By the turn of the century, it is more economical to reprocess spent fast fuel than PHWR fuel. This is because of the vastly greater fissile amount per ton processed.

The first 500 MWe fast reactor (PFBR) will be up for licensing soon. Its construction schedule is also a parameter for the model. After it begins operation, further construction of FBRs will depend on the experience of operating this test reactor and on other modifications to the design the experience may suggest. These refinements may involve a delay of two years before large-scale construction begins, and this is also included in the model. In addition, PFBR is U–Pu oxide fueled. If metallic fuel is to be used, it will mean an additional delay before large-scale deployment, as this technology is still in its infancy.

This initial period for FBR growth is the only stage where the years for construction are modeled as a constraint. After this period, the model assumes the availability of fissile material as the only constraint to the growth of FBRs.

Results

Doubling time

Table 4 shows the doubling times calculated using the model. For comparison, doubling times from INFCE and Indian publications are shown in Table 3. All fuel cycles used in our modeling have been derived from INFCE values (which are the basis for most Indian publications), except for the carbide cycle where we have also

Table 4. Statistics of SDT from model. This assumes 1–3% reprocessing losses, 2–3 years reprocessing delay (1–2 for dry), and 40–75% PLF. In approximately 20% of the Th–^{233}U oxide cycle SDT calculations (if the losses are high enough), there is no net breeding. For these cases, the SDT is taken as 500,000 years

		Oxide	Carbide-Lee	Metal	Metal-Dry	Carbide
^{238}U–Pu	min	31.1	23.3	12.0	9.1	16.5
cycle	median	48.8	35.8	17.9	13.8	24.6
	mean	50.3	37.1	18.4	14.3	25.8
	max	79.3	61.7	29.5	23.4	40.5
	std. dev.	10.8	8.0	3.7	3.0	5.4
Th–^{233}U	min	276	114	166		151
cycle	median	1,024	213	361		302
	mean	107,200	225	425		340
	max	500,000	473	1,117		871
	std. dev.	203,000	67	199		134

(all times in years).

Table 5. Importance analysis for SDT. This assumes 1–3% reprocessing losses, 2–3 years reprocessing delay (1–2 for dry), and 40–75% PLF. Sample size was 500 for all SDT calculations

		Oxide	Carbide-Lee	Metal	Metal-dry	Carbide
^{238}U–Pu	PLF	0.92	0.93	0.94	0.92	0.94
cycle	Reprocessing losses	0.29	0.24	0.17	0.16	0.18
	Delay until reload	0.28	0.37	0.26	0.38	0.35
Th–^{233}U	PLF	0.17	0.62	0.41		0.48
cycle	Reprocessing losses	0.96	0.74	0.89		0.85
	Delay till reload	0.03	0.25	0.17		0.15

used the Indian values, which are termed in this analysis as Carbide-Lee[23].

As can be seen from the two tables, the doubling times we have calculated are significantly longer than those in the INFCE and Indian reports. To determine the relative importance of the uncertain parameters (PLF, percentage losses, and delay until reload), we performed an importance analysis (absolute value of rank order correlation) on these, and the results are presented in Table 5. In this calculation, we assumed a first-order estimate of uniform distributions for the uncertain parameters, across the ranges specified before.

This analysis shows that for the ^{238}U–Pu cycle, PLF is the most important parameter in determining doubling time, while for the Th-^{233}U cycle, losses are of primary importance, followed by PLF.

PHWR growth

The growth of PHWRs is limited by domestic uranium supplies. Figure 3 shows the growth of PHWR capacities based on a high availability of domestic uranium (70,000 tons). The number of PHWRs going on stream is dependent on the number of construction teams building the reactors. The plateau seen in this figure represents equilibrium between new construction and decommissioning of old reactors.

The current construction capability is for 2 twin-reactor sites (of 220 MWe). Even an aggressive construction schedule of 5 teams constructing five twin units with a 7 year construction time would take until 2028 to reach peak capacity. This peak capacity of 18 GWe would last only about a decade, after which PHWR capacity would fall off rapidly.

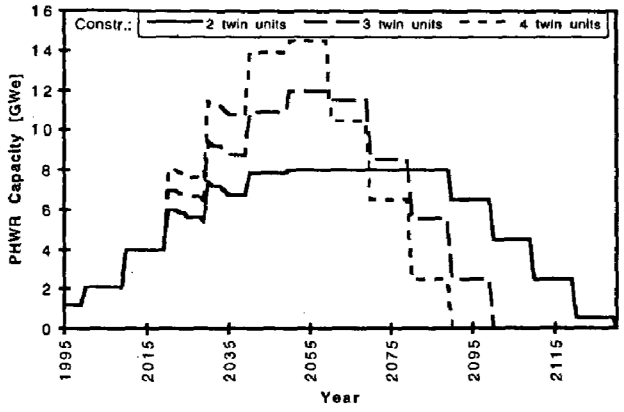


Figure 3. PHWR capacity. This assumes 70,000 tons of uranium available domestically, and varying numbers of teams constructing PHWRs (in pairs). Reactors are assumed to have a 40 calendar year life, and operate at a 55% lifetime PLF. Construction takes 10 years (including the initial infirm period at low power). Reducing the construction time increases the peak PHWR capacity, but reduces the duration of the plateau. It also does not affect FBR growth as reprocessing remains a constraint.

Reactors needed to achieve CSDT

DAE maintains that there is enough Pu available from PHWRs to allow approximately 25 GWe of FBRs to operate[12], at which point CSDT is achieved. We have modeled the growth of FBRs operating only on Pu from FBR breeding. (At some point, Pu from PHWRs would no longer be available, as those reactors would have exhausted the country's supply of U.) As expected, the initial doubling time for a small number of reactors begins with SDT, coming closer to CSDT with more reactors operating. However, there is an offset. This is because the fuel used to start up a new reactor will have to undergo burn-up and reprocessing before joining the pool of fissile material. Stated another way, a specific reactor won't be *outputting* any fissile material for use in a new reactor for a number of years after beginning operation. This shows that SDT as generally defined is itself subject to such an offset. In the long run, this offset is equal to the logarithm$_e$ of the time for burn-up and reprocessing. Table 6 compares the theoretical compound system doubling times with the actual doubling times.

Implications

The fact that nuclear power is capable of providing energy for many years is undisputed. A key question, however, is when that is likely to be realized, given all the constraints discussed in the earlier sections. A useful exercise would be to examine the contribution of nuclear power as a percentage of the total electrical capacity in the country. For this study, we have assumed the contribution of nuclear power to be determined as the sum of PHWR and FBR capacities. We have not included in this calculation light water reactors that are in operation (Tarapur) or are proposed to be acquired.

Table 6. Doubling for ^{238}U–Pu cycle, calculated CSDT vs actual doubling. This assumes 60% PLF, 2% losses, and a three-year reprocessing delay. The offset between the CSDT and actual long-term doubling is because fuel has to undergo burn-up and cooling/reprocessing/fabrication. The reason for there being a range for actual doubling is that after starting as many new reactors as possible, there is typically some leftover fissile material, which reduces the time needed till the next one can start

	Calculated SDT (model)	Implied CSDT = SDT × $\log_e(2)$	Actual doubling
Oxide	51.9	36.0	38–40
Carbide-Lee	38.2	26.5	29–30
Metal	19.0	13.1	16–17
Carbide	26.6	18.5	21–22

(all times in years).

Electrical capacity

To measure the share of nuclear power, one must make assumptions on the overall growth of electrical power in the country. The current installed capacity of around 85 GWe is growing at an annual rate of approximately 5% (ref. 8). While this growth rate will continue for many years due to the present very low per capita availability (around 350 kWh per year), it would eventually slow down. This study assumes that the current growth rate (parametrically varied between 4 and 6%) would continue for about 20–25 years (again, a variable), after which it would linearly decrease towards 0.5% growth as installed capacity approaches saturation capacity. Saturation capacity, in this context, is defined as the amount capable of providing for an annual per capita consumption of 2,300 kWh, based on an overall *net* PLF of 50%. This consumption is approximately equal to the current world average consumption and is well under the US per capita annual consumption of 11,000 kWh (refs 33, 34). The saturation population is estimated to be 1.3 billion. Even though the saturation consumption of electric power may appear low and depressing in meeting India's aspirations, it should be remembered that for achieving even this value, India would have to add 600 GWe of power at an estimated cost of $1 billion per GWe. After reaching saturation capacity, the electrical capacity would continue to grow at the residual growth rate, as is seen in other developed countries with stable populations. Figure 4 shows the growth of electrical capacity including the modeled growth of nuclear power. While these curves are subject to the assumptions mentioned above, they bring out the limited contribution that nuclear power is likely to play in the coming years.

Share of nuclear power

An important result of this study is the share of nuclear power within the country's generation capacity (Figure 5). These calculations are based on the ^{238}U–Pu cycle, as is currently planned by DAE. If the overall electricity growth rate is slower than that assumed in this model, it would likely affect nuclear power more than other forms of power, as it is a highly capital-intensive industry. This may further reduce the share of nuclear power in the overall electricity generation capacity. As the results showed the importance of PLF in doubling, we show in Figure 6 the share of nuclear power for different PLFs.

In the last section we showed that the growth of fissile material through breeding would not be as rapid as desired to allow an increase in nuclear power capacity in the near future. As seen in Figure 6, even with optimistic (but realizable, as recent NPC performance has indicated[7]) load factors, 70% PLF, the share of nuclear

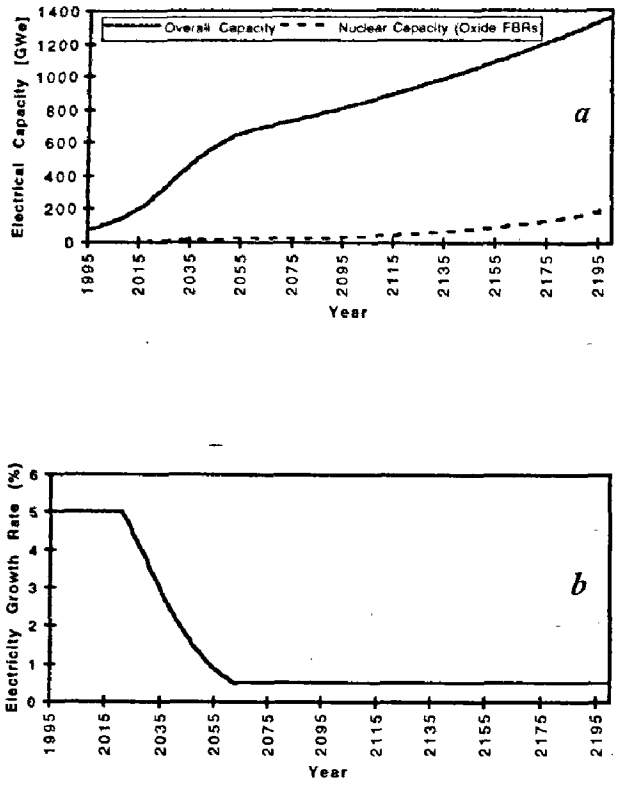


Figure 4. *a*, Installed electrical capacity in India. The nuclear curve is based on fissile availability using oxide FBRs with 0.6 PLF, 2% reprocessing losses, and a 3-year reprocessing delay. It also assumes 70,000 tons U available for PHWRs, and 4 teams constructing PHWR pairs; *b* Electricity growth rate in India. Both the above figures are based on a saturation population of 1.3 billion, overall net PLF of 0.50, a current growth rate of 5% that will continue for 24 years, and a 0.5% residual growth rate. The growth rate decreases linearly until saturation, after which the residual growth rate applies.

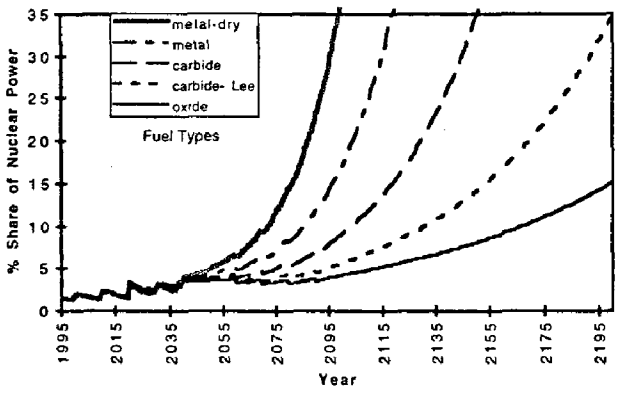
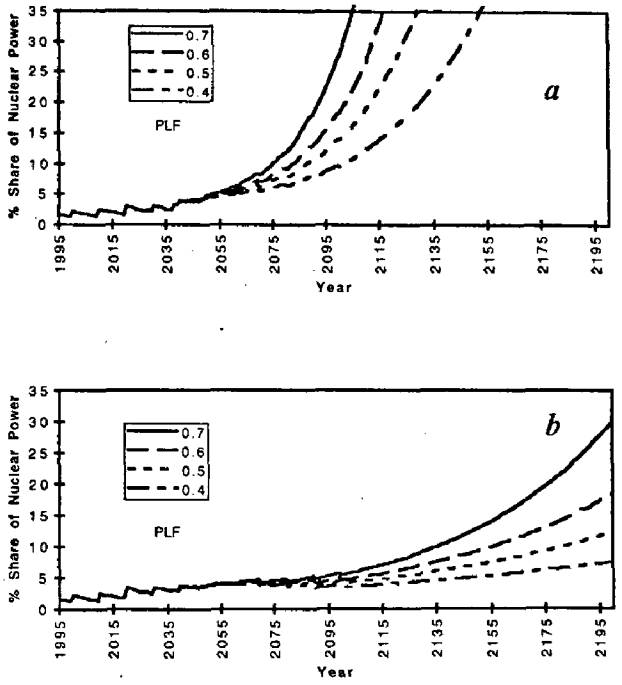


Figure 5. Share of nuclear power (by capacity), with ^{238}U–Pu cycle. The initial share is largely a function of PHWRs. FBR share is the upper-bound on possible capacity due to limited fissile availability. It does not account for contingencies or delays in any aspect of the fuel cycle. It assumes a 60% PLF, 2% reprocessing losses, and a 3-year reprocessing delay (2-year for dry). This share excludes contribution from any imported LWRs, as well as from TAPS 1&2 and RAPS 1&2, which are rated at only 520 MWe, and have only some 10–15 years of life left.



Figures 6. Share of electricity generation capacity by nuclear power with varying PLFs, oxide and metallic fuel cycles. PLF is the parameter most strongly affecting fissile material growth. Other parameters are chosen at optimistic levels, except delay. Losses are 1%, the available uranium is 90,000 tons, 4 teams are available to construct PHWRs, and the reprocessing delay is 3 years. This is a limit on the share based on fissile availability, and does not account for contingencies or delays in any part of the fuel cycle. *a*, Metallic fuel, ^{238}U–Pu cycle. *b*, Oxide fuel, ^{238}U–Pu cycle.

power will continue to remain low. This is due to India's choice of technology, which is path-dependent: the past performance of power reactors combined with their present attainments will strongly determine the future performance of nuclear power in India, as breeders depend on fuel produced from the past operation of reactors.

A succinct method of summarizing the limits on fissile growth based on breeding is comparing the growth rates corresponding to the calculated doubling times with the overall electricity growth rate, which is expected to remain at or above 5%. Table 7 shows the calculated growth rate of fissile material (corresponding to the actual long term doubling times shown in Table 6). If breeding is the only method of increasing fissile material, it would imply a *fall* in the share of nuclear power, at least until the electricity growth rate declines. The fissile material growth rate would not cross 5%, even with dry reprocessing for metallic fuel.

Discussion

Under the current policy scenario, the share of nuclear power will remain very low for the coming five or more decades. The limits on domestic uranium supplies, as well as construction constraints, will restrict PHWR growth. As we showed in an earlier section, breeding

Table 7. Fissile material growth rate (^{238}U–Pu cycle). This assumes 60% PLF, 2% reprocessing losses, and 3 year reprocessing delay

	Fissile growth rate
Oxide	1.73%
Carbide- Lee	2.31%
Metal	4.08%
Carbide	3.15%

will be very slow, especially based on oxide fuel as currently planned. Even the fastest breeding cycle, metallic fuel, will allow only a limited share for nuclear power in the coming decades. This analysis shows the assumptions made by DAE are overly optimistic and, unfortunately, unrealizable.

As shown in the earlier section, it is not possible to breed at a rate equal to SDT × $\log_e(2)$ as there is an offset imposed by burn-up and reprocessing. Even to approach CSDT requires the continuous and immediate use of the excess-bred material in new reactors. Such continuous growth, in all areas of the nuclear fuel cycle, is difficult to sustain.

The limits on construction due to licensing delays, capital availability and its high cost, infrastructure requirements and environmental concerns are very real. Multilateral funding for nuclear power is unavailable, and the Indian government is currently funding only two teams working on twin reactors. Without increasing the number of teams constructing reactors or any other facilities, the installed base will plateau as older units are decommissioned. Reducing the construction time does not alter the outlook much, and there is a limit to how many reactors can be constructed simultaneously. The record for the number of reactors under construction over a five-year period is held by the French, with thirty reactors (average)[32]. The Indian performance is far lower, and that too for reactors of smaller capacity.

Comparing fuel cycles

The importance analysis (Table 5) shows which parameters affect doubling time more strongly. Are there any choices that can be made amongst the various fuel types and fuel cycles that offer the promise of rapid breeding? Only metallic fuel coupled with dry processing appears to be an attractive fuel option. However, this technology has still not developed fully, let alone found commercial use. Even metallic fuel is not adequate to produce fissile material rapidly enough for increasing the percentage share of nuclear power.

Reprocessing and fabrication are batch processes. One can decrease the time needed for these by building extra capacity to operate in parallel. However, this becomes uneconomic after a certain capacity is reached as it increases the plant idle time. Extrapolating from INFCE Table XXI (ref. 20), the reprocessing requirements for 30 GWe of U–Pu oxide FBRs, for a two-year reprocessing and fabrication time, would be 900 tons metal per year! This is based on a 0.55 reprocessing PLF.

Reactors for near-term needs

As breeding is not a viable option for the short or medium-term, increased use of uranium in thermal reactors would be a promising option for increasing the contribution of nuclear power. In addition to intensifying the exploration of uranium ores in the country, India should consider entering into long-term agreements with other countries, with appropriate policy innovations, for importing uranium. This could be utilized not only in the forthcoming 500 MWe PHWRs, but in LWRs as well, which are often 1,000 MWe or larger in size.

The role of thorium in thermal reactors needs further examination. India is developing the Advanced Heavy Water Reactor (AHWR) as a means of utilizing thorium. The AHWR will use a small plutonium seed while extracting some 75% of its energy from thorium, which will be bred and burnt *in situ*[35,36]. The AHWR will provide numerous passive safety features, along with which the simplified design offers lower capital costs. However, the AHWR is only in the planning stages, and will likely take at least two decades before commercial deployment. There is also the problem of such reactors consuming Pu that would be needed for long-term breeding plans. AHWRs would offer very little (if not negative) growth in fissile material.

Conclusions

The Indian nuclear power programme has not met its stated objectives. The three stages of the previously enunciated plan appear to be non-realizable even in a time frame spanning five decades. The key to a reasonable growth of nuclear power will thus have to be the commissioning of a large number of pressurized heavy water and light water reactors. It is therefore necessary to restructure the nation's nuclear power strategy taking into account the present acute scarcity of power in the country and the relevance of nuclear power in India's portfolio of energy options.

Added note: Since submission of this paper, India conducted a series of underground nuclear explosion tests in May 1998. The short- and long-term consequences of these tests for meeting the targets of the Indian nuclear power programme are yet to be assessed.

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What's old and what's new about neutrinos

R. Cowsik

To save the laws of conservation of energy, linear and angular momentum, Pauli suggested the possible existence of an elementary particle possessing no electric charge and very little mass called the neutrino, which was emitted in β-decay of the nuclei. Subsequent work over the decades has proved the existence of this particle and has proved its importance not only to the physics of elementary particles but also to astrophysics and cosmology. This article is triggered by the recent experiments which show that the neutrinos of one flavour, as they speed through space, periodically transmogrify onto other flavours. This discovery is comparable to the discovery of the electron in that it provides the basis to new physics beyond the currently accepted Standard Model.

CONSIDERABLE excitement amongst physicists and astronomers was recently sparked off by the announcement of the confirmation of neutrino oscillations by the Super-Kamiokande collaboration at the XVII International Conference of Neutrino Physics and Astrophysics held in Takayama, Japan early this June. For, the oscillations imply that the neutrino, hitherto regarded as being massless in the Standard Model of the particle physics, has a finite rest mass. Thus, in a dramatic reversal of roles the tiny neutrino may indeed dominate the gravitational dynamics of the whole Universe.

R. Cowsik is at the Indian Institute of Astrophysics, Bangalore 560 034, India.