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India's Nuclear Breeders: Technology, Viability, and Options

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Abstract: India's nuclear power program is based on indigenous materials and technology, with the potential for providing energy security for many centuries. This paper examines the technical validity 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 slow growth of fast breeder reactors. In addition, domestic uranium reserves restrict growth of Pressurized Heavy Water Reactors (PHWRs), which are likely to be the main contributors to nuclear capacity in the short-term. The Th-U233 cycle in fast breeders does not appear attractive, and, for the U238-Pu cycle, only metallic fuel offers hope of rapid increase in available fissile material. 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.

Keywords: Nuclear Power, Fast Breeder Reactors, Three-Phase Plan, Doubling Time, Plutonium

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Abbreviations and Acronyms

AHWR	Advanced Heavy Water Reactor
ANL	Argonne National Laboratory
BWR	Boiling Water Reactor (a type of LWR)
CANDU	Canadian Deuterium Uranium (standard PHWR)
CSDT	Compound System Doubling Time
CTBT	Comprehensive Test Ban Treaty
DAE	Department of Atomic Energy (India)
EF	Ext-Factor (or Ex-Factor in some references)
FBR	Fast Breeder Reactor
FBTR	Fast Breeder Test Reactor
GWe	Gigawatt-electric
HLW	High Level Waste
IFR	Integral Fast Reactor
IGCAR	Indira Gandhi Center for Atomic Research
INFCE	International Nuclear Fuel Cycle Evaluation (conf.)
KARP	Kalpakkam Reprocessing Plant
LEU	Lightly Enriched Uranium
LMFBR	Liquid Metal Fast Breeder Reactor
LMFR	Liquid Metal Fuel Reactor
LWR	Light Water Reactor
MWd	Megawatt-day
MWe	Megawatt-electric
MWt	Megawatt-thermal
NPC	Nuclear Power Corporation (India)
NPT	Non-Proliferation Treaty
PHWR	Pressurized Heavy Water Reactor
PLF	Plant Load Factor
Pu	Plutonium
PWR	Pressurized Water Reactor (a type of LWR)
RAPS	Rajasthan Atomic Power Station
RDT	Reactor Doubling Time
SDT	System Doubling Time
SQ	Significant Quantity- fissile amount req. for a bomb
TAPS	Tarapur Atomic Power Station
Th	Thorium
U	Uranium

Introduction

India's Nuclear Power Program

India's nuclear power program 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 (Bhabha, 1955; Bhabha, 1958; Bhabha and Prasad, 1958).

As India preferred a technology that did not require uranium enrichment, then considered expensive, it was decided to use Pressurized Heavy Water Reactors (PHWRs) for the first stage.[1] These are of the so-called Canadian Deuterium Uranium (CANDU) design.

Table 1: India's Planned Three Phase Nuclear Power Program

	I	II	III
Reactor Type	PHWR	LMFBR	LMFBR *or* HWR
Fuel	Natural Uranium (once-through)	(U-Pu) oxide *or* carbide *or* metal	Thorium-U233 Cycle
Potential	~340 GWe-yr	~16,000 GWe-yr	~168,000 GWe-yr
	Produces Pu in spent fuel	High breeding	Low breeding

Based on the potential for PHWRs, this implies a domestic availability of approximately 60,000 tons natural uranium.

A long-term goal of India's program is to use the vast reserves of thorium available in the country. As thorium is a fertile[2] material and not a fissile one, it is necessary initially to use another fissile material for fuel and also to breed fissionable

[1] The trade-off is that PHWRs require large supplies of heavy water as moderator-cum-coolant. This is a very expensive material and, in India, accounts for double the costs of natural uranium fuel (NPC, 1993).

[2] Fertile materials are those which do not typically undergo fission in a nuclear reactor. These can convert to fissile materials by absorption of a neutron (typically inside a nuclear reactor).

uranium 233 (U233) from the thorium-U233 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[3] (FBRs). 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 program as formulated by the Indian Atomic Energy Establishment is shown in Table 1 (Chidambaram and Ganguly, 1996; Chidambaram, 1995). We can see that the planned use of FBRs involves closing the fuel cycle, reprocessing the discharged fuel from fast (or thermal) reactors.

India is now on the threshold of the second phase of its nuclear power program, using FBRs. India has successfully operated a 40 MWt (Megawatt-thermal) Fast Reactor since 1985, which recently began limited production of electricity. This reactor, the Fast Breeder Test Reactor (FBTR), is the first of a series of planned FBRs, beginning with a 500 MWe U-Pu oxide fueled Prototype Fast Breeder Reactor (PFBR). Today, India operates eight 220 MWe (original rating) PHWRs, with 4 more under construction. In addition to these, India's power reactors include two 160 MWe BWRs, which were constructed before the PHWRs.[4] India also operates numerous research reactors, and has indigenous reprocessing and fuel fabrication facilities.

The Nuclear Power Corporation (NPC) was formed in 1987 as the commercial arm of DAE. As of 1996, NPC had operating losses on the order of 10 billion Rupees[5], mostly because of poor load factors, long construction times, high interest rates, and defaults on payments by state electricity boards[6] (DAE, 1995b). NPC is unlikely to begin commercial deployment of fast reactors until the 500 MWe PFBR or a subsequent reactor operates successfully and safely, and shows economic potential. Assessed from the present technology status and fissile materials availability, India, in the coming decade, is unlikely to commission the third stage power reactor that uses U233 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

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[3] A Fast Reactor is one with no moderator, and typically uses Liquid Metal (sodium) as a coolant. It has the capability of breeding, i.e., producing more fissile material than it consumes. However, one can operate a fast reactor as a burner of fissile material, as the Japanese propose to do, by simply not using fertile material in the blankets.

[4] The BWRs and the first pair of PHWRs are under international safeguards, meaning all inputs and outputs are subject to scrutiny.

[5] Dec. 1996 exchange rate: 1 US$ = 35.5 Indian Rupees.

[6] Default amounts account for about 3/4 of the losses.

2210 MWe is actually de-rated to approximately 1750 MWe[7], and operates at overall (lifetime) plant load factors (PLFs) of only around 50%. This compares unfavorably to the March 1996 world average lifetime PLFs of nearly 70% for Pressurized Water Reactors (PWRs) and 66% for PHWRs (Nuclear Engineering International, 1996). Though the performance of the most recent year shows improvement in PLFs to 67% (NPC, 1997), these may not be adequate to compensate for a less than optimal performance in the past decades.

Concerns on performance aside, the importance of the nuclear power lies in its potential as well the energy security it might provide. While the current share of nuclear power is only about 2% by generation (Ministry of Power, 1996), 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 (The Hindu, 1996). 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 (Bahadur, 1996).

In this paper, we discuss the technology and viability of breeder reactors using different fuels in India's nuclear power program. 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. We compare doubling times across the fuel types: oxide, carbide, and metallic,[8] while mainly for the U238-Pu cycle, but also for the Th-U233 cycle.[9]

Based on the results obtained from our modeling, we discuss the performance of India's Nuclear Power Program 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.

This paper also briefly examines international acceptability, especially regarding plutonium breeding and reprocessing. India is not a signatory to the Non-Proliferation Treaty (NPT), and, along with Pakistan and Israel, is classified as a threshold nuclear weapons state ("rogue state" to some). Because of that fact, India is isolated from the world in terms of both nuclear technology and nuclear materials (such as fuels). Even for countries operating under full-scope safeguards, such as Japan, there are international concerns (especially American) over the use of plutonium (Solomon, 1993; Oye, Skolnikoff *et al.*, 1995).

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[7] RAPS 2 is currently undergoing coolant channel replacement; for the coming years, the usable capacity will only be 1550 MWe.

[8] Metallic fuel as considered consists of a ternary alloy of uranium, plutonium, and zirconium.

[9] Appendix A contains brief descriptions of nuclear reactor designs, as well as fuel cycle equations.

Model and 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 between 30-70 thousand tons (DAE, 1995a; Sundaram, 1996). 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 (1997) 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, or only 407 GWe-yr based on 70,000 tons uranium.[10]

Through fuel reprocessing and breeding in FBRs, the fuel supplies can be extended by a factor of 50. This is achieved by converting fertile U238 (found in natural uranium as well as spent fuel) into fissile Pu239 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.

Presented below is a sample of the flow of fissile material (Balakrishnan, 1990; Krishnan, 1996):

1 ton natural U used in PHWRs	$\rightarrow$	3.5 kg Pu in spent fuel
$\frac{118 \text{ tons U/yr.}}{\text{GWe PHWRs}}$ {@ .685 PLF}	$\rightarrow$	413 kg Pu/yr.

Doubling Time

Plutonium processed from the spent PHWR fuel can be loaded into a FBR for breeding more plutonium from U238. 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 (Waltar and Reynolds, 1981; Marshall, 1983):

[10] This is at 6,700 MWt-days/ton rated burn-up.

Reactor Doubling Time (RDT)

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 (SDT)

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 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 Eqn. 2. SDT, by definition, is longer than RDT.

Compound System Doubling Time (CSDT)

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 $\times$ $\log_e(2)$. 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 .7 $\times$ SDT, approximately $\log_e(2)$ $\times$ 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.

Below is a list of factors affecting doubling time:

- Initial in-pile inventory
- Plant Load Factor (PLF)
- Breeding ratio
  — Function of reactor design and fuel choice
- What fraction of the fuel undergoes fission (burn-up)
- Reloading fraction
- Cooling, reprocessing and fabrication times
- Other delays
- Construction schedule

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 is fissile availability.

SDT Calculations

Presented in Appendix B is a simplified influence diagram for the model used for SDT calculations.

The equations used to calculate SDT are given below:

$$SDT = \frac{FBOC \times EF}{(FG - FL) \times \frac{cycles}{yr}} \qquad \text{Equation 1}$$

$$FBOC = Fissile Beginning of Cycle$$

$$EF = Ext - Factor = \frac{cycle\ fissile\ inventory}{core\ fissile\ inventory} \qquad \text{Equation 2}$$

$$FG = Fissile\ gained\ per\ cycle\ (net)$$

$$FL = Fissile\ lost\ per\ cycle\ (outside\ reactor)$$

$$FL = FPL + Pu - 241\ Decay \qquad \text{Equation 3}$$

$$FPL = Fission\ Processing\ Loss$$

Existing Experience and Literature on Fast Reactors and Breeding

Appendix C (Paranjpe, 1992) shows the world's experience with fast reactors. Today, only India, Japan, Russia, and China are interested in fast reactors in the near future. Even countries such as France and UK that advocate reprocessing fuel to extract plutonium have put their fast reactor development on hold (Energy Committee, 1990).

The primary source for breeding data is the 1980 International Nuclear Fuel Cycle Evaluation (INFCE) report, compiled with international collaboration (INFCE, 1980). 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 role of thorium, he pictured a doubling time of only 5-6 years for U-233 in the Th-U233 cycle (Bhabha and Prasad, 1958; Dayal, Paranjpe *et al.*, 1958). INFCE pictures this as at least 70 years. The reason was Bhabha assumed use of LMFRs, Liquid Metal *Fuel* Reactors (molten fuel). These were never built due to technical difficulties.

Table 2: Published Doubling Times

	U-238 Pu Cycle		Th U-233 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 times in parentheses are converted from the published times by multiplication or division by loge(2). The oxide, carbide, and metal calculations are from INFCE (1980); carbide- Lee is based on Lee *et al*. (1990).

Publications by DAE scientists, especially Ramanna and Lee, suggest an optimistic picture for the growth of fissile material through breeding (Lee and Kimura, 1988; Lee, John *et al*., 1990; Ramanna and Lee, 1986). Table 2 shows doubling times from these publications. The INFCE data are based on CSDT as simply SDT $\times$ $\log_e(2)$. Most Indian data also use INFCE calculations, except for the carbide fuel. Indian calculations (given as Carbide-Lee in Table 2) 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 Appendix D (Lee, John *et al*., 1990; INFCE, 1980).

Model Parameters for Analysis

The cycles/year are a direct function of PLF. All publications assume a sustained, average PLF of .75, which is high for any large FBR. On average, Indian lifetime PLFs have been more modest for PHWRs, averaging between 30 and 60% (IAEA, 1995; Bhoje, 1996b). In our model, the PLF was varied from .4 to .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).

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 U233 (Purex or Thorex processes respectively). There is an alternative method envisioned for metallic fuel, known as pyro-processing or dry reprocessing (Chang, 1989; Battles, Miller *et al.*, 1992).[11] 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.[12]

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 (Lee, 1996). Finally, our model accounts for decay of Pu241 during cooling, reprocessing, and fabrication (Pu241 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 Pu241 are magnified by its reactivity worth of 1.5. In addition, Pu241 decay leads to a build-up of Americium-241, a neutron poison. This reduces the actual fissile worth even further (Solomon, 1993).

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

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[11] Pyrometallurgical reprocessing (pyro-processing) is an electrolytic separation process. It is expected to offer a number of advantages, including lower capital and O&M costs, as well as the possibility of using spent fuel with less cooling (Chang, 1989; Chang and Till, 1986; Battles, Miller *et al.*, 1992).

[12] Reprocessing plant sizes depend on the number of reactors served. An issue worth considering is that co-location or transportation to off-site plants affects risks of proliferation (diversion) and safety.

ignored the continual decay of Pu241 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 (Editors, 1995; Krishnan, 1996). 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 (Krishnan, Tongia *et al.*, 1997). 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 Appendix E.

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 3 shows the doubling times calculated using the model. For comparison, doubling times from INFCE and Indian publications are shown in Table 2. 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 used the Indian values, which are termed in this analysis as Carbide-Lee (Lee, John *et al.*, 1990).

Table 3: Statistics of SDT from Model

		oxide	carbide-Lee	metal	metal-dry	carbide
U238-Pu Cycle	min	31.1	23.3	12.0	9.1	16.5
	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-U233 Cycle	min	276	114	166		151
	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)

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-U233 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.

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 4. In this calculation, we

assumed a first-order estimate of uniform distributions for the uncertain parameters, across the ranges specified before.

Table 4: Importance Analysis for SDT

		oxide	carbide-Lee	metal	metal-dry	carbide
U238-Pu Cycle	PLF	0.92	0.93	0.94	0.92	0.94
	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-U233 Cycle	PLF	0.17	0.62	0.41		0.48
	reprocessing losses	0.96	0.74	0.89		0.85
	delay till reload	0.03	0.25	0.17		0.15

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.

This analysis shows that for the U238-Pu cycle, PLF is the most important parameter in determining doubling time, while for the Th-U233 cycle, losses are of primary importance, followed by PLF.

PHWR Growth

The growth of PHWRs is limited by domestic uranium supplies. Figure 1 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.

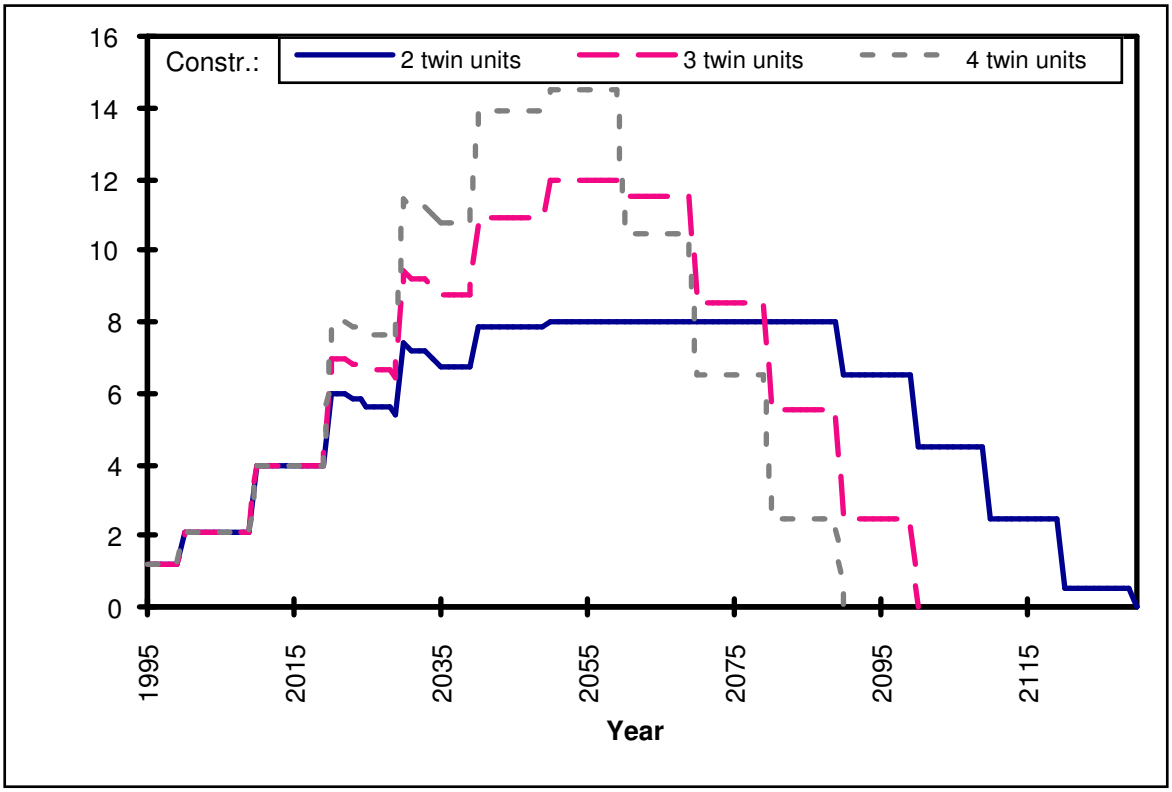

Figure 1: PHWR Capacity Projections

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.

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.

Reactors Needed to Achieve CSDT

DAE maintains that there is enough Pu available from PHWRs to allow approximately 25 GWe of FBRs to operate (Sundaram, 1996), 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 5 compares the theoretical compound system doubling times with the actual doubling times.

Table 5: Doubling Times for U238-Pu Cycle, Calculated vs. Actual Doubling

	Calculated SDT (model)	implied CSDT = SDT ∞ $\log_e(2)$	Actual Doubling
oxide	51.9	36.0	38 to 40
carbide -Lee	38.2	26.5	29 to 30
metal	19.0	13.1	16 to 17
carbide	26.6	18.5	21 to 22

(all times in years)

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.

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.

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% (Ministry of Power, 1996). 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 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 (DOE/EIA, 1995; Rodriguez, 1996). 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. Appendix F 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 2).[13] These calculations are based on the U238-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 Appendix G the share of nuclear power for different PLFs.

[13] 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.

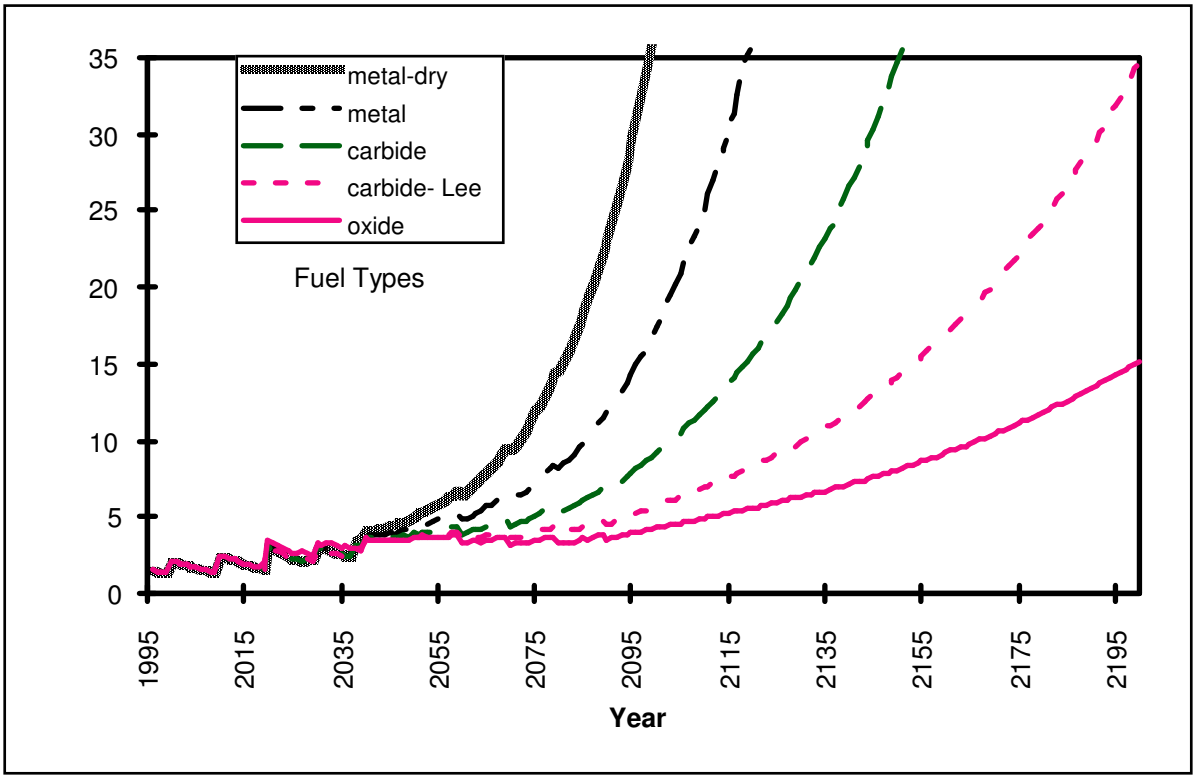

Figure 2: Share of Nuclear Power (by Capacity), for U238-Pu Cycle

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).

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 Appendix G, even with optimistic (but realizable, as recent NPC performance has indicated (NPC, 1997)) load factors, 70% PLF, the share of nuclear 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 6 shows the calculated growth rate of fissile material (corresponding to the actual long term doubling times shown in Table 5). 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.

Table 6: Fissile Material Growth Rate (U238-Pu Cycle)

	Fissile Growth Rate
oxide	1.73%
carbide- Lee	2.31%
metal	4.08%
carbide	3.15%

This assumes 60% PLF, 2% reprocessing losses, and 3-year reprocessing delay.

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 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 $\times$ $\log_e(2)$ 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) (Krishnan, Tongia *et al.*, 1997). The Indian performance is far lower, and that too for reactors of smaller capacity.

What Can Be Done?

Comparing Fuel Cycles

The importance analysis (Table 4) 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

When comparing fuel cycles for rapid growth of fissile material, most variables affect SDT monotonically. PLF[14] is an important variable, not only for reducing doubling time but also for the production of greater (and more economic) power. Losses, which should be minimized for safety and economic reasons, affect SDT most strongly in the Th-U233 cycle.

Reprocessing and fabrication are batch processes. One can decrease the time needed for reprocessing by building extra capacity to operate in parallel. However, this is uneconomic after a point as it increases the plant idle time. Extrapolating from INFCE Table XXI, the reprocessing requirements for 30 GWe of U-Pu oxide FBRs, for a two-year reprocessing and fabrication time, are on the order of 900 tons heavy metal per year! This is based on a .55 reprocessing PLF. It is important to note that reprocessing fuel with such a high burn-up has not been economically tested. Such highly fissile fuel will typically have a much lower throughput than spent LWR fuel for safety reasons (Krishnan, 1996). Clearly, the economics of reprocessing need further examination.

Metallic fuel is also attractive if rapid breeding is the goal. However, it will have a longer lead-time, as the technology is less developed in India. The bulk of the experience with metallic fuel lies with US national labs, such as Argonne National Laboratory (ANL).

Costs are difficult to assign to these technologies, if only because of limited worldwide experience. In addition, Indian conditions are likely to be different. What is known is that dry reprocessing promises to be cheaper (capital- and O&M-

[14] PLF improvement is not directly a function of reactor configuration or fuel choice, but is a very important part of rapid breeding.

wise) than wet reprocessing. In addition, fabrication in the Th-U233 cycle is likely to be more expensive than for the U238-Pu cycle, due to the γ-radioactive daughter products. It requires remote fabrication. In terms of fuel costs, the metallic core requires the least inventory[15], and should thus be the cheapest, regardless of reprocessing costs.

In terms of flexibility, reactors designed for metallic fuel can accommodate oxide fuel, but the reverse is generally not true (Bhoje, 1996a).

Metallic Fuel Development

Metallic fuel is the only one easily capable of dry reprocessing, which offers many advantages (listed below). It improves doubling time because of a reduced Ext-Factor, as well as a reduced offset from CSDT. However, while reprocessing and fabrication can be sped up in this process, the minimum cooling time will be constrained to the time until the next reload. In fact, there are incentives to increase the burn-up (to over 150,000 MWd/ton), resulting in a longer reload delay[16]. This is because each reload (and simultaneous repairs and inspections) results in a down-time on the order of one or more months (Krishnan, 1996). Also, each reprocessing leads to losses.

There are other reasons why the metallic fuel cycle in conjunction with pyro-processing is attractive. Safety is an important consideration. ANL has shown that metallic fuel can offer passive safety, as was demonstrated by their 1986 test at EBR II (Chang and Till, 1989; Chang, 1989). While metal fuel has a lower melting point than oxide, the increased thermal conductivity more than compensates for this. Dry reprocessing offers the possibility of actinide partitioning, meaning that many of the long-term radioactive products can be reloaded into the reactor and burnt. If one burns not only the plutonium but also other minor actinides such as Americium, Neptunium, Curium, etc., the resulting high level waste (HLW) will decay to lower radiation levels much sooner than standard HLW, in hundreds instead of thousands of years (Chang and Till, 1989; Hannum, 1991). Appendix H shows the effects of actinide burning on radiation levels from wastes (Oye, Skolnikoff *et al.*, 1995).

Non-Proliferation

One other crucial reason in favor of dry reprocessing is non-proliferation. Spent fuel is not considered direct bomb material, because of its radiation levels as

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15 This is for the U238-Pu cycle.

16 Metal fuel is capable of higher burn-ups because of a harder neutron spectrum, resulting in lower excess criticality at the beginning of a cycle (Chang and Till, 1989).

well as its poor fissionability. It has to be reprocessed to extract plutonium, and this is the reason that reprocessing is of major concern regarding proliferation. Even countries that have signed NPT, such as Japan, have been under pressure not to use plutonium (safeguarded or not) (Oye, Skolnikoff *et al.*, 1995; Solomon, 1993). The throughput of plutonium envisioned for large-scale FBR deployment is on the order of hundreds of tons per annum. Satisfactory techniques for safeguarding the large quantities of plutonium found in reprocessing plants are still under evolution, and this may turn out to be a major issue. However, the short-term policy of simply disallowing reprocessing is not necessarily the answer.

Because of concerns arising out of large-scale availability of plutonium, the US strongly discourages not only setting up new reprocessing plants but also development of new technologies for reprocessing. This concern even extends to the further development of FBR technology. One technology that could address these concerns is pyro-processing.

The Purex process, the conventional method for reprocessing fuel in the U238-Pu cycle, was developed with the goal of obtaining relatively pure plutonium. That is no longer the goal as fast reactors can accept plutonium with impurities. In dry reprocessing, plutonium is not isolated. A mixture of uranium and plutonium with minor actinides and some fission products (making weapons virtually impossible) can be the end product (Chang, 1989; National Research Council, 1992). The smaller size of the dry reprocessing plant and fabrication facility leads to the concept of an Integral Fast Reactor (IFR) (Chang and Till, 1986; Chang, 1989; Battles, Miller *et al.*, 1992). This involves co-location of the dry reprocessing facility with the reactor, reducing transportation costs and risks, both safety and of diversion.

The Th-U233 cycle, though slower in growth, offers some non-proliferation advantages. U233 (fissile) can be denatured by mixing it with U238 (fertile). Weapons grade enrichment would then require isotopic separation.

Other Options for Nuclear Power in India

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 (Kakodkar and Balakrishnan, 1990; Balakrishnan, Vyas *et al.*, 1994). 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

For renewed interest in nuclear power by policy-makers, NPC must complete a number of steps currently planned. These include improving the current PLFs, reducing the construction time through better management of manpower and funds, and increasing the reactor size to 500 MWe. This last will go a long way in providing increased power for a given construction project.

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. In light of the results of this study, we feel that importing nuclear reactors must also be considered. With international technology will also come the possibility of international funding, which will be a boon for such a capital-intensive industry. Already, there are negotiations proceeding for import of two large-sized Russian Light Water Reactors.

Policy Direction

The Indian nuclear power program is more than four decades old, and DAE is unwaveringly following the path charted in the early years of the program. As far as known, the current Three-Phase Plan has not been reviewed or modified (schedule changes aside). India continues to believe separating Pu is necessary, with the intent of eventually converting thorium into U233 in the third phase.

The main problem DAE sees is the lack of funding, which has delayed its first phase drastically. India today only has some 2 GWe of nuclear power, instead of the 10 GWe as scheduled till recently. The unspoken but experienced factors limiting growth have been export embargoes from nuclear producers (Katz and Marwah, 1982), the absence of a proven, reliable, and acceptable breeder technology with a fast enough breeding gain to provide energy security, and limited realization of the U233 cycle. Meanwhile, India continues to stockpile spent fuel from its power reactors. Crude calculations show that India had almost three tons of plutonium in the form of unsafeguarded spent fuel as of March 1997, and produces hundreds of kilograms of Pu in the form of spent fuel every year (at the current PHWR capacity

and modest load factors).[17] Using the definition of 8 kg plutonium as significant quantity (SQ) (Oye, Skolnikoff *et al.*, 1995), an amount enough for a nuclear weapon, India could choose to reprocess this fuel and make hundreds of nuclear devices[18].

India has not signed NPT or the Comprehensive Test Ban Treaty (CTBT), citing the discriminatory nature of these treaties as well as their explicitly not mandating a time-frame for elimination of nuclear weapons (Subrahmanyam, 1985). India was one of three countries that voted against CTBT. As a non-member of NPT, CTBT, or the London Club, India is the only country with proven nuclear weapons capabilities and fuel processing experience. In spite of a perceived isolation from treaties, verification and control regimes, India has exercised great maturity in safeguarding its materials and technology. As if to emphasize this restraint, India has not yet reprocessed spent fuel from its power reactors on a large scale, and has refused to yield to the demands of certain countries to share its plutonium technology (Hussein, 1992).

The US nuclear euphoria of earlier decades, which led to a nuclear capacity of almost 100 GWe being installed, has withered away. Other than plutonium fears, which have led to an almost total abandonment of reprocessing and breeder technology development[19], the major concerns are environmental, safety, and lack of commercial viability compared to other conventional and non-conventional energy sources.

The present standoff between other nations and India on these issues has neither brought down worldwide concerns of nuclear proliferation nor provided reliable and safe power for India. We see four policy options available, and these are summarized in Table 7.

The first option is to maintain the status quo of India going alone along its Three-Phase Plan, with no transfer of technology or material (fuel) from other countries. This is a lose-lose situation. This study shows that given the present technology status and level of implementation, breeding with its touted energy security is not going to be realizable at all. The world will have to be content with India's track record in securing its nuclear technology and materials. Reduction of plutonium stockpiles and bringing India under international safeguards will remain elusive.

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[17] These amounts exclude safeguarded reactors (RAPS 1&2, TAPS 1&2).

[18] Of course, this plutonium is not weapons-grade, meaning that a larger quantity than 8 kg is needed for criticality.

[19] The recent announcement by DOE to consider burning weapons-grade plutonium in reactors (Wald, 1996) (along with vitrification as the other option) shows that the US government might change its views.

There is the possibility of the US practicing a benign neglect of India's nuclear power program, not opposing India's acquiring technology from other members of the London Club (second quadrant of Table 7). While India might consider this a win situation, this may merely continue India's dependence on the Three-Phase Plan, which this analysis has shown to be non-viable. The present high costs of nuclear power[20] and uncertainties regarding imports (especially without US approval) make widespread deployment of even thermal power reactors difficult. The world will not benefit, as India will continue to pursue its use of plutonium without participating in international efforts to contain plutonium. Only those reactors and materials of international origin would be subject to (islanded) safeguards.

The policy opposite to the status quo, of persuading India to sign NPT and CTBT, is no option at all. All political parties in India are united in their opposition to India signing these treaties, which they perceive as perpetuating the current nuclear hegemony and bringing India back in the colonial era. In addition, even signing these treaties does not appear to ensure a free hand towards pursuing plutonium use for power needs.

The fourth option is for other nations to agree to help set up thermal nuclear reactors with an understanding that all the spent fuel would either be shipped back to the supplier or remain under international safeguards. The irradiated fuel would not be reprocessed until technologies that address proliferation concerns are developed. India would also agree to not step up its plans for reprocessing the current spent fuel, and would bring all domestic power reactor fuel under international safeguards. It would also open up all its power reactors for inspection. This option is worth considering as it will help India increase nuclear power contribution in the short term while increasing worldwide collaboration for plutonium containment and development of proliferation-resistant reprocessing options. The world will also benefit with a "new" market for its nuclear industry.[21]

At present, India is in the first quadrant of Table 7. The US would ideally like to see India in the third quadrant, while India would want the US to accept the second. The fourth quadrant appears to be a worthwhile option, with benefits for India and the US. The US, guaranteeing supplies of imported uranium[22], would

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[20] Estimated final cost for a 500 MWe reactor (first of a kind) is 80 million rupees/MW (Sethi, 1996).

[21] Since completion of this work, our attention was drawn to a January 1997 article by Haass and Rose suggesting Indo-US cooperation in nuclear power reactor development, but in a general manner (Haass and Rose, 1997).

[22] The proven worldwide reserves of uranium under $130/kg are estimated at 4400 thousand tons, with an equal amount directly inferred. The current worldwide requirement is approximately 60

take the pressure off India's using FBR technology based on plutonium obtained through wet reprocessing. India's energy concerns would be met, as would US concerns about plutonium proliferation.

To make this last option realizable, laws in both countries will have to be reinterpreted, or even modified. These may demand new presidential waivers and policy directives. However, the improvements over the status quo for all concerned are worth the pursuit of this option.

Table 7: Policy Options for India and the US

IV. Evolution of Cooperation: All power reactors open for inspection Import of nuclear technology and material (once-through), with islanded safeguards Dry reprocessing developed; no Pu separated	**I. Status Quo**: No cooperation No future importing LWRs India continues developing FBRs India still has tons of Pu available from the Purex process, all of which remains unsafeguarded
III. Full Cooperation: India signs NPT India gains full access to Uranium, LWRs US gains "victory", with enhanced non-proliferation Reprocessing remains an issue	**II. Benign Neglect**: Limited access to imported Uranium, even LEU for LWRs India continues Three-Phase Plan US concerns about Pu usage not addressed

thousand tons uranium per year, for an installed capacity of approximately 350 GWe (Krishnan, Tongia *et al.*, 1997). These reserves are sufficient to allow India to import uranium and use it on a once-through basis for many decades.

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Appendices

Appendix A Nuclear Reactors and Reactions

Table 8: Reactor Designs

	Characteristics	Status
LWR	· Light-water coolant/moderator · Req. Lightly-Enriched Uranium	Commercial
PWR	· Type of LWR where coolant doesn't boil	Most popular design today
PHWR (or CANDU)	· Heavy-water coolant/moderator · Uses natural uranium	Commercial
LMFBR	· Unmoderated design using liquid metal coolant · Req. high enrichment	Under development

Nuclear Cycle Equations:

Th-U233 Cycle

$$^{232}_{90}\mathrm{Th} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{233}_{90}\mathrm{Th}\ (\text{unstable}) \xrightarrow{\beta^-} {}^{233}_{91}\mathrm{Pa} \xrightarrow{\beta^-} {}^{233}_{92}\mathrm{U} \qquad \text{Equation 4}$$

U238-Pu Cycle

$$^{238}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \longrightarrow {}^{239}_{92}\mathrm{U}\ (\text{unstable}) \xrightarrow{\beta^-} {}^{239}_{93}\mathrm{Np} \xrightarrow{\beta^-} {}^{239}_{94}\mathrm{Pu} \qquad \text{Equation 5}$$

Subsequent absorption of neutrons by Pu239 leads to higher isotopes of Pu, as well as other elements heavier than Pu known as higher actinides.

Appendix B SDT Model Influence Diagram

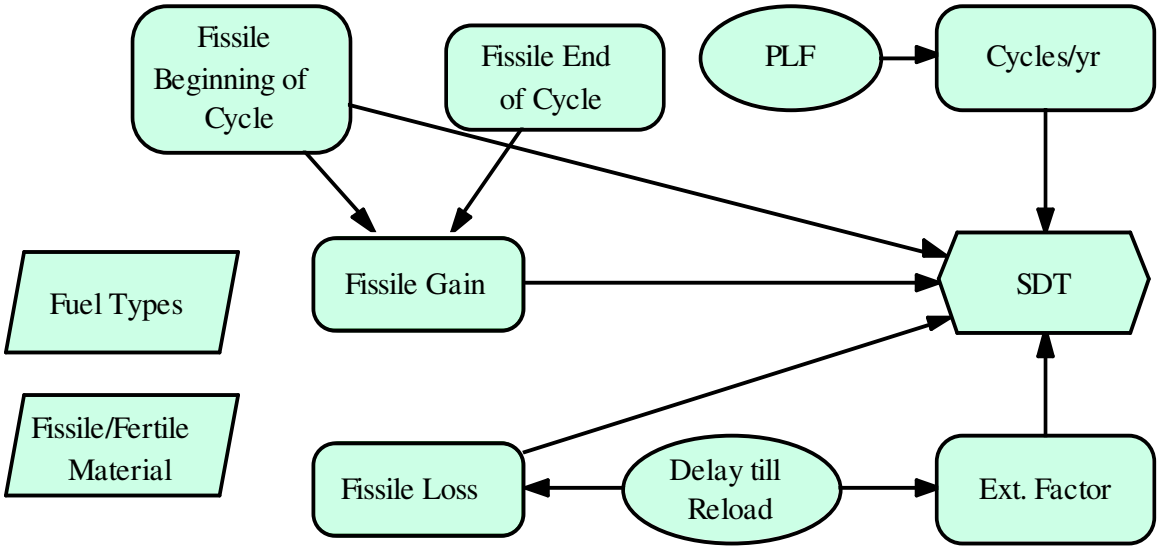

Figure 3: Influence Diagram for System Doubling Time Model

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.

Appendix C Worldwide Experience with Fast Reactors

Table 9: Worldwide Experience with Fast Reactors

Country	Plant	Power Output MWt/MWe	Characteristics	Status (Jan. 92)
France	Rapsodie	24 (40)/ -	MOX, loop-type	shutdown
	Phenix	563/254	MOX, pool-type	operating since 1974
	Superphenix	3000/1240	MOX, pool-type	operating since 1985
	SPX 2	/1500	MOX, pool-type	development on hold
DeBeNe	KNK II	60/21	MOX, loop-type	operating since 1974
	SNR 300	730/327	MOX, loop-type	Abandoned after construction on account of political decision
	SNR 2	/1380	MOX	development on hold
India	FBTR	42/15	Carbide, loop-type	operating since 1985, electrical generation expected in 1997
	PFBR	1200/500	MOX, pool-type	under development
Italy	PEC	123/	MOX, loop-type	abandoned due to political reasons
Japan	JOYO	100/	MOX, loop-type	
	MONJU	714/280	MOX, loop-type	under construction
USSR	BR 2	.1/	Pu-Metal, Hg coolant	dismantled
	BR5 (BR 10)	5 (10)/	multi-fuel, pool-type	in operation
	BOR 60	60/12	UO/MOX, pool-type	power operation since 1970
	BN 350	700/280	UO/MOX, pool-type + desalination	in operation
	BN 600	1470/600	UO fuel, pool-type	in operation
	BN 800	2100/800	MOX , pool-type	planned

Country	Plant	Power Output MWt/MWe	Characteristics	Status (Jan. 92)
UK	DFR	72/15	U-metal, loop-type	shut down
	PFR	600/270	MOX, pool-type	expected shut-down Apr. 94
	CDFR	3300/1320	MOX, pool-type	development abandoned
USA	Clementine	0/	Pu-metal, Hg coolant	dismantled
	EBR I	1.2/.2	U/Pu Metal, NaK coolant	dismantled
	LAMPRE	1/	Molten Pu	dismantled
	EFFBR	200/66	U-metal, loop-type	decommissioned
	EBR II	62/20	metal fuel, pool-type	operating since 1961
	SEFOR	20/	MOX, loop-type safety test facility	decommissioned after completing mission
	FFTF	400/	MOX, loop-type	operated 1980-1992, shut down after completing mission
	CRBR	975/380	MOX, pool-type	construction stopped

Source: Annexure 1, *Beyond FBTR* (Paranjpe, 1992)

Appendix D Nuclear Reactors and Reactions

Table 10: Reactor Profiles as Used in the Study

		Oxide	Carbide-Lee	Metal	Carbide
U238-Pu Cycle	Blanket Material	dep. U	dep. U	dep. U	dep. U
	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-U233 Cycle	Blanket Material	Th	Th	Th	Th
	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

These are based on INFCE(1980) and Lee *et al,* (1990).

Appendix E Thermal Fuel Reprocessing Capacity

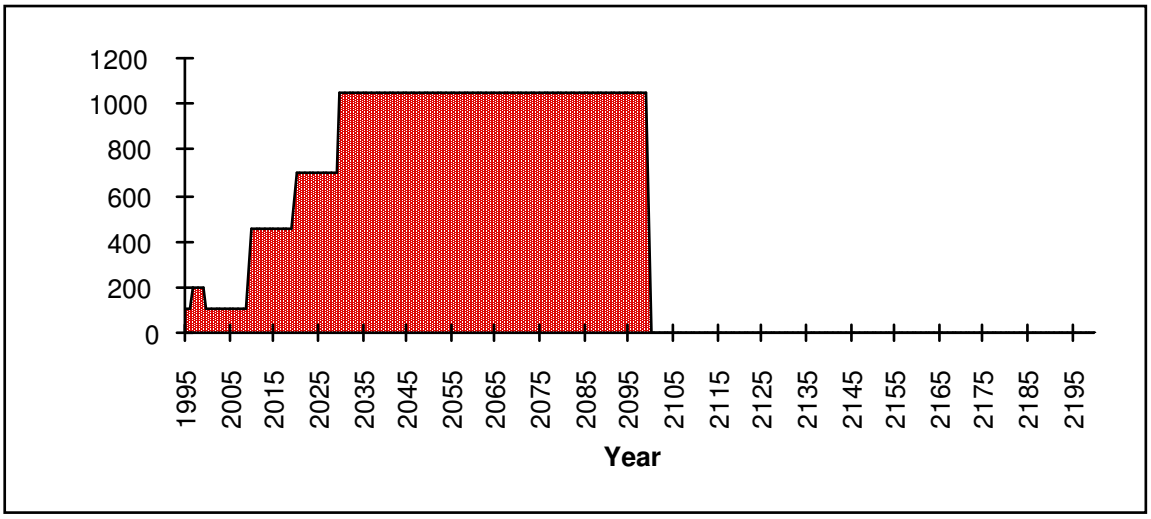

Figure 4: Projected PHWR Spent Fuel Reprocessing Capacity

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.

Appendix F Growth of Electrical Capacity

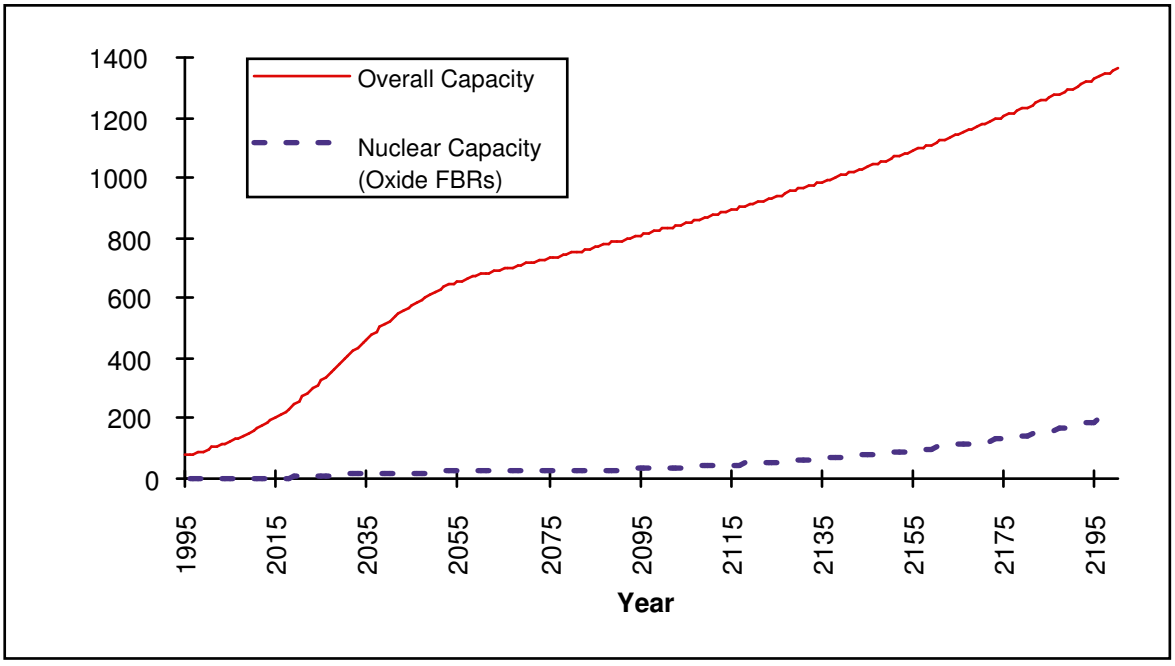

Figure 5a: Installed Electrical Capacity in India

The nuclear curve is based fissile availability using oxide FBRs with .6 PLF, 2% reprocessing losses, and a 3-year reprocessing delay. It also assumes 70,000 tons U available for PHWRs, and 4 teams constucting pairs of PHWRs.

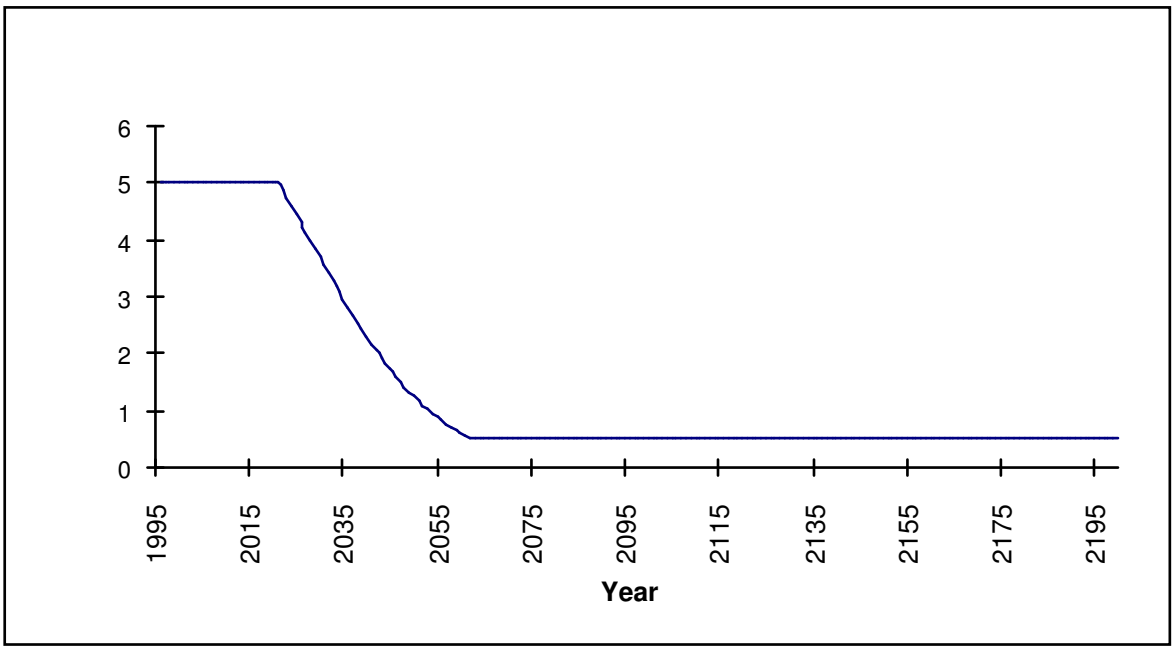

Figure 5b: Electricity Growth Rate in India

Both the above figures are based on a saturation population of 1.5 billion, overall PLF of .55, a current growth rate of 5% that will continue for 24 years, and a .5% residual growth rate. The growth rate decreases ≈linearly until saturation, after which the residual growth rate applies.

Appendix G Share of Nuclear Power—Varying PLFs

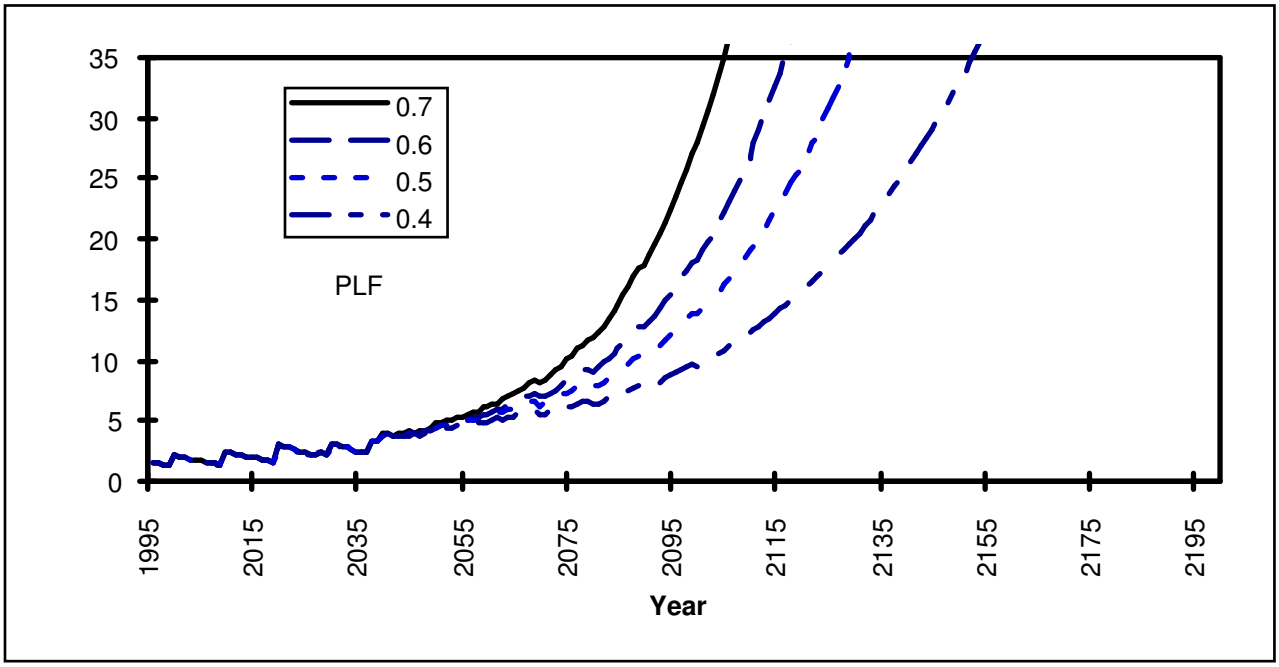

Figure 6a: Metallic Fuel, U238-Pu Cycle.

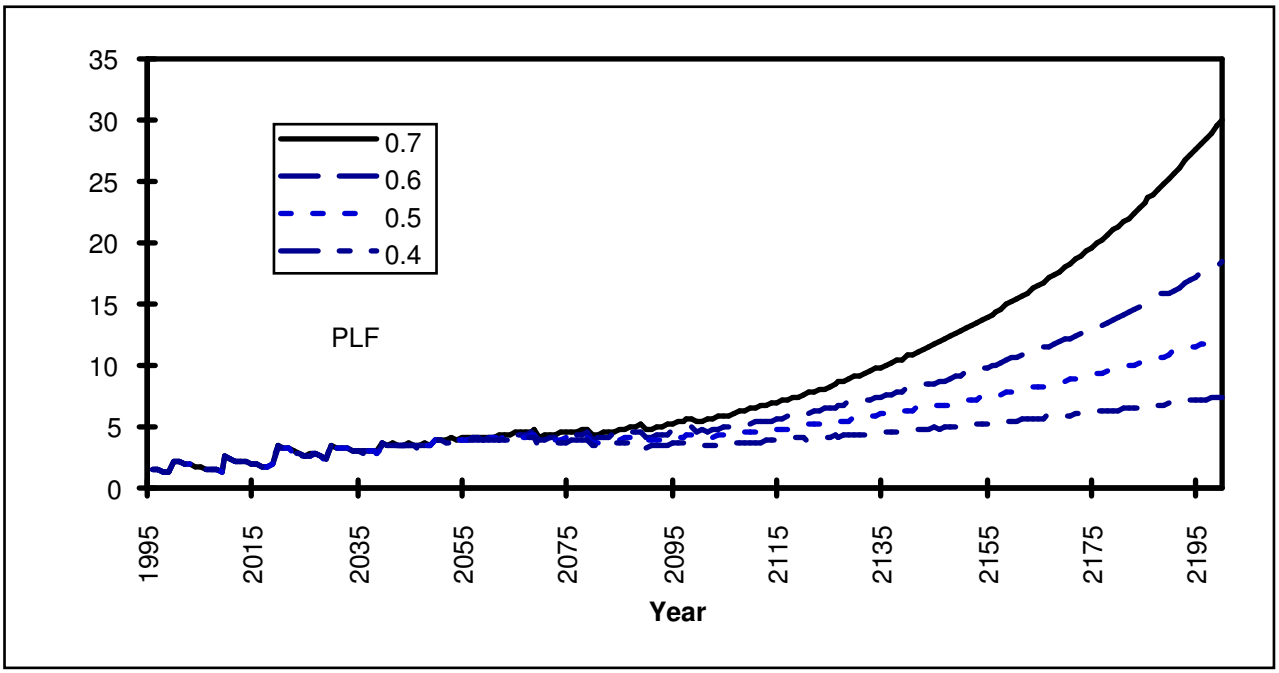

Figure 6b: Oxide Fuel, U238-Pu Cycle.

Figures 6a and 6b: 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%, 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.

Appendix H Radiological Toxicity

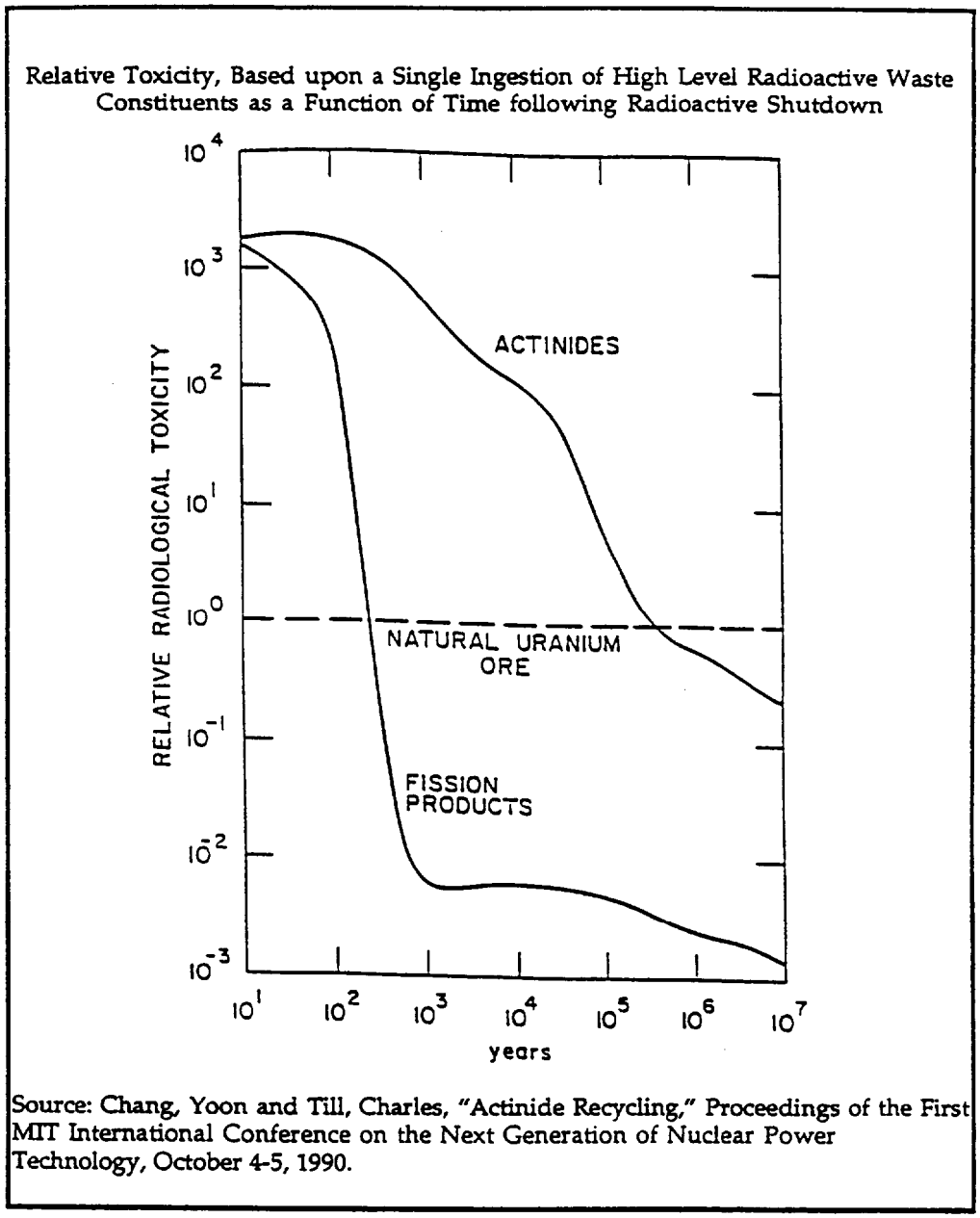

Figure 7: Benefits of Actinide Burning

This shows only the radiation reduction benefits from actinide burning (Oye *et al*, 1995).

Further Study

A comprehensive analysis must examine the role of Pu from a commercial point of view. What credit is being assigned (or implicitly computed) by DAE for Pu? Worldwide today, there is no credit for plutonium.

Reprocessing in general also needs further examination, especially in view of higher losses expected for various designs planned in India. There is limited experience reprocessing fast reactor fuel anywhere in the world. While the technical feasibility of wet reprocessing has been shown at Dounreay (Anderson, Frew *et al.*, 1994), this says nothing of likely batch sizes or throughputs.

The policy changes in US and India necessary for moving out of the status quo is a matter of great importance. The realizability of the four policy options presented in Table 7 is material for further study.