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Weapon-Grade Plutonium Production Potential in the Indian Prototype Fast Breeder Reactor

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India is building a 500 MWe Prototype Fast Breeder Reactor, which is scheduled to be operational by 2010. India has refused to accept international safeguards on this facility, raising concerns that the plutonium produced in its uranium blankets might be used to make nuclear weapons. Based on neutronics calculations for a detailed three-dimensional model of the reactor, we estimate that up to 140 kg of weapon-grade plutonium could be produced with this facility each year. This article shows how India's large stockpile of separated reactor-grade plutonium from its unsafeguarded spent heavy-water reactor fuel could serve as makeup fuel to allow such diversion of the weapon-grade plutonium from the blankets of the fast breeder reactor. We describe and assess the most plausible refueling strategies for producing weapon-grade plutonium in this way.

BACKGROUND

On July 18, 2005, U.S. President George W. Bush and Indian Prime Minister Manmohan Singh issued a joint statement, laying the grounds for the resumption of full U.S. and international nuclear aid to India. Under this agreement, the United States has amended its own laws and policies on nuclear technology transfer and is seeking changes in international controls on the supply of nuclear fuel and technology so as to allow "full civil nuclear energy cooperation

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Disclaimer: This work was carried out in compliance with the MCNP/MCNPX software single user software license and relevant U.S. Department of Commerce regulations.
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and trade with India."[1] In exchange, India's government has identified a list of civilian nuclear facilities. These will be separated from its nuclear weapons complex and offered for safeguards by the International Atomic Energy Agency (IAEA). One contentious issue in this agreement and the related civil-military separation of nuclear facilities is the status of the fast breeder reactor program.

The Indian Department of Atomic Energy (DAE) declared that sites and facilities related to the breeder program would not be put under safeguards.[2] This includes the entire Kalpakkam site near Madras, where the existing 40 MW_{th} Fast Breeder Test Reactor (FBTR) and the upcoming 500 MW_e Prototype Fast Breeder Reactor (PFBR) are located. The site also features a reprocessing plant and two operational heavy-water reactors. The construction of a second reprocessing plant at the site is planned. Existing stockpiles of separated plutonium or spent fuel from some heavy water reactors would equally remain outside safeguards.

To explain this position, the head of the DAE said in an interview to a leading Indian newspaper: "Both, from the point of view of maintaining long-term energy security and for maintaining the minimum credible deterrent, the fast breeder programme just cannot be put on the civilian list."[3] All of this suggests that India's nuclear establishment envisions the use of plutonium generated in the breeder reactor to make weapons.

The present work is an attempt to understand what contribution the PFBR could make to India's stockpile of weapon-grade plutonium. Current estimates of the stockpile are around 500 kg, sufficient for about 100 weapons, with production continuing at about 30 kg/year.[4] As will be seen, the PFBR is capable of producing about 140 kg of weapon-grade plutonium annually. Thus, it would allow a nearly five-fold increase in the rate of production of weapon-grade plutonium in India.

We start with a brief overview of the Indian fast breeder program, followed by a discussion of the design of the PFBR and a "back of the envelope" estimate of its plutonium production. Computer simulations using a detailed model of the reactor are the central part of this article. The methodology of the calculations and the details of the model are presented. We then discuss the main results of the calculations, in particular the equilibrium-core conditions and associated annual fuel flows. We identify dedicated options to produce weapon-grade plutonium, estimate the amounts that can be generated with these strategies, and also show how the growing stockpile of unsafeguarded reactor-grade plutonium suffices as makeup fuel to allow this diversion.

OVERVIEW OF THE INDIAN BREEDER PROGRAM

Breeder reactors were originally embraced by the Indian nuclear establishment as part of a three-stage nuclear program proposed in the 1950s as a way to expand nuclear power using the domestic resources of uranium ore, which are both limited and of poor quality.[5] The first stage of this strategy involved the use of uranium fuel in heavy-water reactors, followed by reprocessing of the irradiated spent fuel to extract plutonium. In the second stage, the accumulated plutonium is to be used in turn as fuel in the cores of fast breeder reactors. These cores could be surrounded by a blanket of either (depleted) uranium or thorium to produce additional plutonium or uranium-233, respectively. So as to ensure that there is adequate plutonium to fuel these second-stage breeder reactors, a large fleet of breeder reactors with uranium blankets would have to be commissioned before thorium blankets are introduced. The third stage involves breeder reactors using uranium-233 in their cores and thorium in their blankets.[6]

The DAE started work on the PFBR more than twenty years ago. After several delays, construction of the reactor started in October 2004, and the facility is now expected to be commissioned in 2010. The PFBR is to be the first of the many breeder reactors that the DAE envisions building.[7]

MAIN CHARACTERISTICS OF THE PROTOTYPE FAST BREEDER REACTOR

The design of the PFBR has been evolving over more than two decades and hence different pieces of information, sometimes at odds with each other, can be found in the literature published by the DAE or by scientists and engineers working on the project.[8]

The PFBR will have a thermal power of 1,250 MW (500 MW_e) and use mixed-oxide fuel (MOX, PuO_2/UO_2) in the core and depleted UO_2 in the radial and axial blanket regions. The reactor design involves a homogeneous core with two enrichment zones.[9] The inner zone of the core consists of 85 fuel assemblies with a plutonium fraction of 21%. The outer zone has 96 assemblies with an increased plutonium fraction of 28% to flatten the power distribution in the core. Each core fuel assembly contains 217 pins with an outer diameter of 6.60 mm and a fueled-length of 100 cm. The pins also have axial blanket sections extending above and below the fueled region (2×30 cm). The radial blanket of the reactor consists of 120 assemblies, which surround the core in two rows. Each of these assemblies contains 61 pins with an outer diameter of 14.33 mm and a fueled-length of 160 cm. All these dimensions are summarized in Tables 4 and 5 at the end of this article.

The radial blanket is surrounded by a thick steel neutron reflector. Beyond the reflector, there are several rows of shielding and storage positions. The reactor is equipped with a total of twelve control and safety rods. Figure 1 illustrates the general reactor configuration as modeled in MCNP as well as the geometry of the core and blanket fuel assemblies.

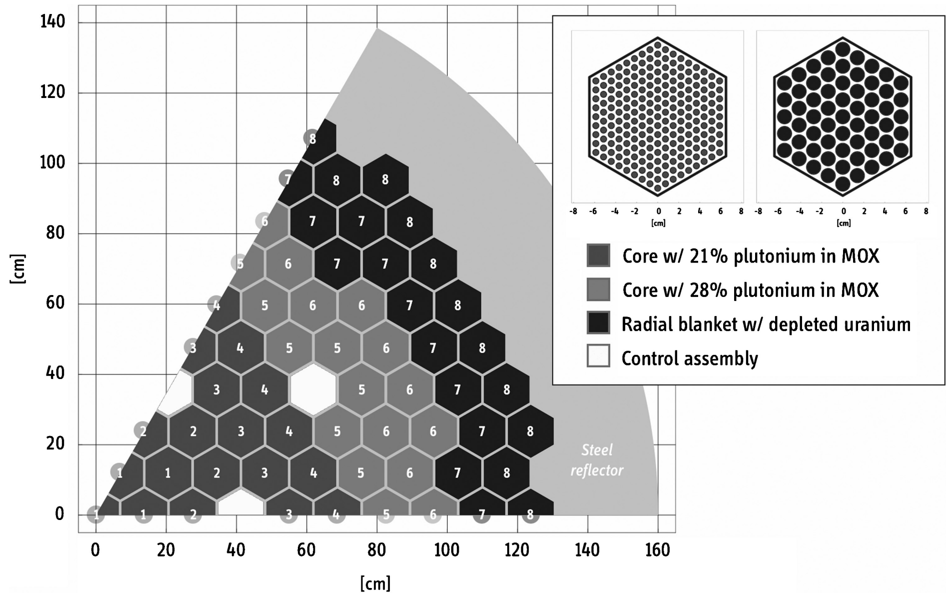

Figure 1: View of the MCNP model of the PFBR reactor core. A three-dimensional sixty-degree segment is modeled with appropriate reflecting surfaces. Shielding and storage positions beyond the steel reflector are not modeled in MCNP. The numbers in the fuel assemblies (1–8) designate a subset of the burnup zones. The fuel pin configuration within the assemblies (217 and 61 pins in the core and radial blanket, respectively) are illustrated in the inset. The remaining six burnup zones (9–14) are in the axial blanket of the core and are not depicted in this figure.

APPROXIMATE ANALYTICAL ESTIMATE

To get a first-order estimate of the magnitude of the net plutonium production by the PFBR, the detailed analysis was preceded by a simple estimate based on the neutron balance in a generic breeder reactor with a homogeneous core.[10] Main input data used for this estimate are:

Power distribution. On average, at the middle of an equilibrium-cycle, about 90% of the total power is generated in the core, whereas the remainder is generated in the axial and radial blankets.[11]

Consumption of fissile material. Assuming a capacity factor of 75% and an energy release of 200 MeV per fission event, the amount of material fissioned to generate 1,250 MW_{th} for one year can be estimated.

$$\frac{0.239\ \text{kg}}{6.022\ 10^{23}}\ \frac{0.75\,(1250\ 10^{6})\ \text{W}\,(365\ 24\ 3600)\ \text{s}}{200\ 10^{6}\ 1.6022\ 10^{-19}\ \text{J}} = 366\ \text{kg}$$

Not all fissions in the fuel occur in fissile materials. For the given fuel composition and for a typical fast neutron spectrum, only about 80% of the fission processes can be expected to take place in Pu-239, Pu-241, and U-235. The non-fissile isotopes U-238 and Pu-240 contribute the remainder.[12] At the

Table 1: First estimate of the net fissile material production in the blankets of the PFBR. Estimate based on typical power distribution between core and blankets, annual consumption of fissile material in the reactor, and breeding ratios for the overall reactor (1.05) and for the core only, where values between 0.60 and 0.70 are assumed. See text for further details.

	Reactor		**Core (only)**	
Thermal power	1250 MW		1125 MW	
Fissile material consumption	366 kg/yr		330 kg/yr	
Breeding/conversion ratio	1.05	0.60	0.65	0.70
Fissile material production	385 kg/yr	198 kg/yr	215 kg/yr	231 kg/yr
Net fissile material production	19 kg/yr	−132 kg/yr	−115 kg/yr	−99 kg/yr
Net fissile material production in blankets (inferred)				
		151 kg/yr	134 kg/yr	118 kg/yr

same time, neutron capture in fissile isotopes—converting for example Pu-239 into Pu-240—increases overall fissile material consumption in the fuel. Typical capture-to-fission ratios are in the order of 0.25, which essentially counterbalances the first effect. In the following, it is therefore assumed that there is a fissile material consumption of 366 kg per year.

Breeding ratio (overall and core). The overall breeding ratio of the PFBR is reported to be 1.05, and this value is used for for this estimate.[13] Published core breeding (or conversion) ratios for liquid-metal-cooled reactors are in the range of 0.6–0.9.[14] The PFBR core design has a reduced core conversion ratio. Here, illustrative values between 0.6 and 0.7 are used.

The results obtained with these assumptions are summarized in Table 1. For the range of core breeding ratios considered here, the PFBR would produce between about 120 kg and 150 kg of weapon-grade plutonium in the axial and radial blankets. As will be shown, the results of the detailed calculations presented in what follows predict an annual weapon-grade production rate in the upper half of this range.

METHODOLOGY

Based on the main reactor design parameters and using further operational characteristics, such as the proposed refueling frequency and pattern, a modeling strategy and detailed reactor model have been developed. These are discussed in what follows, along with the computer codes used.

Neutronics Codes and Reactor Model

Figure 2 illustrates the functional relationship of the individual codes that constitute the computational system (M^3O). The system has been developed and described in Glaser (2005) and is based on extensively validated computer codes.[15]

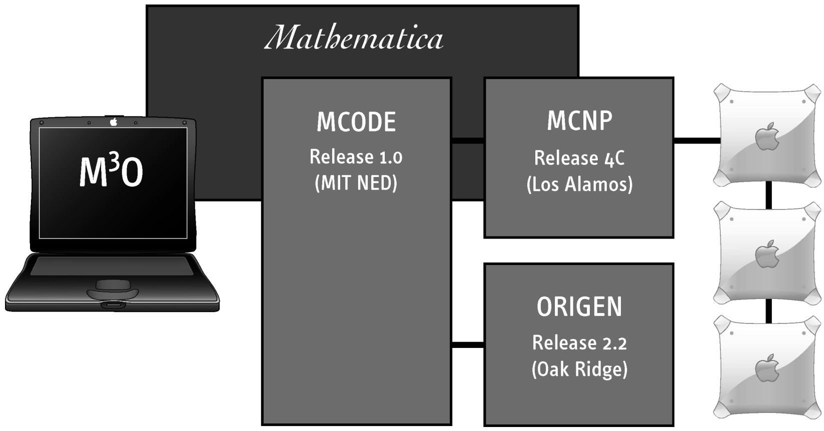

Figure 2: Computational system for neutronics calculations.

At the most fundamental level of the computational structure, the Monte Carlo particle transport code MCNP and the general point-depletion code ORIGEN2 perform the actual neutronics calculations.[16] Communication between both programs is established and coordinated by the linkage program MCODE.[17] Most importantly, MCODE regularly updates the base MCNP input deck of the reactor model to enable determination of required flux and power distributions and to enable calculation of spectrum-averaged one-group cross-section data for the most important nuclides.[18] These data are then passed to ORIGEN2 for the next burnup step.

The general setup and geometry of the three-dimensional MCNP computer model used for all calculations is shown in Figure 1. The sixty-degree segment of the reactor with appropriate reflecting surfaces captures the symmetry of the configuration. In the radial direction, the model extends to a radius of 160 cm and includes the steel reflector with a thickness of about 30 cm. Shielding and storage positions beyond the reflector are *de facto* irrelevant for the neutronics of the reactor and are not modeled in the MCNP simulations. A 100 cm axial column of a steel-sodium pseudo-reflector resides above and, by reflection, below the core.

The total density of the MOX-fuel is set to 10.45 g/cc, which corresponds to about 92% of the theoretical density of the material at 300 K. Using the pin geometry listed in Table 4, this density yields an initial inventory of 9.4 kg (21% loading) and 12.5 kg (28% loading) of plutonium per fuel assembly. The total plutonium inventory in the core amounts to about 1,900 kg at the middle of an equilibrium-cycle, which is close to the figure of 1,978 kg listed as the total plutonium content of the core in the IAEA Fast Reactor Database.[19] The total fuel inventory (plutonium and uranium oxide) works out to be about 9.15 tons, consistent with official design figures.[20]

Control and safety rod movements are not simulated in the calculations. In order to take into account the net impact of these B_4C-rods on the flux and power distributions, a constant concentration of 0.25 g/cc of natural boron is established in the respective rod positions. The presence of this neutron absorber results in a reactivity decrease of the reactor configuration of $\Delta\rho \approx -0.05$ throughout the cycle and yields a cycle-averaged total reactivity of the core close to zero.

Modeling Strategy

For the burnup calculations, a set of 14 burnup zones has been defined: four zones for the inner core (21% PuO_2), two zones for the outer core (28% PuO_2), four zones for the radial blanket, and four zones for the axial blanket.[21] Average inventories and isotopics for each of these zones are generated during irradiation of fuel and blanket.

Refueling of the reactor is planned every 180 effective-full-power days (EFPDs).[22] Our simulations are based on a simplified refueling plan for the reactor that is designed to reproduce average core conditions. We assume that one third of the core and one eighth of the radial blanket are exchanged during each reload. This corresponds to a mean irradiation time of 540 EFPDs for a core assembly and of 1,440 EFPDs for a radial blanket assembly. Calculations for the core and for the radial blanket are performed separately in order to guarantee that the core is exposed to average blanket conditions and vice versa. This approach is shown in Figure 3.[23]

A different or more detailed refueling scheme could slightly affect the average burnup of the fuel and the blanket. As shall be seen, however, plutonium depletion in the core and production in the blankets are virtually linear functions of burnup (Figure 3). Thus, if the average burnup of the fuel was, for example, lower than assumed in the simulations, then the plutonium content per element would decrease correspondingly, but be compensated by an increased rate of fuel discharge and reprocessing. Changes in the refueling scheme therefore would only have a small impact on the net production of weapon-grade plutonium per year.

RESULTS

The primary objective of the simulations is to obtain inventory data under equilibrium conditions for *average* fuel assemblies of the core and the axial and radial blankets as a function of burnup. Based on these results, annual plutonium production, breeding ratio, and other quantities of interest can be calculated.

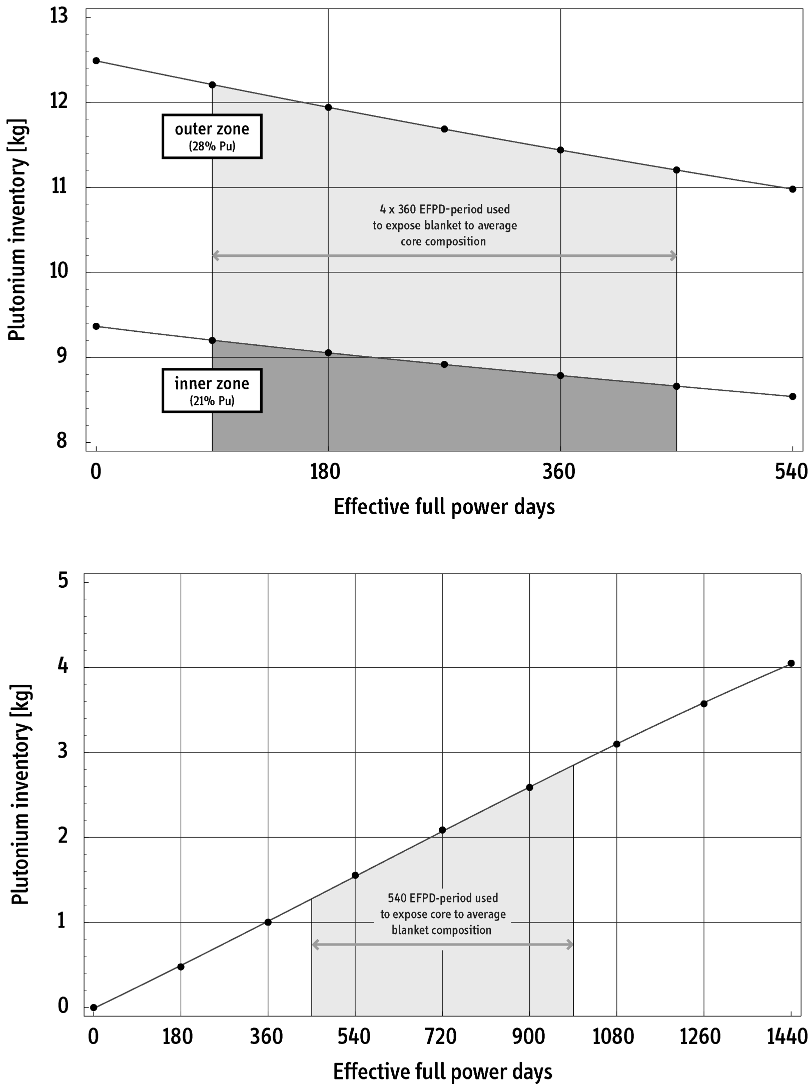

Figure 3: Plutonium inventory in an average fuel assembly of the core (top) and of the radial blanket (bottom). There are 85, 96, and 120 fuel assemblies in the inner core, the outer core, and the radial blanket. Plutonium buildup in the axial blanket is not depicted.

Determination of Equilibrium Core Conditions

In order to determine the equilibrium core conditions for the reactor, several candidate plutonium compositions with varying fissile fractions were explored. These initial calculations provide a first rough approximation for the equilibrium plutonium vector. Using these preliminary isotopics, and after several iterations (i.e., full reactor simulations), a plutonium vector can be identified that satisfies the requirements for equilibrium: the plutonium composition of the fresh fuel is virtually identical to the vector of the discharged fuel if the plutonium recovered from the core *and* from axial and radial blankets is combined in the reprocessing stage and reused to fabricate the fuel for the next core. The main results of the simulations for the PFBR under equilibrium-core conditions are listed in Table 2.

According to these results, the equilibrium fissile fraction of the plutonium in the PFBR is 69.2% (65.2% Pu-239 and 4.0% Pu-241). Simultaneously, the equilibrium composition of the uranium is determined. Here, we assume that the initial uranium used to produce the MOX fuel for the core and the blanket is recovered from irradiated pressurized-heavy-water-reactor-type (PHWR-type) fuel and therefore already depleted in U-235. Respective uranium fractions used for initial calculations are 0.25% U-235, 0.07% U-236, and 99.68% U-238, which corresponds to an average PHWR-type fuel burnup of about 6.7 MWd/kg (see Appendix for details). However, the uranium is quickly further depleted as it is recycled indefinitely and used to fabricate MOX for the core and UO_2 for the blankets.

Due to conversion to plutonium and fast fission, about 5% of the uranium is consumed during irradiation (382 kg per year). In addition, about 2% of the total inventory is typically lost in the reprocessing, conversion, and fabrication stages.[24] Based on the annual uranium discharge, this corresponds to 160 kg of uranium. In total, more than 540 kg of uranium are needed as annual makeup. If the reference material is used for this purpose (0.25% U-235), this admixing weakly counteracts the depletion process. Still, the uranium in the PFBR will ultimately be deeply depleted in U-235 with a fissile fraction of not more than 0.07% (see Table 2).

Reload and Discharge Analysis for the Equilibrium Core

The plutonium inventory fuel assembly during irradiation is shown in Figure 3. In the core, 10–12% of the initial plutonium content is consumed before the end-of-life of an average fuel assembly is reached after 540 effective full power days. At the same time, 0.57 kg of plutonium is produced in the axial blanket sections of a core fuel assembly (not shown). In addition, 4.05 kg of plutonium is generated in an average assembly of the radial blanket, when it reaches its end-of-life after an irradiation period of 1,440 EFPDs.

In the context of this article, the plutonium isotopics in the blanket regions of the reactor are of particular interest and they are shown in Figure 4. As is characteristic for a fast neutron reactor, weapon-grade plutonium is recovered from both radial and axial blankets with respective Pu-239-fractions of 93.7% and 96.5%. This contrasts with typical plutonium compositions in thermal reactors, where Pu-240-buildup is much more pronounced once the Pu-239-content in the fuel is non-negligible.[25]

Table 2: Main results obtained in M^3O neutronics calculations for the Prototype Fast Breeder Reactor operated under equilibrium-core conditions. For the uranium needed for the annual makeup, it is assumed that 2% of the discharged uranium is lost in the various processing stages and that the uranium-235 fraction in the makeup is 0.25%.

					Annual Reload			
	U-235	**U-236**	**U-238**	**Pu-238**	**Pu-239**	**Pu-240**	**Pu-241**	**Pu-242**
Core	2.6 kg	3.1 kg	3075.9 kg	1.8 kg	659.5 kg	283.7 kg	41.8 kg	24.7 kg
Axial blanket	2.0 kg	2.5 kg	2450.9 kg	0.0 kg	0.0 kg	0.0 kg	0.0 kg	0.0 kg
Radial blanket	2.2 kg	2.7 kg	2617.8 kg	0.0 kg	0.0 kg	0.0 kg	0.0 kg	0.0 kg
Total	6.8 kg	8.3 kg	8144.6 kg	1.8 kg	659.5 kg	283.7 kg	41.8 kg	24.7 kg
Overall total		8160 kg of uranium				1012 kg of plutonium		
Fissile fraction		0.08%				69.3%		
					Annual discharge			
	U-235	**U-236**	**U-238**	**Pu-238**	**Pu-239**	**Pu-240**	**Pu-241**	**Pu-242**
Core	1.5 kg	3.1 kg	2870.1 kg	1.6 kg	545.5 kg	288.4 kg	42.2 kg	24.8 kg
Axial blanket	1.6 kg	2.5 kg	2391.7 kg	0.017 kg	50.15 kg	1.78 kg	0.056 kg	0.001 kg
Radial blanket	1.5 kg	2.6 kg	2502.8 kg	0.066 kg	86.33 kg	5.71 kg	0.262 kg	0.008 kg
Total	4.6 kg	8.2 kg	7764.6 kg	1.6 kg	682.0 kg	295.9 kg	42.5 kg	24.8 kg
Overall total		7777 kg of uranium				1047 kg of plutonium		
Fissile fraction		0.06%				69.2%		
Blanket subtotals	52.0 kg of plutonium with a fissile fraction of 96.5% contained in the axial blanket 92.4 kg of plutonium with a fissile fraction of 93.7% contained in the radial blanket 144.4 kg of plutonium with a fissile fraction of 94.8% contained in both blankets combined							
					Annual Makeup			
	U-235	**U-236**	**U-238**	**Pu-238**	**Pu-239**	**Pu-240**	**Pu-241**	**Pu-242**
	1.4 kg	0.4 kg	540.2 kg					
Total		382 kg + 160 kg						
Fissile fraction		0.25%						
Makeup + Discharge	6.0 kg	9.0 kg	8305.0 kg					
Total		8320 kg						
Fissile fraction		0.07%						

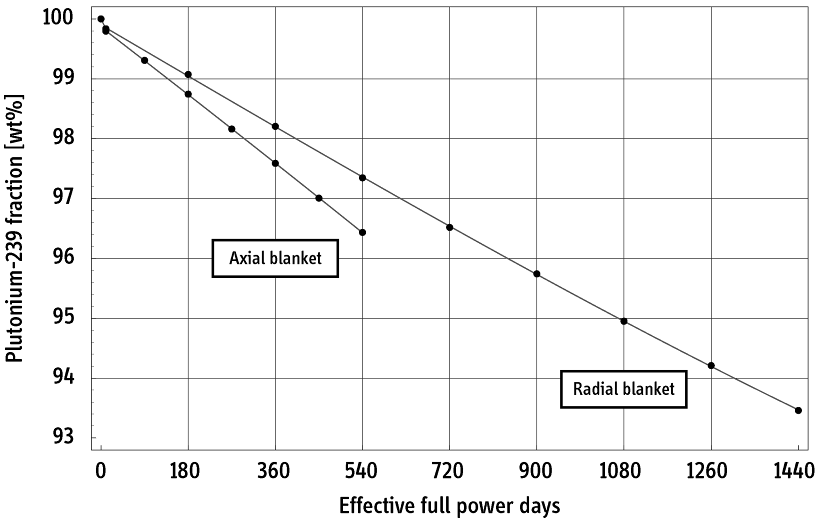

Figure 4: Plutonium isotopics in axial and radial blankets.

The inventories per fuel assembly can be used and rescaled to obtain data for a reload-discharge analysis under equilibrium conditions. Table 2 lists the mass balances for all relevant uranium and plutonium isotopes normalized per calendar year. Here, we assume a capacity factor of 75%, which is equivalent to about 274 EFPDs per year.[26] As mentioned, reloading is planned once every 180 EFPDs, when one third of the core and one eighth of the radial blanket are exchanged. On average, there are therefore 1.52 reloads per calendar year and a total of 43.1 fuel assemblies from the 21%-core, 48.6 assemblies from the 28%-core, and 22.8 assemblies from the radial blanket are unloaded in this period.

Breeding Ratio

As shown in Table 2, 1,012 kg of plutonium are loaded to the reactor each year. In the same period, 1,047 kg of plutonium are discharged: 903 kg from the core and 144 kg from the blankets. A more detailed discussion of plutonium production follows, but the overall data can be used to determine the breeding ratio of the reactor. The following definition of the breeding ratio is used:[27]

$$BR = 1 + \frac{M_{\mathrm{DISC}} - M_{\mathrm{LOAD}}}{M_{\mathrm{DEST}}}$$

All numbers are for the same reference period, for example per calendar year, and correspond to the fissile material discharged (M_{DISC}), the fissile material loaded (M_{LOAD}), and the fissile material destroyed (M_{DEST}). The values for M_{DISC}

and M_{LOAD} can be taken directly from Table 2. Fissile materials are Pu-239, Pu-241, and U-235. The amount of fissile material destroyed M_{DEST} cannot be directly obtained from the results obtained in the simulation. It includes the fissile material that is consumed by neutron absorption, leading to fission or capture, during the cycle and requires an isotope-by-isotope analysis of the spectrum-averaged cross-sections. Here, the same value that was used for the analytical estimate earlier (M_{DEST} = 366 kg) is chosen. The breeding ratio for the PFBR can be determined to:

$$BR = 1 + \frac{729\ \text{kg} - 708\ \text{kg}}{366\ \text{kg}} \approx 1.057$$

This value is in accordance with the predictions of the reactor designers.[28] Equivalently, net fissile material production of 21 kg per year is close to the value found in the simple analytical estimate from earlier.

Refueling Options for the PFBR and Weapon-Grade Plutonium Production Potential

In a fast neutron reactor operated in a "civilian" mode, there is no need or incentive to reprocess the plutonium in the blanket separately from the plutonium in the core.[29] In principle, the entire stock of spent fuel discharged from the reactor can be processed together yielding a blended-average plutonium composition. If operated in a "military" mode, however, there are two main options.

Separate processing of radial blanket. Reprocessing of the radial blanket in a separate campaign is straightforward and does not require any special provisions or equipment. As listed in Table 2, 92 kg of weapon-grade plutonium per year can be obtained with this strategy.

Separate processing of radial and axial blanket. This approach requires chopping of the core fuel assemblies in order to isolate the top and bottom sections containing the axial blanket material. Such a strategy could require dedicated equipment and procedures in the reprocessing stage,[30] but it would yield an additional 52 kg of weapon-grade plutonium per year. For weapon-use, the isotopics of this material would be even superior to those of the radial blanket (96.5% versus 93.7% Pu-239). Total annual weapon-grade plutonium production increases to 144 kg.

If the reactor is operated in a military mode, in which some or all of the plutonium from the blankets is diverted, refueling options are preferable that do not disturb the equilibrium conditions of the core. Also, the PFBR will no longer be self-sustaining, if more than 35 kg of plutonium are diverted, without even taking into account inevitable processing losses.[31]

The plutonium for the initial core and reloads of the PFBR comes from reprocessing spent fuel from pressurized heavy-water reactors, which form the mainstay of the current reactor fleet in India. It is likely that the reprocessing will be carried out at the Kalpakkam Reprocessing Plant (KARP), next to the PFBR-construction site.[32] Estimates of the isotopics of this plutonium are presented in the Appendix. For a typical burnup of 6.7 MWd/kg and a cooling period of 5 years, the fissile fraction of this material is about 77.1%. As the preceding discussion has shown and as illustrated in Figure 5, the composition of the PHWR-type plutonium is entirely different from the vector that establishes itself in the PFBR under equilibrium conditions. In other words, the reactor will have to go through an initial phase, in which it is operated off equilibrium. Depending on the details of operation, refueling and reloading patterns in particular, non-equilibrium conditions will persist for several reloads and years, during which the momentary breeding ratio will generally be lower than one. A more detailed analysis of this transitory phase is beyond the scope of this article.

As of May 2006, the DAE's PHWRs are estimated to have produced about 11.5 tons of such reactor-grade plutonium. It is likely that a major fraction of this plutonium has already been separated, but the entire stockpile will remain outside of safeguards.[33] There is thus a sufficient quantity of plutonium available for both the initial loading and the first few reloads of the reactor. Similarly, reactor-grade plutonium from the DAE's PHWRs could also be used

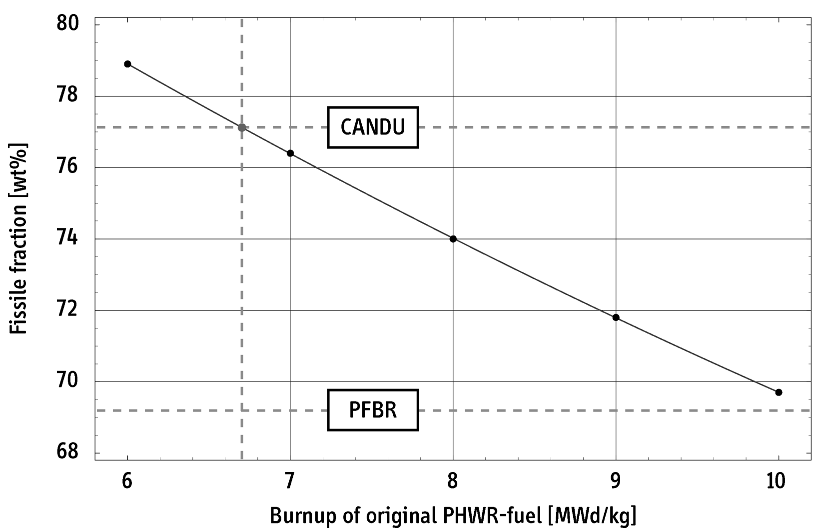

Figure 5: Fissile fraction of cooled PHWR-type (CANDU) plutonium as a function of original burnup. For a reference burnup of 6.7 MWd/kg, the fissile fraction (Pu-239 + Pu-241) is about 77.1%, if corrected for a five-year cooling time between discharge and reprocessing. In contrast, in equilibrium, the PFBR plutonium contains only about 69.2% of the fissile isotopes Pu-239 and Pu-241. See Appendix for further details.

Table 3: Refueling options for the PFBR.

	Core 902.6 kg 65.1% Pu-FIS	Axial blanket 52.0 kg 96.5% Pu-FIS	Radial blanket 92.4 kg 93.7% Pu-FIS	PHWR (unlimited supply) 77.1% Pu-FIS
Option 1	1012 kg from the core and both blankets (plus 35 kg surplus)			
Option 2	776.2 kg (plus 178.4 kg surplus)		(not reused)	235.8 kg
Option 3	666.2 kg (plus 236.4 kg surplus)	(not reused)	(not reused)	345.8 kg

1,012 kg of plutonium are needed annually to fabricate fuel for the reactor. Slightly more than this amount can be extracted from the discharged fuel (Option 1). Alternatively, existing PHWR-type plutonium can be used as makeup if weapon-grade plutonium from the blanket is diverted (Options 2 and 3). With the correct mixing ratios, the fuel is neutronically equivalent to the original fuel and characterized by a fissile fraction in plutonium of 69.2%.

to make up for the weapon-grade plutonium that might be diverted for military purposes, if the PFBR is operated in military mode. Because the fissile fraction of the PHWR-type plutonium is higher than the fissile fraction needed for operation of the PFBR in equilibrium, practical blending strategies are feasible and straightforward. These options are shown in Table 3.

Option 1 in Table 3 is included for reference purposes and corresponds to operation of the reactor in civilian mode, that is, without diversion of any plutonium from the blankets.

As discussed earlier, the most straightforward military option is to divert the material from the radial blanket (Option 2). Table 3 shows the mixing ratio that yields the needed amount of plutonium for refueling with the desired composition. About 236 kg of the reference PHWR-type plutonium are needed to divert the plutonium from the radial blanket entirely (92 kg). In addition, a surplus of PFBR plutonium remains: about 178 kg instead of only 35 kg, if processing losses are neglected. The general procedure for Option 3 is similar to the one for Option 2. Here, about 346 kg of PHWR-type plutonium are needed to divert 144 kg of weapon-grade plutonium from the blankets, while keeping an additional surplus of 236 kg of plutonium from the PFBR-core.

In essence, Options 2 and 3 are plutonium "purification" schemes that convert one pre-existing stock of PHWR-type plutonium into two separate stocks of about the same combined size: one weapon-grade and one reactor-grade with poorer isotopics than the original one.[34] The left over reactor-grade plutonium could be used as initial fuel for a future breeder reactor, which would then achieve equilibrium somewhat more quickly because the isotopics match the equilibrium-value more closely.

In addition to the pure diversion options defined in Table 3, in which the entire plutonium inventory from the blankets is diverted at once, "mixed" strategies are also feasible. Here, using adequate mixing ratios, only a fraction of the blanket material would be used for military purposes. Obviously, mixed strategies reduce the requirements for externally supplied PHWR-type plutonium.

It is worth noting that reactor operation is unaffected by a decision to divert or not to divert blanket plutonium. If the correct fissile fractions are adjusted in the fuel fabrication stage, the recycled fuel delivered to the reactor is neutronically identical for both modes of operation and for all diversion options outlined earlier.

Plutonium Production Potential Using Future Breeder Reactors

The DAE plans to construct four more 500 MWe fast breeder reactors with the same MOX-fueled design as the PFBR by 2020.[35] After that, it plans to construct only fast breeder reactors using metallic fuel—and a different, as yet unspecified, core design. Thus far, there are no indications that any of these breeder reactors would be safeguarded.

As part of the Indo-U.S. agreement, the Indian government has announced a phased plan for putting a fraction of its current fleet of PHWRs under safeguards. Accordingly, six additional 220 MW_e reactors will be offered for safeguards between 2010 and 2014.[36] Until then, these reactors will produce an additional 4,300 kg of reactor-grade plutonium, assuming they operate at 80% capacity factor. Those PHWRs that will be retained in the military part of the nuclear complex will produce about 1,250 kilograms of unsafeguarded reactor-grade plutonium per year.

Thus, the annual production of unsafeguarded reactor-grade plutonium would be sufficient for indefinitely operating all five MOX-fueled breeder reactors, while diverting the radial blankets for weapon-purposes (Option 2). However, assuming no more unsafeguarded PHWRs are built, annual supplies of reactor-grade plutonium would be insufficient to sustain these five breeder reactors under Option 3. Yet, if one includes the accumulated stockpile, there would be sufficient reactor-grade plutonium to meet the makeup requirements of operating five breeder reactors under Option 3 until about 2050.[37] With five such reactors operating outside of safeguards, the combined weapon-grade plutonium production based on India's breeder program could therefore reach 500–700 kg/year after 2020.

CONCLUSION

The Indian Department of Atomic Energy has stated that it might use the Prototype Fast Breeder Reactor for military purposes. In this article, we have shown how the breeder program could allow for a several-fold increase in India's annual weapon-grade plutonium production.

Assuming a capacity factor of 75%, the results of the neutronics calculations for a detailed three-dimensional model of the core predict that about 140 kg of weapon-grade plutonium will be produced in the blankets of this reactor each year. Annual plutonium production would scale with the capacity factor and would also depend on the details of the refueling pattern. With respect to the latter, however, the results of the simple analytical estimate, and the fact that plutonium buildup in the blankets largely remains in the linear regime, suggest that deviations from the predicted value would be small.

If the reactor is operated in a military mode, and blanket material is diverted for weapon-purposes, then about 240–250 kg of reactor-grade plutonium from PHWR spent fuel would be required as makeup breeder fuel for every 100 kg of weapon-grade plutonium diverted. India could easily meet this demand for plutonium from either its existing stock of unsafeguarded PHWR spent fuel or from ongoing spent fuel discharges from its unsafeguarded PHWR reactors.

Based on the estimated existing reactor-grade plutonium stockpile and production rates, and assuming that India successfully implements its plan to build and operate a total of five fast breeder reactors by 2020, weapon-grade plutonium production rates could reach 700 kg per year. This would correspond to a twenty-fold increase in India's current weapon-grade plutonium production capacity. India could sustain this level of production for several decades without building additional heavy-water reactors.

Because India supports a verifiable fissile material cutoff treaty, a more farsighted option would be to put these reactors under safeguards now in order to prevent an accelerated arms race in the region, which appears almost inevitable otherwise.

Table 4: PFBR fuel pin and fuel assembly design data.

	Core and axial blanket	Radial blanket
Pellet diameter:	5.330 mm	12.760 mm
Gap thickness:	0.185 mm	0.185 mm
Cladding thickness:	0.450 mm	0.600 mm
Outer diameter of fuel pin:	6.600 mm	14.330 mm
Fuel pins per assembly:	217	61
Lattice pitch:	13.50 cm	
Outer width across flats:	13.16 cm	
Thickness of hexcan:	0.32 cm	
Inner width across flats:	12.52 cm	
Available volume in assembly:	135.75 cc per cm	
Fuel fraction:	35.66%	57.46%
Void fraction:	5.13%	3.38%
Cladding fraction:	13.90%	11.63%
Sodium fraction:	45.31%	27.53%

Table 5: PFBR core design data.

Active height of core:	100 cm
Active height of radial blanket:	160 cm
Active height of axial blanket:	2 × 30 cm
Core zone 1 volume:	1153.88 l
Core zone 2 volume:	1303.20 l
Radial blanket volume:	2606.40 l
Axial blanket volume:	1474.24 l
Total reference volume:	6537.72 l
Thermal power:	1250 MW
Average power density:	191.20 kW/l

The reference volume, which is used to specify the average power density in the core, only includes the available volume inside the fuel assemblies (135.75 cc per cm; see Table 4) and is identical to the volume of all burnup zones in the neutronics calculations.

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4. Z. Mian, A. H. Nayyar, R. Rajaraman, and M. V. Ramana, "Fissile Materials in South Asia and the Implications of the U.S.–India Nuclear Deal," *Science and Global Security*, 14 (2006): 117–143. A more detailed report by the authors has been published by the International Panel on Fissile Materials and is available at www.ipfmlibrary.org/southasia.pdf (July 14, 2007).

5. H. J. Bhabha and N. B. Prasad, "A Study of the Contribution of Atomic Energy to a Power Programme in India," presented at the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, 89–101.

6. This was the original plan. More recently a variety of ways of utilizing thorium in non-breeding thermal reactors and accelerator-driven systems have also been explored. Department of Atomic Energy, "Shaping the Third Stage of the Indian Nuclear Power Programme," 2001, www.dae.gov.in/publ/3rdstage.pdf (July 14, 2007) and K. Balakrishnan, S. Majumdar, A. Ramanujam, and A. Kakodkar, "The Indian Perspective On Thorium Fuel Cycles," in Thorium Fuel Utilization: Options and Trends, IAEA-TECDOC-1319, International Atomic Energy Agency, November 2002.

7. The DAE's current projections are that nuclear power would grow from the present 4,100 MW (as of May 2007) to 20,000 MW by 2020 with fast reactors contributing about 2,500 MW. By mid-century, the DAE expects that most of its reactors will be breeders, generating a total capacity of 275,000 MW. See R. B. Grover, and S. Chandra, "Scenario for Growth of Electricity in India," *Energy Policy* 34(17) (November, 2006): 2834–2847; M. R. Srinivasan, R. B. Grover, and S. A. Bhardwaj, "Nuclear Power in India: Winds of Change," *Economic and Political Weekly* XL(49) (December 3, 2005): 5183–5188.

8. We have relied largely on the figures found in the International Atomic Energy Agency's Fast Reactor Database, www-frdb.iaea.org (February 1, 2007). Some of the other documents used are: "Design of Prototype Fast Breeder Reactor," Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, December 2003; "Prototype Fast Breeder Reactor: Preliminary Safety Analysis Report," IGCAR, Kalpakkam, 2004; and S. C. Chetal *et al.*, "The Design of the Prototype Fast Breeder Reactor," *Nuclear Engineering and Design*, 236(2006): 852–860.

9. There are no assemblies without fissile material, that is, purely fertile material, in a homogeneous core. Such assemblies are only in the blanket region. In contrast, heterogeneous cores include assemblies with purely fertile material within the central core.

10. A. E. Waltar and A. B. Reynolds, *Fast Breeder Reactors* (New York: Pergamon Press, 1981), pp. 123–134.

11. D. G. Roychowdhury *et al.*, "Thermal Hydraulic Design of PFBR Core," in LMFR core thermohydraulics: Status and prospects, IAEA-TECDOC-1157, International Atomic Energy Agency, Vienna, June 2000, pp. 41–55, www.ipfmlibrary.org/iaea00.pdf (July 14, 2007).

12. These are average values expected for the overall reactor. Note that the importance of fission processes in fissile isotopes is lower in the blanket regions, where U-238 plays a more important role due to its very high relative content.

13. For the detailed design, a value of 1.049 is quoted in "National Presentations: India," pp. 3–5 in "Primary Coolant Pipe Rupture Event in Liquid Metal Cooled Reactors," Proceedings of a technical meeting held in Kalpakkam, India, January 13–17, 2003, IAEA-TECDOC-1406, August 2004, www.ipfmlibrary.org/iaea04a.pdf (July 14, 2007).

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15. A. Glaser, Neutronics Calculations Relevant to the Conversion of Research Reactors to Low-Enriched Fuel, Ph.D. Thesis, Department of Physics, Darmstadt University of Technology, April 2005.

16. J. F. Briesmeister, ed., MCNP—A General Monte Carlo N-Particle Transport Code, Version 4C, LA-13709-M, Los Alamos National Laboratory, December 2000; A. G. Croff, A User's Manual for the ORIGEN2 Computer Code, ORNL/TM7175, Oak Ridge National Laboratory, July 1980; and S. Ludwig, Revision to ORIGEN2, Version 2.2, Transmittal Memo, May 23, 2002.

17. Z. Xu, Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors, Ph.D. thesis, Massachusetts Institute of Technology, January 2003; and Z. Xu, P. Hejzlar, M. J. Driscoll, and M. S. Kazimi, "An Improved MCNP-ORIGEN Depletion Program (MCODE) and its Verification for High Burnup Applications," PHYSOR, Seoul, October 7–10, 2002. MCODE is used with the kind permission of its author.

18. MCNP-generated, spectrum-averaged cross-sections are generated for the 16 most important actinides and for the 45 most important fission products. These nuclides account for more than 99.95% of all neutron absorptions in the fuel. For the remainder, cross-sections from ORIGEN2 fast-reactor libraries (AMOPUUUC.LIB, AMOPUUUR.LIB, and AMOPUUUA.LIB) are used.

19. Fast Reactor Database: 2006 Update, 2006, *op. cit*. The database does not indicate at what point in the cycle is this the inventory of plutonium.

20. Preliminary Safety Analysis Report, 2004, *op. cit*.

21. Except in the case of the axial blanket zones, each burnup zones roughly corresponds to one circular row of fuel assemblies.

22. S. M. Lee *et al.*, "Conceptual Design of PFBR Core," in "Conceptual Designs of Advanced Fast Reactors," Proceedings of a Technical Committee meeting held in Kalpakkam, India, October 3–6, 1995, IAEA-TECDOC-907, International Atomic Energy Agency, Vienna, October 1996, pp. 83–99, www.ipfmlibrary.org/iaea96.pdf (July 14, 2007).

23. The simulation for the core begins with an initial exposure of the radial blanket of 450 EFPDs. When the average core assembly reaches its end-of-life, the blanket has been irradiated for 990 EFPDs. The average exposure of the blanket for this simulation is therefore equivalent to 720 EPFDs, that is, to the average exposure of the material in the real radial blanket. A similar approach is pursued for the irradiation of the radial blanket. Here, average core conditions are simulated by replacing the core and the axial blanket once every 360 EFPDs with material of an initial exposure of 90 EFPDs.

24. See Section 7.4, pp. 149–151, in M. Benedict, T. H. Pigford, and H. W. Levi, *Nuclear Chemical Engineering*, Second Edition (New York: McGraw-Hill, 1981).

25. The primary reason for this phenomenon is the fundamentally different cross-section ratios for neutron absorption in Pu-239 relative to U-238 depending on the energy of the incident neutron. Assume, for instance, a material composed of 1% Pu-239 and 99% U-238. In a fast neutron spectrum, less than 10% of all neutron absorptions occur in Pu-239, that is, plutonium buildup from neutron capture in U-238 is the still dominant process. In a thermal spectrum, however, about 75% of all neutrons are already absorbed in Pu-239, even though its concentration is so low. In other words, Pu-239 fission *and* Pu-240 buildup are now the main processes. Plutonium in spent fuel from light-water reactors typically contains less than 60% Pu-239 and more than 20% Pu-240.

26. Our choice of capacity factor is the same as the one chosen by the DAE in its studies, although it might be somewhat large in comparison with the experience of other prototype breeder reactors. The French Phénix reactor and the British PFR had cumulative load factors of 44% and 24%, respectively. The Russian BN-600 had a cumulative load factor of 74%, but has suffered 15 sodium fires in 23 years.

27. Waltar and Reynolds, *op. cit.*, pp. 234–236.

28. "National Presentations: India" in IAEA-TECDOC-1406, 2004, *op. cit.*, quotes a value of 1.049, the Preliminary Safety Analysis Report, 2004, *op. cit.*, a value of 1.04. Earlier design variants of the PFBR were estimated to have breeding ratios of 1.07 and higher.

29. Benedict *et al.*, *op. cit.*, remark: "Although it would be possible to reprocess the radial blanket material separately from the core and axial blanket material, it is proposed that these be mixed [. . .] and reprocessed together" (p. 151). Such combined reprocessing would lower the average burnup and thus the heat generation and the concentration of fission products that the reprocessing plant would have to deal with. Separate reprocessing of blanket and core has also been considered for uranium-fueled breeder reactors because it would reduce the requirement for fresh enriched uranium. See, for example, Appendix V of the Report to the American Physical Society by the Study Group on Nuclear Fuel Cycles and Waste Management, American Physical Society, New York, 1977.

30. APS Study Group, 1977, *op. cit.*, Appendix V.

31. If processing losses are considered, virtually no extra plutonium is generated in the reactor. For the calculated breeding ratio of 1.05, an annual plutonium discharge of 1047 kg, and processing losses of 2%, about 21 kg of plutonium are lost to the waste streams per year. This leaves a surplus of only 14 kg.

32. M. Hibbs, "DAE Reprocessing Program Remains Modest in Scope," *Nuclear Fuel*, 28(8), (April 14, 2003), p. 9.

33. Mian *et al.*, 2006, *op. cit.*

34. The possibility of using the breeder to convert reactor-grade plutonium to weapon-grade plutonium has been approvingly noted by at least one media commentator in a national newspaper: R. Ramachandran, "Is Breeder Needed for Strategic Purposes?," *The Hindu*, February 22, 2006.

35. Electricity from these is not likely to be economical in comparison to heavy-water reactors even if the latter are fueled with uranium costing several times more than current Indian or global prices. M. V. Ramana and J. Y. Suchitra, "Economic and Environmental Costs of Nuclear Power," in eds., B. S. Reddy and J. Parikh, *The Dynamics of Energy, Environment and Economy*: *New Challenges and Opportunities* New Delhi: Oxford University Press, forthcoming; M. V. Ramana, "Nuclear Economics in a Developing Country: The Case of India," Conference on the Future of Nuclear Energy, The Bulletin of the Atomic Scientists and the University of Chicago, November 1–2, 2006, Chicago, Illinois, USA.

36. Mian *et al.*, 2006, *op. cit.*, p. 143.

37. Assuming that two remaining reactors come online in 2015 and the last two reactors come online in 2020.

APPENDIX: ISOTOPICS OF PHWR-TYPE PLUTONIUM AND AVAILABLE INVENTORIES

The initial plutonium required to fabricate the fuel for the PFBR will be recovered from the spent fuel of several pressurized-heavy-water reactors (PHWRs) operated in India. We have generated candidate compositions of this fuel in burnup calculations simulating an infinite lattice of PHWR fuel assemblies of

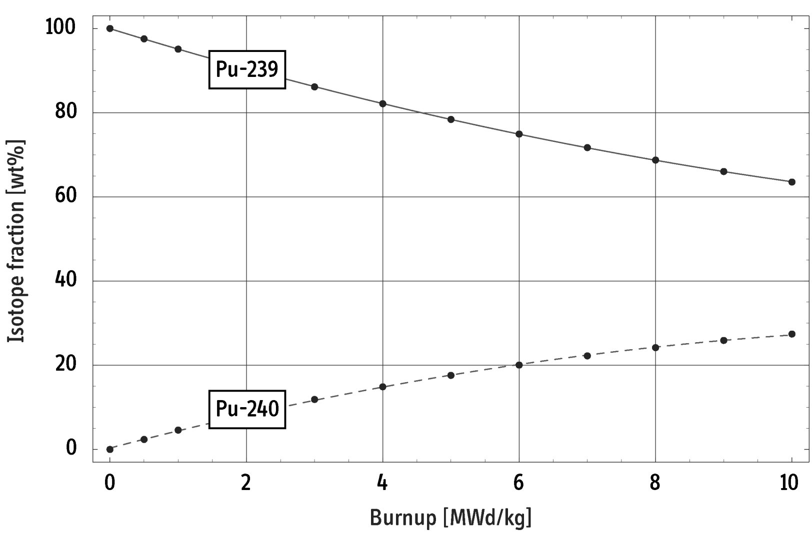

Figure 6: Plutonium-239 and -240 fractions in PHWR-fuel during irradiation.

Table 6: Plutonium compositions in PHWR-type fuel irradiated to various discharge burnup levels. Decay-corrected compositions are for a five-year storage period before reprocessing of the fuel.

		Pu-238	**Pu-239**	**Pu-240**	**Pu-241**	**Pu-242**
6.0 MWd/kg	at discharge	0.07%	74.92%	20.05%	4.19%	0.77%
	cooled	0.07%	75.60%	20.23%	3.32%	0.78%
7.0 MWd/kg	at discharge	0.09%	71.71%	22.23%	4.89%	1.08%
	cooled	0.09%	72.48%	22.46%	3.88%	1.09%
8.0 MWd/kg	at discharge	0.11%	68.75%	24.18%	5.52%	1.44%
	cooled	0.11%	69.58%	24.46%	4.39%	1.46%
9.0 MWd/kg	at discharge	0.13%	66.03%	25.90%	6.09%	1.85%
	cooled	0.13%	66.91%	26.24%	4.85%	1.87%
10.0 MWd/kg	at discharge	0.16%	63.52%	27.42%	6.61%	2.29%
	cooled	0.15%	64.44%	27.81%	5.27%	2.33%

the 19-pin type. Figure 6 shows the Pu-239 and Pu-240 fractions of irradiated PHWR-type fuel and Table 6 lists plutonium vectors for several burnup levels. In addition to the composition at discharge, a typical storage period of five years is assumed before the fuel is reprocessed. During that period, a fraction of the Pu-241 decays (half-life: 14.4 years), which increases the relative importance of some other plutonium isotopes. We do not take into account subsequent buildup of americium-241 or other isotopes in the material and in effect assume that the plutonium is fabricated into MOX immediately after reprocessing and used to fuel the PFBR.