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India's Baseline Plan for Nuclear Energy Self-sufficiency

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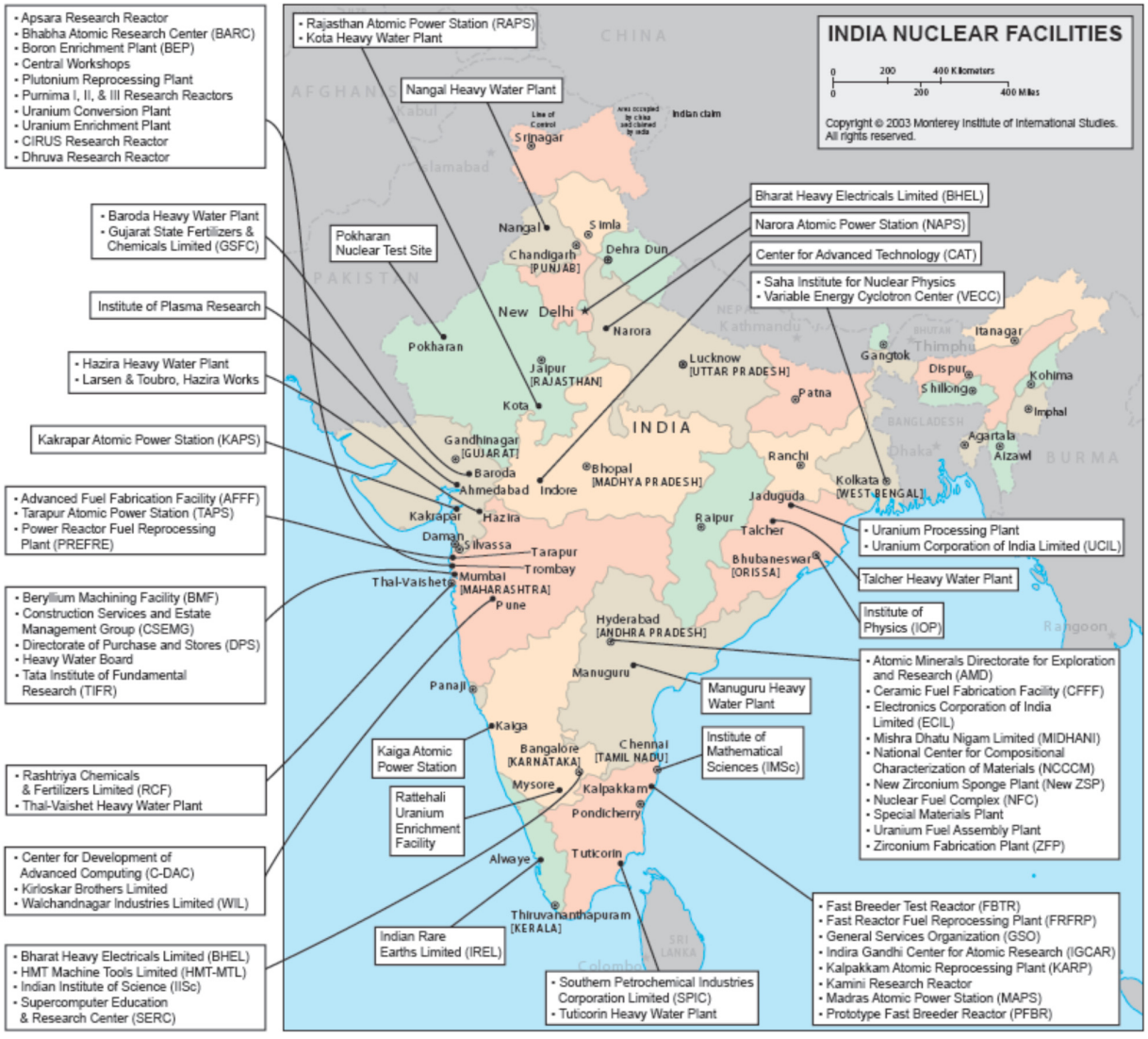

National Nuclear Security Administration

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On the cover
India's Nuclear facilities

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India's Baseline Plan for Nuclear Energy Self-sufficiency

by
R.G. Bucher
Technical Nonproliferation Policy Support Section
National Security and Nonproliferation Department
Nuclear Engineering Division
Energy Science and Engineering Directorate
Argonne National Laboratory

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1.0 Introduction

> *"Energy is the engine for economic growth and when the country moves ahead on the growth path, it is necessary to exploit every energy resource available in the country."*[1]

India's nuclear energy strategy has traditionally strived for energy self-sufficiency, driven largely by necessity following trade restrictions imposed by the Nuclear Suppliers Group (NSG) following India's "peaceful nuclear explosion" of 1974. On September 6, 2008, the NSG agreed to create an exception opening nuclear trade with India, which may create opportunities for India to modify its baseline strategy. The purpose of this document is to describe India's "baseline plan," which was developed under constrained trade conditions, as a basis for understanding changes in India's path as a result of the opening of nuclear commerce. Note that this treatise is based upon publicly available information. No attempt is made to judge whether India can meet specified goals either in scope or schedule. In fact, the reader is warned a priori that India's delivery of stated goals has often fallen short or taken a significantly longer period to accomplish.

It has been evident since the early days of nuclear power that India's natural resources would determine the direction of its civil nuclear power program. It's modest uranium but vast thorium reserves dictated that the country's primary objective would be thorium utilization.[i] Estimates of India's natural deposits vary appreciably,[2,3] but its uranium reserves are known to be extremely limited, totaling approximately 80,000 tons, on the order of 1% of the world's deposits; and nominally one-third of this ore is of very low uranium concentration. However, India's roughly 300,000 tons of thorium reserves account for approximately 30% of the world's total. Confronted with this reality, the future of India's nuclear power industry is strongly dependent on the development of a thorium-based nuclear fuel cycle as the only way to insure a stable,[ii] sustainable,[iii] and autonomous[iv] program.[4]

The path to India's nuclear energy self-sufficiency was first outlined in a seminal paper[5] by Drs. H. J. Bhabha and N. B. Prasad presented at the Second United Nations Conference on the Peaceful Uses of Atomic Energy in 1958. The paper described a three stage plan for a sustainable nuclear energy program consistent with India's limited uranium but abundant thorium natural resources. In the first stage, natural uranium would be used to fuel graphite or heavy water moderated reactors. Plutonium extracted from the spent fuel of these thermal reactors would drive fast reactors in the second stage that would contain thorium blankets for breeding uranium-233 (U-233). In the final stage, this U-233 would fuel thorium burning reactors that would breed and fission U-233 in situ. This three stage blueprint still reigns as the core of India's civil nuclear power program.

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[i] Thorium, unlike uranium, does not contain an isotope capable of sustaining the fission chain reaction necessary for a nuclear reactor. However, thorium can be converted into such an isotope [uranium-233 (U-233)] in, for example, nuclear reactors, providing fuel for a nuclear power industry; this conversion process is reviewed in detail in Appendix B.

[ii] stable: free from supply disruption and market volatility

[iii] sustainable: renewable through utilization of available resources to breed additional nuclear fuel

[iv] autonomous: self-sufficient via indigenous capabilities so as to not be subject to external influence

India's progress in the development of nuclear power, however, has been impacted by its isolation from the international nuclear community for its development of nuclear weapons and consequent refusal to sign the Nuclear Nonproliferation Treaty (NPT).[6,7] Initially, India was engaged in numerous cooperative research programs with foreign countries; for example, under the "Atoms for Peace" program, India acquired the Cirus reactor, a 40 MWt research reactor from Canada moderated with heavy water from the United States. India was also actively engaged in negotiations for the NPT. But, on May 18, 1974, India conducted a "peaceful nuclear explosion" at Pokharan using plutonium produced by the Cirus reactor,[8] abruptly ending the era of international collaboration. India then refused to sign the NPT, which it viewed as discriminatory since it would be required to join as a non-nuclear weapons state. As a result of India's actions, the Nuclear Suppliers Group (NSG) was created in 1975 to establish guidelines "to apply to nuclear transfers for peaceful purposes to help ensure that such transfers would not be diverted to unsafeguarded nuclear fuel cycle or nuclear explosive activities."[9] These nuclear export controls have forced India to be largely self-sufficient in all nuclear-related technologies.

2.0 Overview of India's Three Stage Program

In order to provide a secure, sustainable supply of nuclear energy given its limited uranium but abundant thorium reserves, India has remained committed to the three stage nuclear power program. The stages are essentially as initially proposed by Bhabha and Prasad:

Stage 1: Pressurized Heavy Water Reactors (PHWRs)
fueled with indigenous natural uranium,

Stage 2: Fast Breeder Reactors (FBRs)
fueled with plutonium, initially from PHWRs, and

Stage 3: Advanced Nuclear Power Systems
utilizing the thorium - uranium-233 fuel cycle.

Thorium utilization is the long term core objective of the India's nuclear power program.

Stage 1 is in the commercial domain. There are fifteen PHWRs presently generating a total of 3800 MWe with three PHWRs generating an additional 660 MWe scheduled to be in commercial operation by May, 2010. Although PHWRs are the mainstay of the first stage, there are also four foreign, turnkey reactors – two small vintage reactors in operation since 1969 and two 1000 MWe power reactors expected to be in operation in one to two years.

The stage 2 fast breeder reactors are in the technology demonstration phase. The Fast Breeder Test Reactor (FBTR), which became critical in October, 1985, has served primarily as a test bed for fuel development but has also generated a small amount of power for the electrical grid since 1997. The construction of a 500 MWe Prototype Fast Breeder Reactor (PFBR), which is to use mixed plutonium and uranium oxide fuel, was begun in October, 2004, and is expected to be commissioned in September, 2010.

The thorium utilization systems of stage 3 are in the technology development phase. These systems include a thermal reactor design based in part on demonstrated PHWR

technologies, an accelerator driven subcritical system similar in design to that of other countries, and a small, modular high temperature reactor for use in remote locations. The research and development for the thermal reactor have sufficiently advanced such that a 300 MWe technology demonstration is being planned for the near future.[10]

India's embrace of the thorium fuel cycle, which is necessary for the country's nuclear power program to be self sufficient, is unique in the international community. Its acceptance, together with India's isolation from the international nuclear community, dictated that implementation of this three stage program be accompanied by the indigenous development of many reactor and fuel cycle technologies. In the nearly half century since the program's inception, India has made significant progress, even without foreign technological or industrial assistance. A hallmark of their program since Bhabha has been to achieve self-sufficiency "in almost all aspects of nuclear science and technology, which include the entire gamut of operations relating to the nuclear fuel cycle,"[11] that is, not only in the science and engineering of power reactors and the related fields of construction and industrial manufacturing, but also in mining/milling operations, heavy water production, fuel fabrication, fuel reprocessing, and waste management.

2.1 Stage 1 Reactors

The objective of the first stage is twofold: to generate plutonium for fueling the second stage fast breeder reactors and to produce power for India's electrical grid. The stage 1 reactors consist of both indigenously developed and foreign origin systems. The indigenous stage 1 reactors would, over their lifetimes, consume all of India's limited uranium reserves;[v] however, the plutonium separated from the spent fuel of these first stage reactors would provide the fissile material needed to start the deployment of the second stage fast breeder reactors. The foreign reactors, which have recently become more important in the planning to meet India energy goals, would be fueled with foreign fabricated fuel; however, all of the plutonium produced in these reactors would also be available for the fast breeder reactors.[12]

2.1.1 Pressurized Heavy Water Reactors (PHWRs)[13]

Pressurized Heavy Water Reactors (PHWRs), which are similar to Canada's Canada Deuterium-Uranium (CANDU) reactors,[14,15] were determined to be the most efficient reactor design in terms of uranium utilization.[16] PHWRs use natural uranium oxide (UO_2) as fuel and heavy water[vi] (D_2O) as both moderator and coolant. The heavy water rapidly thermalizes (slows) fission neutrons with minimal neutron absorption; the resulting high neutron economy not only allows the use of natural uranium fuel but also provides more

[v] More specifically, the indigenous stage 1 reactors would consume that portion of the uranium reserves directed to India's civil nuclear power program; the allocation between its civil nuclear power and its military nuclear weapons programs is not known.

[vi] Heavy water (D_2O) is a form of water which contains two deuterium (D) atoms rather than two hydrogen (H) atoms. Deuterium is an isotope of hydrogen with an atomic weight of two since the nucleus contains a neutron in addition to the proton found in the hydrogen nucleus.

"surplus" neutrons to breed plutonium, important for the startup of the second stage fast breeder reactors.

The PHWR was also a natural choice given India's existing infrastructure. The use of natural (unenriched) uranium as fuel eliminated the need for India to develop enrichment technologies and construct costly facilities that would only be required for stage 1 reactors.[vii] In addition, India already possessed a heavy water production facility which had been in operation since 1962.[17] Finally, because the PHWR does not use a high-pressure reactor vessel, India's indigenous industrial technologies were sufficient to fabricate the required reactor components. Therefore, the selection of PHWRs as the stage 1 reactors provided India not only the optimum path for power generation and plutonium production from its modest uranium reserves but also the most economical utilization of its nuclear-related technology and infrastructure.

PHWRs differ from light water reactors in the coolant and moderator, using heavy water rather than "light" water (H_2O), and in the fuel, using natural uranium (0.7% U-235) rather than low enriched uranium (nominally less than 5% U-235).

The PHWR design is radically different from typical light water reactor designs, especially for the moderator and coolant which in the PHWR are distinct, independent systems. The heavy water moderator, at nearly ambient pressure and temperature, is contained by a low-pressure reactor vessel, named the "calandria," whereas the coolant is confined in a separate high pressure, high temperature circuit around each fuel bundle. The horizontal-oriented calandria contains several hundred channels that span the length of the vessel. Each of these channels has two concentric tubes: the outer, identified as the calandria tube, and the inner, the pressure tube. The calandria tubes form the "inner" surface confining the heavy water moderator in the calandria and providing a channel for the pressure tubes to pass entirely through the calandria. The pressure tubes contain the bundle of fuel rods and the heavy water coolant at an elevated pressure and temperature. This pressure tube design concept provides a high fertile conversion ratio since the neutron spectrum is well-thermalized due to the low moderator temperature achieved by separating the coolant and moderator systems.

India's first PHWR, Rawatbhata-1, was a collaborative venture between Atomic Energy of Canada Ltd (AECL) and Nuclear Power Corporation of India Ltd (NPCIL); Canada's Douglas Point CANDU reactor served as the reference design. This first unit was commissioned in 1973; though initially rated at nominally 200 MWe, it was derated to 100 MWe early in its lifetime. The second PHWR, Rawatbhata-2, based on the same design but constructed with significantly less Canadian assistance, was commissioned in 1981; it is currently rated at 200 MWe.

Following India's nuclear weapons test in 1974, NPCIL was able to continue the construction of PHWRs, reducing the import content to 10-15 %.[18] By 2001, ten additional PHWRs, each rated at 756 MWt / 220 MWe, had been completed and placed in

[vii] The nuclear device that India tested in 1974 was based on plutonium, not enriched uranium; hence, enrichment technologies had most probably not been developed for India's nuclear weapons program.

commercial operation; one additional unit of this capacity has just recently been added to this list. NPCIL also has three 220 MWe systems nearing completion at existing nuclear power plant sites – Kaiga and Rawatbhata; all three units are expected to be on-line by May, 2010.

In the past two years, NPCIL has added two 540 MWe PHWRs, Tarapur-3 and -4. These plants are scaled-up versions of the standard 220 MWe unit with upgraded design and safety features and "new systems" as required, all based on indigenous technologies and capabilities. With the commissioning of these facilities, India claimed that it "has entered the era of electricity generation from large size nuclear reactors."[19]

The experience gained from the development and construction of the 540 MWe PHWRs at Tarapur is being utilized for uprating the unit size to 700 MWe – the larger capacity further reducing the capital cost per MWe. Four of these reactors are proposed for existing sites, two at Rawatbhata and two at Kakrapar.[20] The detailed design work for these 700 MWe units has progressed well and actions required for the procurement of long lead time items have begun.[21]

Since India's known uranium reserves are quite limited, the first stage was envisioned as only a 10,000 MWe PHWR program with duration of thirty years.[22] The PHWR currently installed generating capacity of 3800 MWe and soon to be installed capacity of 660 MWe totals 4460 MWe, or approximately 45% of the planned deployment. The existing fifteen PHWRs, and two turnkey foreign BWRs discussed below, are specified in the list of stage 1 nuclear power plants in operation summarized in Table I. Three PHWRs, and two foreign VVERs also discussed below, nearing completion are identified in the stage 1 nuclear power plants under construction in Table II. In stage 1, India has "been able to demonstrate commercial excellence with indigenously developed technology both with respect to operating power reactors as well as with reactor projects under construction."[23]

2.1.2 Foreign Stage 1 Light Water Reactors (LWRs)

In addition to the indigenous PHWRs, stage 1 also includes four turnkey Light Water Reactor (LWR) systems of foreign design, constructed with foreign origin components.

Two of these reactors, General Electric 160 MWe Boiling Water Reactors (BWRs), date from the inception of India's nuclear power program. The contract for these units, awarded in 1964, allowed Indian scientists and engineers to participate in all phases of design, construction, and commissioning – the "well thought out strategy" being to acquire the technology and train the professionals to establish a nuclear culture.[24]

Table I. Stage 1 Nuclear Power Plants in Operation [25]

Unit ID and Location			Reactor Type	Power (MWe)	Date of Commercial Operation
1	TAPS-1	Tarapur, Maharashtra	BWR	160	28-Oct-1969
2	TAPS-2	Tarapur, Maharashtra	BWR	160	28-Oct-1969
3	RAPS-1	Rawatbhata, Rajasthan	PHWR	100	16-Dec-1973
4	RAPS-2	Rawatbhata, Rajasthan	PHWR	200	01-Apr-1981
5.	MAPS-1	Kalpakkam, Tamilnadu	PHWR	220	27-Jan-1984
6	MAPS-2	Kalpakkam, Tamilnadu	PHWR	220	21-Mar-1986
7	NAPS-1	Narora, Uttar Pradesh	PHWR	220	01-Jan-1991
8	NAPS-2	Narora, Uttar Pradesh	PHWR	220	01-Jul-1992
9	KAPS-1	Kakrapar, Gujarat	PHWR	220	06-May-1993
10	KAPS-2	Kakrapar, Gujarat	PHWR	220	01-Sep-1995
11	KAIGA-1	Kaiga, Karnataka	PHWR	220	16-Nov-2000
12	KAIGA-2	Kaiga, Karnataka	PHWR	220	16-Mar-2000
13	RAPS-3	Rawatbhata, Rajasthan	PHWR	220	01-Jun-2000
14	RAPS-4	Rawatbhata, Rajasthan	PHWR	220	23-Dec-2000
15	TAPS-4	Tarapur, Maharashtra	PHWR	540	12-Sept-2005
16	TAPS-3	Tarapur, Maharashtra	PHWR	540	18-Aug-2006
17	KAIGA-3	Kaiga, Karnataka	PHWR	220	06-May-2007

Table II. Stage 1 Nuclear Power Plants under Construction [26]

Unit ID and Location			Reactor Type	Power (MWe)	Scheduled Commercial Operation
1	KAIGA-4	Kaiga, Karnataka	PHWR	220	Mar-2008
2	RAPP-5	Rawatbhata, Rajasthan	PHWR	220	Feb-2008
3	RAPP-6	Rawatbhata, Rajasthan	PHWR	220	Oct-2008
4	KK-1	Kudankulam, Tamilnadu	VVER	1000	Dec-2007
5	KK-2	Kudankulam, Tamilnadu	VVER	1000	Dec-2008

Since the BWRs operation began in October, 1969, each unit has received "comprehensive indigenous backfits and upgrades" but has remained in service. After India conducted its first nuclear weapons test in 1974, the United States government halted fuel shipments to these reactors. Since that time, India's "experience in securing reliable fuel supply has not been satisfactory." [27] Recently, Russia has supplied the necessary low enriched uranium,[28,29] which India fabricated into fuel assemblies at its Nuclear Fuel Fabrication (NFC) facility.[30]
In order to accelerate the nuclear power program, India has adopted the policy of importing LWR technology in parallel with its indigenous three stage program.[31] The other two foreign origin reactors are 1000 MWe Vodo-Vodyanoy Energeticheskyi Reaktors (VVERs), being constructed at Kudankulam "in collaboration with Russian Federation." Russia's export of this technology to India is not subject to Nuclear Suppliers Group (NSG) controls because the project predates the establishment of the NSG.[32] These units are scheduled to be in operation in December 2010 and 2011; up to six VVERs are envisaged at that site.[33]

2.2 Stage 2 Reactors

The objective of stage 2 is to breed uranium-233 (U-233) as well as additional plutonium for second and third stage reactors and to produce power to support India's future economic expansion. The fissile content (U-235) of the natural uranium reserves allocated to the civilian nuclear power program will be depleted by the first stage PHWRs. Hence, it is necessary for India to close the nuclear fuel cycle – that is, to reprocess the spent PHWR fuel and recycle the fissile and fertile materials in fast breeder reactors. These reactors will increase the fissile material inventory of both plutonium and U-233 and also expand India's nuclear power capacity. In fact, studies suggest that the plutonium bred from India's uranium reserves is sufficient to support 500,000 MWe of power generation in second and third stage reactors.[34]

India's nuclear power program envisions using breeder reactors to produce fissile material by converting two fertile isotopes, U-238 and Th-232, into fissile materials.[viii]
U-238, which comprises 99.3% of natural uranium, is converted to the fissile element plutonium (Pu-239, Pu-240, etc.); and Th-232, which is essentially 100.0% of natural thorium, is converted to the fissile isotope U-233. The conversion of U-238 allows India to more effectively utilize its limited natural uranium resources, whereas the conversion of Th-232 allows it to access the energy potential of its massive thorium reserves.

Estimates of the increase in uranium utilization from the use in fast breeder reactors vary by nearly a factor of two, depending on numerous parameters. By one estimate, domestic uranium can provide about 29 EJ from a once-through fuel cycle in PHWRs, whereas the same uranium can provide about 3700 EJ through multiple cycles in fast breeder reactors, that is, 128 times more energy.[35] However, a conservative estimate is "at least sixty five times more energy."[36]

[viii] Details of the conversion of these fertile isotopes into fissile materials, along with the production of "less-desirable" isotopes, are presented Appendix B.

The second stage fast breeder reactor cores will be fueled with reactor grade plutonium and depleted uranium, both obtained by reprocessing the spent fuel from the first stage PHWRs. (The Indian Department of Atomic Energy also intends to extract the plutonium produced

in the foreign origin LWRs for fueling the second stage fast breeder reactors.[37]) The plutonium will be the fissile component to drive the reactor and the uranium, the fertile component to breed the additional plutonium. The blankets surrounding the cores will be composed of thorium, from India's abundant reserves, in order to breed U-233. Of the newly-bred fissile materials, the plutonium will be used to fuel additional stage 2 fast breeder reactors; and both the plutonium and U-233 will become the driver fuel for the startup of the thorium utilization reactors, and other systems, of stage 3.

2.2.1 Fast Breeder Test Reactor (FBTR)[38,39,40,41]

India's initial stage 2 reactor, the Fast Breeder Test Reactor (FBTR), is a 40 MWt / 13.2 MWe, sodium-cooled, loop-type fast reactor. The core, an hexagonal grid of 745 assemblies, contains a central fuel region, primarily a small mixed carbide fuel zone whose makeup has varied over time, surrounded respectively by a nickel reflector, thorium dioxide (ThO_2) (also called thoria) blanket, and steel reflector.

FBTR serves not only to investigate the science and engineering of fast breeder reactors but also to develop heat removal and electrical generation technologies. The two sodium loops of the primary cooling system transfer heat generated in the core to the corresponding secondary sodium loop via intermediate heat exchangers; each of the secondary loops, in turn, transfers the heat to two once-through steam generators. The steam from all four steam generators is fed to a common steam-water circuit with a turbine-generator that is synchronized to the electrical grid. The schematic flow diagram for the FBTR power plant is presented in Fig. 1.

The FBTR was conceived to serve as a test-bed for the irradiation of fuels and materials for fast reactors and to gain experience in sodium technology and fast reactor operation. Its use for fuel development was immediately realized when, because the required enriched uranium was not available, the core had to be reconfigured from the planned mixed oxide to an untested, unique mixed carbide composition; subsequent irradiation tests have examined other FBR fuels and PHWR claddings. With more that twenty years of operating experience, FBTR has "produced the full complement of personnel, fully competent in the operation and maintenance of fast reactor systems," including the immediate response and corrective actions to two major incidents.[42]

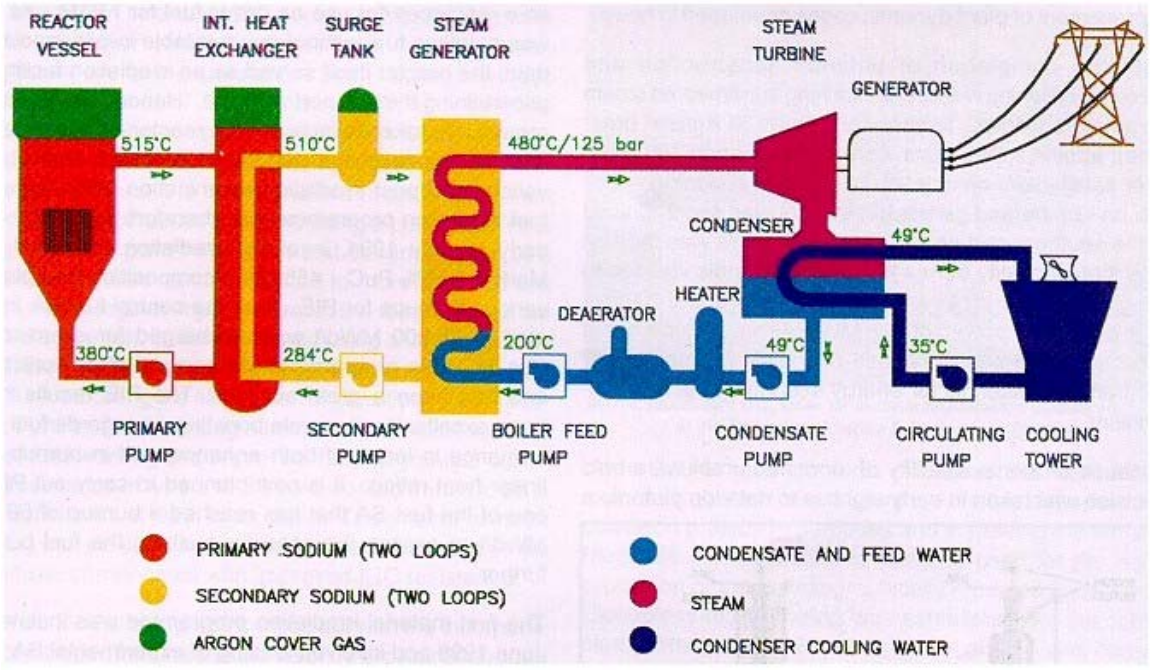


Fig. 1. Schematic Flow Sheet for the Fast Breeder Test Reactor (FBTR) Power Plant [43]

2.2.1.1 FBTR Development History

The FTBR was largely based on the French Rapsodie Fortissimo reactor at Cadarache. Facility civil construction began at the Indira Gandhi Center for Atomic Research (IGCAR)[44] in Kalpakkam in 1972 and was completed by 1977. However, because of India's nuclear isolation, the anticipated supply of highly enriched uranium required to fabricate the mixed oxide fuel for the initial core design was not received from France. The core was reconfigured to be a smaller, mixed carbide core with a significantly reduced power; initial criticality was not achieved until October, 1985. At this time, India became the sixth member of the elite club of nations with fast reactors, joining the United States, France, Russia, the United Kingdom, and Japan.

The fabrication and installation of equipment for electrical power generation progressed over the next twelve years. Steam generators were connected to the secondary sodium loops and put in "sodium service" in 1989. With the commissioning of the steam-water and leak detection systems, the steam generators produced superheated water for the turbo-generator which was then synchronized with the electrical grid. Thus, in July, 1997, stage 2 fast breeder reactors began to contribute to India's power requirements.

At its inception, India had intended that FBTR would be constructed primarily with indigenous technologies and industrial capabilities. However, the French contributed significantly in several ways: 1) the basic design of the nuclear systems was obtained from the French Atomic Energy Commission (CEA), 2) manufacturing technology for some important nuclear system components was transferred from France, and 3) a few operational personal were trained at Rapsodie. India's modifications, in addition to

reconfiguring the reactor core from a mixed oxide to a mixed carbide fuel, included most notably the secondary heat removal system and electrical generation equipment, which were required since Rapsodie was an experimental fast reactor which simply dissipated its heat to the atmosphere. In the end, India estimates that the indigenous component was more than 80%.[45]

2.2.1.2 FTBR Initial Core Loadings, Mixed Carbide Fuels

The FBTR as initially designed was to operate at 40 MWt with a core of sixty-five mixed oxide (MOX) fuel assemblies containing 30% PuO_2 and 70% UO_2 with a uranium enriched of 85%, the same composition as the core of the French Rapsodie reactor. The plutonium was to be obtained from the reprocessed spent PHWR fuel and the highly enriched uranium, from France.

The reconfigured FBTR core consisted of only twenty-two mixed carbide fuel assemblies containing 70% PuC and 30% UC with the uranium being unenriched. This small carbide core achieved criticality in October, 1985, but its power was rated at only 10.2 MWt. Additional fuel assemblies with this Mark-I composition have been inserted to compensate for the reactivity loss due to burnup. In 1996, in order to increase the power level, thirteen Mark-II fuel assemblies, with a composition of 55% PuC and 45% UC, were added at the core periphery; the maximum power from this configuration was 17.4 MWt, reached in 2002.[46]

The original fuel assemblies at the core center were not replaced in the 1996 core expansion so that the ultimate burnup capability of the Mark-I fuel could be assessed. Since the carbide fuel design was unique and hence untested, its initial burnup limit was set at a modest 25,000 MWd/t. Numerous post-irradiation inspections of fuel pins removed from these assemblies at various levels of burnup have been used to monitor fuel performance metrics, such as fuel-clad gap, fuel voids, and fuel pin swelling. The burnup rating for the Mark-I carbide fuel has been steadily increased; by July, 2006, it had reached a burnup of 155,000 MWd/t without a single fuel pin failure.[47,48]

In spite of India's success with carbide fuel in FBTR, there are problems implementing a carbide fuel cycle.[49] For example, because carbide fuel is highly pyrophoric and highly susceptible to oxidation and hydrolysis, large scale fuel fabrication is extremely difficult. In addition, because this fuel does not readily dissolve in nitric acid, leaving behind organic complexes, reprocessing spent carbide fuel also presents technical challenges. Therefore, FBTR is being used as a test bed for developing fuels for the follow-on fast breeder reactors.

2.2.1.3 FBTR Fuel Irradiation Studies, Mixed Oxide Fuels

In 2003, configured with twenty-eight Mark-I and thirteen Mark-II mixed carbide fuel assemblies,[50] FBTR began its first test of the MOX fuel designed for the Prototype Fast Breeder Reactor, discussed below. The mission was to irradiate a single, simulated MOX fuel assembly, containing 29% PuO_2 and 71% UO_2, to a burnup of 100,000 MWd/t at the

designed linear heat rating of 450 W/cm. To achieve this high linear heat rating, the uranium in the test MOX fuel has been enriched in U-233. This experimental MOX assembly had reached a burnup of 59,200 MWd/t by 2006;[51] the irradiation to the target burnup is still in progress.[52] It is anticipated that this study will provide "insights" to both MOX and U-233 irradiation behavior.

The next step in MOX fuel testing at FTBR began in late 2006 with the loading of a hybrid core. The hybrid core consists of a mixed carbide central zone surrounded by mixed oxide fuel with a high plutonium content - the carbide zone serves as the primary driver with the oxide zone increasing the total power. In the first phase of the transition to the hybrid core, the MOX zone consists of only eight assemblies; the fuel is 44% PuO_2 and 56% UO_2, the same composition that is being studied for actinide burning in fast reactors by other countries.[53] Irradiation began in February, 2007, at the target power of 16.6 MWt.[54] It is anticipated that that the fuel's irradiation performance will "contribute to the current international knowledge base on this fuel."[55]

2.2.1.4 FTBR Additional Fuels Development and Material Irradiations

Metallic, rather than oxide, fuel technologies are being developed in order to accelerate the rate of deployment of breeder reactors for meeting the future energy demands more effectively. FBTR will be deployed to test this metallic fuel, including activities necessary to demonstrate closing the fuel cycle.

There is also a plan to tap the breeding potential of FBTR for generating U-233 in the thorium dioxide blankets. It is estimated that FBTR would be able to generate about one hundred kilograms of U-233 in its operating lifetime.[56]

In addition, FBTR has successfully demonstrated that the irradiation creep behavior of the indigenously developed zirconium-niobium (Zr-Nb) alloy conforms to international standards.[57] This material has replaced Zircaloy in the pressure tubes in all PHWRs constructed since KAPS-1. This upgrade extends the working lifetime of the pressure tubes beyond the twelve to fifteen years afforded by the Zircaloy tube, reducing reactor maintenance.[58]

2.2.2 Prototype Fast Breeder Reactor (PFBR)[59,60]

Building on the knowledge and experience gained from FBTR, India has embarked on the construction of an industrial-scale fast breeder reactor, the Prototype Fast Breeder Reactor (PFBR). Although the construction has only recently begun, extensive design and engineering analyses and experiments in all aspects of the related technologies have been undertaken since the early 1980s in order to optimize the various systems and maximize their economics.[61] The as-designed PFBR is a 1250 MWt / 500 MWe[ix] sodium-cooled, pool-type fast reactor with mixed plutonium-uranium oxide fuel and a thorium oxide

[ix] The power rating was chosen in order to standardize the design of reactor components and to utilize the turbine from fossil fueled plants.

blanket. The main objective of PFBR is to demonstrate the technical and commercial viability of fast breeder reactors.

In PFBR's unique pool design,[62] the main reactor vessel contains a hot (820° K) and a cold (670° K) sodium pool separated by an inner vessel. The heat generated in the reactor core is removed by two primary cooling loops which circulate the sodium from the cold pool, through the bottom grid plate into the core, to the hot pool. The heat in the radioactive sodium in this primary loop is transferred through four intermediate heat exchangers to two independent non-radioactive secondary sodium loops, each of which contains a sodium pump, two intermediate heat exchangers, and four steam generators. The steam from the eight steam generators is directed to the turbo-generator which is connected to transformers on the electrical grid. The schematic flow diagram for the PFBR power plant is presented in Fig. 2.

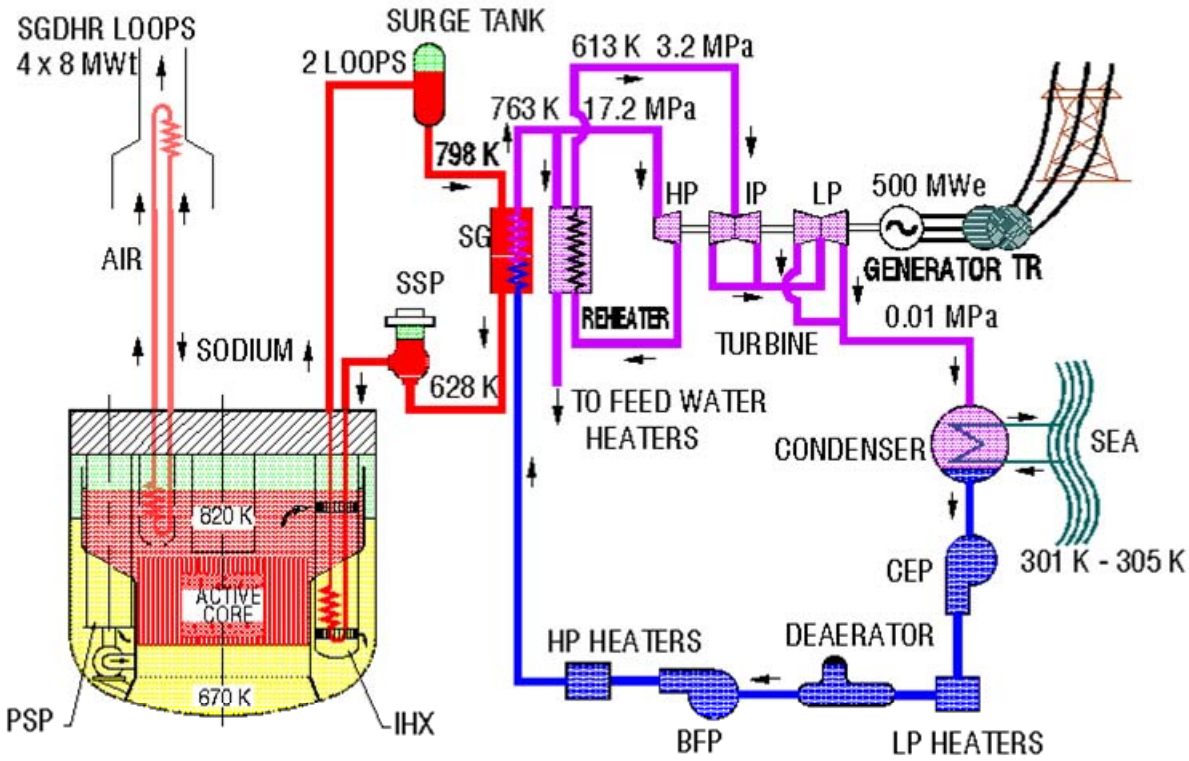


Fig. 2. Schematic Flow Sheet for the Prototype Fast Breeder Reactor (PFBR) Power Plant[63]

The PFBR is being constructed at Kalpakkam by Bharatiya Nabhikiya Vidyut Nigam Ltd.[64] (BHAVINI). BHAVINI is a new company that was formed at the suggestion of Indian Department of Atomic Energy (DAE) specifically to build the PFBR – the intent was to combine the project management skills of NPCIL with the scientific and technological expertise of the Indira Gandhi Centre for Atomic Research (IGCAR), Bhabha Atomic Research Center (BARC), and other DAE laboratories.[65] Site preparation began in August, 2003, with the first concrete pour in October, 2004. According to the initial schedule, criticality was to be achieved in 2010 and commercial operation, to commence

the following year. However, construction is progressing faster than planned and now the facility is expected to be commissioned in September, 2010.[66] The design life of the PFBR is expected to be 40 years.[67]

2.2.2.1 As-Designed PFBR Core Configuration

The core of the PFBR consists of 181 fuel assemblies using two enrichment zones in order to flatten the radial power distribution – an inner zone with 21% PuO_2 and 79% UO_2 and an outer zone with 28% PuO_2 and 72% UO_2, where in each zone the uranium is unenriched. The fuel region has a diameter of 1.90 m and a height of 1.00 m; the larger core volume, relative to FBTR, allows the use of a fuel with a low plutonium content and natural uranium because the neutron leakage is reduced. Oxide fuel, rather than carbide fuel as in FBTR, was chosen because of the difficulties in both large scale fabrication and spent fuel reprocessing of carbide fuel.[68]

Surrounding this core are two rings[x] of thorium dioxide blanket, two rings of steel reflector, and one ring of inner shielding. The two core enrichment zones have upper and lower axial blankets, each with a thickness 0.30 m. Around the inner shield are two rings of "internal storage," in which fuel assemblies are cooled for one irradiation campaign, three additional rings of steel reflector, and nominally six rings of outer shielding.

2.2.2.2 Advanced Fuel Design for PFBR

An enhanced reliance on fast breeder reactors is anticipated in order to help India satisfy its increasing energy demand. Since the growth in India's installed capacity of PHWRs is limited by its uranium reserves, the necessary increase in fissile material inventory can only be accomplished by the fast breeder reactor's potential to produce sufficient additional plutonium. Hence, the rate of increase in power production by stage 2 and stage 3 breeder reactors is dependent on the rate of plutonium production in the stage 2 fast breeder reactors. The greater the fast breeder reactor's breeding ratio,[xi] the shorter its doubling time,[xii] and a short doubling time allows more frequent construction of additional reactors.

The breeding ratio, and hence the doubling time, of the as-designed PFBR oxide fuel, however, are not sufficient to support the required growth rate. The breeding ratio of a reactor is dependent not only on its design, especially its neutron economy, but also on the chemical form of its fuel. The maximum breeding ratio[69,70] achievable with oxide fuel,

[x] In the hexagonal grid of a reactor core, a ring is an annular region with the thickness of a single assembly; a ring is sometimes also referred to as a row.

[xi] The breeding ratio is the ratio of the amount of fissile material produced to that consumed. A breeding ratio greater than one means more fissile material is created in the operation of the reactor than is required to refuel it; the excess fissile material would contribute to fueling an additional reactor.

[xii] The doubling time is the time required to breed sufficient excess / surplus fissile material to fuel another reactor of the same design; the doubling time is proportional to the inventory of the fissile material and inversely proportional to the rate of consumption of the fissile material and the breeding ratio. This doubling time considers only the irradiation time in the reactor, not the time required by the fuel cycle operations to reprocess the spent fuel and to fabricate fresh fuel; the latter must be included in any proper performance assessment of the total fuel cycle.

such as that designed for the PFBR, is nominally 1.1. For carbide fuel, such as that used in FBTR, and for nitride fuel, it ranges around 1.2 to 1.3. The highest breeding ratio is obtained from metal fuel, from 1.4 to 1.5. Correspondingly, the doubling time[71, 72] is nominally twenty-two years for oxide fuel, thirteen years for carbide fuel, and ten years for metal fuel. The time required to obtain the fissile material to fuel an additional reactor is also impacted by the cooling and unloading / loading times and the "out-of-pile" fuel cycle operations to reprocess the spent fuel and to fabricate fresh fuel. India estimates the out-of-pile time to be one year for metal fuel and two years for oxide and carbide fuels.[73]

Hence, even though the PFBR and other early units of the same design are to be fueled with oxide fuel, India has a focused program for the development of short doubling time metal fuel for the follow-on fast breeder reactors.[74] The long term objective is to convert to metal fuel to provide the needed fuel performance enhancement when the technology becomes available, anticipated to be shortly after 2020.

2.3 Stage 3 Systems[75]

The objective of stage 3 is to achieve a sustainable nuclear fuel cycle by developing thorium–U-233 based systems that utilize India's vast thorium reserves to provide long-term energy security with nuclear power. New systems are being engineered to optimize the use of plutonium produced in stage 2 fast breeder reactors, one, to maximize the conversion of thorium to U-233, two, to extract power in-situ[xiii] from the thorium fuel, and, three, to recycle the bred U-233 in additional reactors. In addition, systems based on the thorium fuel cycle offer both neutronic and non-proliferation advantages over plutonium fuel cycles. These stage 3 concepts are to be implemented in parallel with the continuing development and deployment of stage 1 and stage 2 reactors and fuel cycle operations.

India has three innovative nuclear concepts in the design and development phases. The primary system for implementing the thorium utilization strategy is an advanced thermal reactor that draws on the proven PHWR pressure tube and heavy water technologies to satisfy each of the three design objectives stated above. A second, non-reactor, technology utilizes an accelerator and a subcritical assembly not only for the efficient conversion of thorium and possible generation of power but also for the incineration of long-lived actinides and fission products obtained from spent fuel reprocessing. The third system is a compact modular reactor suitable either for the production of electrical energy in remote areas or for the generation of process heat for the conversion of fossil fuels. Each of these concepts is discussed in detail below.

If these stage 3 systems achieved the goals of good economic performance and high levels of passive safety, India could begin large scale commercial deployment for electrical generation as well as for non-electrical applications, such as the desalination of sea water and the generation of portable, non-fossil fuels.

[xiii] Power is generated in-situ through fissioning of U-233 bred from thorium in the fuel since its fabrication.

India's early stage 3 initiatives have encompassed small scale activities in all relevant technologies for the entire thorium fuel cycle. This includes the successful development and operation of Kamini, the Kalpakkam Mini Reactor, the only U-233 fuelled reactor in the world currently in operation. Constructed at the IGCAR with fuel that was bred, processed, and fabricated indigenously, this reactor achieved criticality in 1996 and began full power operation at 30 kWt the following year. In addition to its mission as a U-233 fueled test reactor, it also functions as a neutron source for radiography and activation analysis. These modest but important indigenous research activities have laid the foundation for the development of the thorium fuel cycle and signaled India's entrance into the third stage of its nuclear power program.

2.3.1 Advanced Heavy Water Reactor (AHWR)[76,77]

The primary reactor system envisioned for stage 3, the Advanced Heavy Water Reactor (AHWR), contains both evolutionary and revolutionary design concepts. The AHWR is a 920 MWt / 300 MWe heavy water moderated but light water cooled[xiv] reactor that uses the well-proven pressure tube technology of the PHWR. The low-pressure reactor vessel, the "calandria," and the concentric calandria and pressure tubes are similar in design to those in the PHWR, except the calandria is oriented vertically, rather than horizontally. Vertical pressure tubes allow the removal of core heat through natural circulation of the boiling light water coolant, avoiding the need for primary coolant pumps and, hence, adding a measure of operational reliability and passive safety. The schematic flow diagram for the AHWR power plant is presented in Fig. 3.

In addition to power generation, the AHWR is also intended to desalinate sea water at the rate of 500 cubic meters (132,000 gal) per day. If desired, desalination capacity can be increased, with each 1000 cubic meters (264,000 gal) per day reducing the gross electrical output an estimated 0.95 MWe.

The present concept for the equilibrium[xv] AHWR core, which is housed in the low-pressure calandria, contains 452 pressure tubes, each loaded with an identical fuel cluster. This cluster, as shown in Fig. 4, consists of three concentric rings of fuel pins around a central rod assembly. The twenty-four pins in the outer ring are loaded with a mixture of thorium and plutonium oxides (ThO_2-PuO_2); in order to obtain "favorable minimum critical heat flux ratios," the plutonium content is 4.0% in the lower half of the active fuel and 2.5% in the upper half. The other two rings of fuel pins contain a mixture of thorium and U-233 oxides (ThO_2–U-$233O_2$) with a U-233 content of 3.0 % in the twelve pins of the inner ring and 3.75% in the eighteen pins of the middle ring. The central rod assembly contains twelve pins of dysprosium oxide in a zirconium dioxide matrix (DyO_2-ZrO_2) and a central

[xiv] In contrast to the PHWR which uses heavy water as both moderator and coolant, the AHWR is able to use heavy water as the moderator and light water as the coolant because thorium, rather than uranium, is the dominant component of the fuel mixture. Light water is a stronger absorber of neutrons than heavy water, but thorium is a stronger absorber than uranium; the net effect is that the fraction of neutrons lost by "parasitic" absorption in coolant, moderator, and structural material is reduced.

[xv] The equilibrium core, in this instance, is achieved once operations in the fuel cycle have produced sufficient fissile materials such that the fissile loading can be maintained; that is, the core loading pattern is in steady state.

channel for water from the Emergency Core Cooling System (ECCS). The light water coolant flows through the cluster in the spaces among the three concentric rings of fuel pins, but outside the central rod assembly.

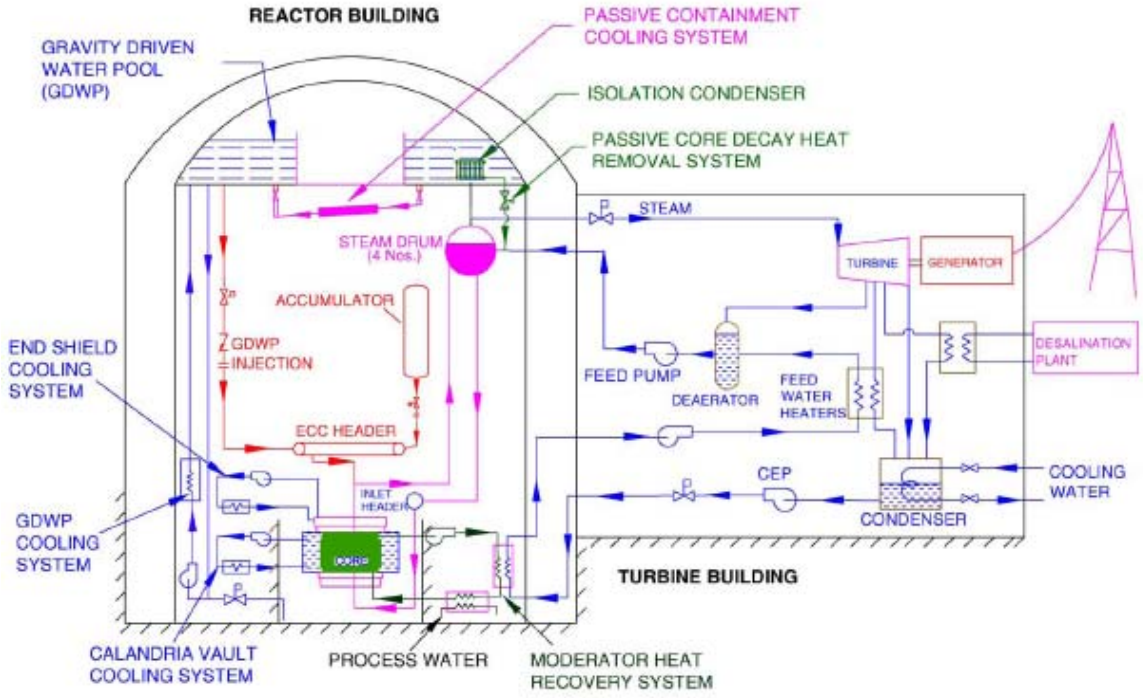


Fig. 3. Schematic Flow Sheet for the Advanced Heavy Water Reactor (AHWR) Power Plant[78]

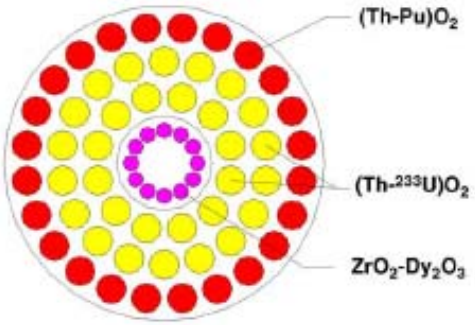


Fig. 4. Cross Section of AHWR Fuel Cluster[79]

The first of the two primary objectives of the equilibrium core design was to optimize thorium utilization, by maximizing both the conversion of thorium to U-233 and the power extracted in-situ from thorium. Maximum breeding of U-233 is needed to produce sufficient fissile material for recycle in AHWRs in order to attain the self-sufficiency characteristic required from a stage 3 reactor design. Power extracted in-situ is important for minimizing the initial inventory and consumption of the plutonium. Overall, the fuel

cluster design was to achieve at least 60% of the power production from thorium and U-233 while maintaining an average burnup of 24,000 MWd/t at discharge.

The second of the two primary objectives of the equilibrium core design was to enhance passive safety. Inherent safety means, most importantly, insuring a negative void coefficient of reactivity[xvi] under both operating and accident conditions. This is achieved by hardening[xvii] the neutron spectrum and inserting a burnable poison. The harder spectrum is obtained by decreasing the relative quantity of the heavy water moderator in the vicinity of the fuel cluster, that is, by positioning the clusters closer together. The burnable poison is added by inserting the neutron absorbing element dysprosium in the central rod assembly. Inherent safety is also achieved by engineered safety systems that actuate passively. In one scenario, depressurization of the main heat transport system during a loss of coolant accident causes failure of a rupture disc and floods the central channel in each fuel cluster from the ECCS accumulator tanks. In another, high steam pressure caused by loss of heat sink causes failure or another rupture disc and introduces a neutron absorbing "poison" into the moderator to quench the fission chain reaction.

The AHWR concept envisions a closed nuclear fuel cycle. Initially, both the thorium and U-233 recovered from the spent fuel will be used to fabricate fresh fuel pins for the AHWR fuel cluster. However, the reprocessed plutonium, with its increased concentration of higher plutonium isotopes and the presence of higher actinides, is to be stored for later fueling of fast breeder reactors; the plutonium for the fresh fuel will be obtained from recycled spent PHWR fuel. In the long term, when transmutation systems based on fast breeder reactors and accelerator driven subcritical systems have sufficient capacity, the fuel cycle will be extended so as to take advantage of the synergies between the various concepts in all three stages of India's nuclear program.

2.3.2 Accelerator Driven Subcritical Systems (ADS)

A second power production system for thorium utilization is being investigated – an Accelerator Driven Subcritical System (ADS). In such a system, a high current proton accelerator delivers a beam of protons onto a spallation[xviii] target that functions as an external source of neutrons to drive a subcritical assembly.[xix] In the Indian concept, fissile U-233 and fertile Th-232 in the subcritical assembly are bombarded by the externally produced neutrons, causing either fission in the U-233, producing more neutrons and generating heat, or neutron capture by the Th-232, breeding more fissile U-233. However,

[xvi] A negative void coefficient of reactivity means that the core's reactivity decreases as coolant's voids increase, that is, as the coolant density decreases. Therefore, as the reactor power increases, as in an accident condition, the coolant density decreases and the reactivity decreases, driving the reactor toward a safer configuration.

[xvii] A "hard" spectrum contains, on average, faster neutrons than a thermal spectrum; a spectrum is hardened by not allowing the neutrons to slow to thermal velocities, that is, by limiting collisions in the moderator.

[xviii] In nuclear physics, spallation is the process in which a heavy nucleus bombarded by high energy protons emits a large number of nucleons (neutrons and protons).

[xix] A subcritical assembly is not able to sustain a fission chain reaction – the number of neutrons produced by fission is not sufficient to overcome the number lost by capture or leakage.

since the configuration is subcritical, these processes continue while the external neutron source is present but decay away when the accelerator is turned off.

The ADS could be a technology for providing not only a self sufficient power production system but also a method to help alleviate problems associated with waste disposal. A properly designed ADS system will be able, in principle, to produce more fissile material than is consumed, also offering the promise of being able to reduce the doubling time for fissile material production, even with thorium.[80] In addition, the fissions generate a sufficient heat to produce several times more electricity than is needed in the operation of the facility. An ADS system also affords the possibility of reducing the quantity of high-level nuclear waste that requires long-term geologic storage by transmuting long-lived actinides and fission products separated during the recycling operations into shorter-lived isotopes; this technology is being investigated, using both accelerators and fast reactors, in research laboratories in many nations.

India has outlined a phased approach to develop an ADS system, including critical facility measurements to validate relevant neutronic data, design and construct of cyclotron and linear accelerator facilities, and development of a molten, heavy metal spallation target. Physics design of the Low Energy High Intensity Proton Accelerator (LEHIPA) has been completed at BARC[81] and simulation studies are being conducted on the main magnet of the 10 MeV cyclotron at the Variable Energy Cyclotron Center.[82] Development of the spallation source is also in-progress.[83]

2.3.3 Compact High Temperature Reactor (CHTR)[84,85]

India is also developing a small modular reactor concept, the Compact High Temperature Reactor (CHTR), as an integral component of the stage 3 objective of utilizing its thorium resources to satisfy various energy needs. In addition, its development is serving as a demonstration of technologies relevant for next generation high temperature reactor systems.

The CHTR is intended to be able to produce either electrical power or process heat. It's compact, modular design makes it suitable for supplying electrical power to remote, difficult to access areas not connected to the nation's power grid. Its high operating temperature provides process heat suitable for the production of alternative transportation fuels such as hydrogen and the refinement of low–grade coal and oil to recover liquid fuels.[86] Both forms of the CHTR's energy output can be used concurrently in the cogeneration of electricity and potable water.

The CHTR reactor is fuelled mainly with mixed uranium-thorium carbide having a U-233 content of 33.75%, moderated with beryllium oxide, and cooled with metallic lead-bismuth eutectic alloy. The reactor is able to operate without on-site refueling for an estimated fifteen years of continuous duty at full power. The current design operates at 1000°C but generates only 100 kWt in this, the design prototype; larger units are to be explored. The CHTR has passive systems for reactor control, shutdown, and heat removal under normal and postulated accident conditions. In addition, there are many inherent safety features:

strong negative Doppler coefficient for the fuel, high thermal inertia and low power density of the core, large temperature margin on fuel design, negative temperature coefficient for the moderator, etc.

The CHTR core, shown in Fig. 5, consists of three rings (nineteen elements) of hexagonal ("prismatic") beryllium oxide (BeO) "blocks," each containing a graphite "fuel tube" with twelve "fuel bores" around a central coolant channel. The fuel is made of TRISO[xx] coated fuel particles embedded in a graphite powder matrix loaded in cylindrical "fuel compacts" and inserted in the fuel bores. A ring of eighteen beryllium oxide blocks surround the core region serving as a reflector to scatter neutrons back into the core and housing a passive control / safety system responsive to outlet coolant temperature. In addition, graphite material in the outer ring contains channels to return the coolant to the bottom of the core.

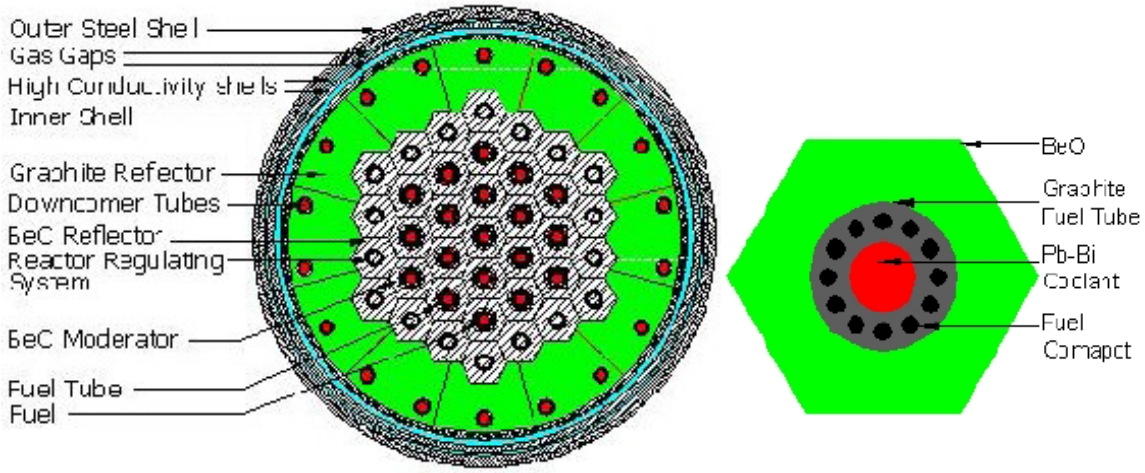


Fig. 5. Compact High Temperature Reactor (CHTR) Core Cross-Section Design (Left) and Single Fuel Block (Right)[87]

This assemblage is contained in a shell of high-temperature resistant and liquid-metal-corrosion resistant material. A vertical cross-section diagram of the reactor identifying not only the various core elements but also the control / shutdown, coolant and structural components is presented in Fig. 6.

Heat is passively removed from the core under both normal and abnormal operating conditions. During normal operations, the lead-bismuth coolant flows by natural circulation from the lower to the upper plenum through the central channels in the fuel tubes, returning through the "downcomers" at the reactor periphery. A set of sodium heat pipes in the upper plenum then passively transfers this heat with a minimum temperature loss to heat-utilization vessels which provide the interface to systems for high temperature heat applications.

There are also three passive safety systems for heat removal in the event of an abnormal condition or accident. One is a siphon system to flood gas-filled gaps in the core with

[xx] TRISO – tristructural-isotropic

molten metal to facilitate heat conduction to the outside. Another is a set of six variable conductance heat pipes located in the upper plenum to transfer the core's heat to the atmosphere under the loss of load condition when the primary coolant circuit remains intact. The last is a system of twelve carbon composite variable conductance heat pipes located in the core that service the need when the primary coolant is lost. These systems, each of which is individually capable of removing the equivalent of 200 kWt, can act together or independently to limit the temperature increase in the core and coolant.

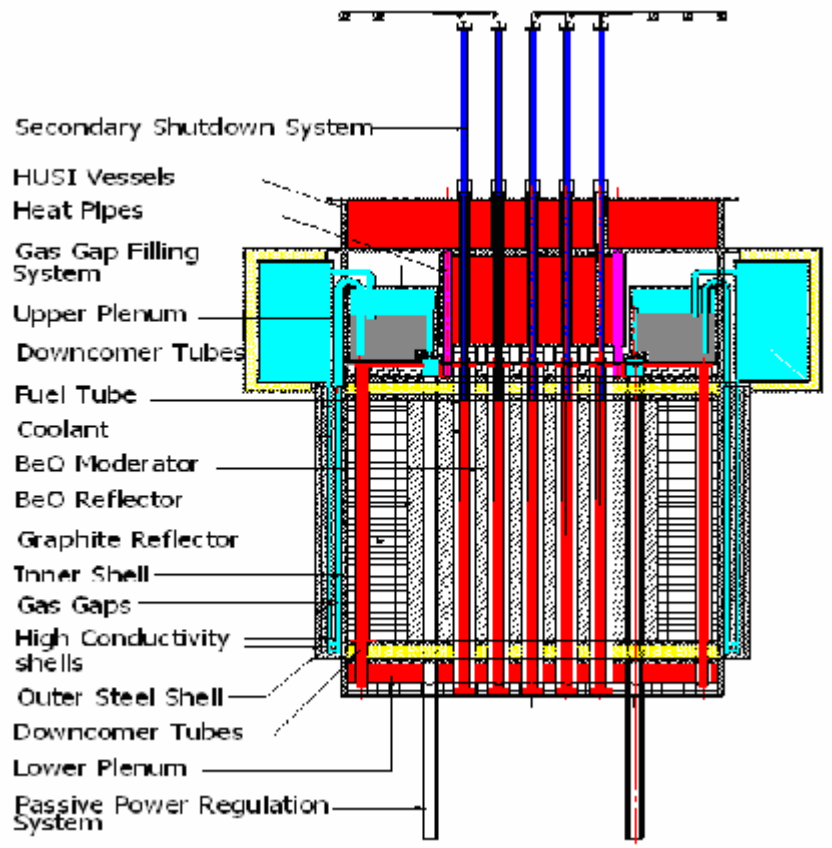


Fig. 6. Compact High Temperature Reactor (CHTR) Vertical Cross-Section Design[88]

3.0 Nuclear Power Production Goals[89,90,91]

For nuclear power to contribute significantly to the energy needs of India's one billion-plus people and rapidly expanding economy, the DAE has adopted a near-term goal of generating 20,000 MW (20 GW) of electricity from nuclear reactors by the year 2020, a program initially identified as "Vision 20-20." To achieve this target, a major construction effort will need to take place in the decade from 2010 to 2020.[92] Reactors from each of the three stages are expected to help produce the 20,000 MW.

Naturally, the stage 1 reactors will supply the largest portion of this energy. After the five reactors listed in Table II have been commissioned, currently scheduled for the end of 2011, the total installed capacity for stage 1 reactors will be 6,780 MW – 4,460 MW from

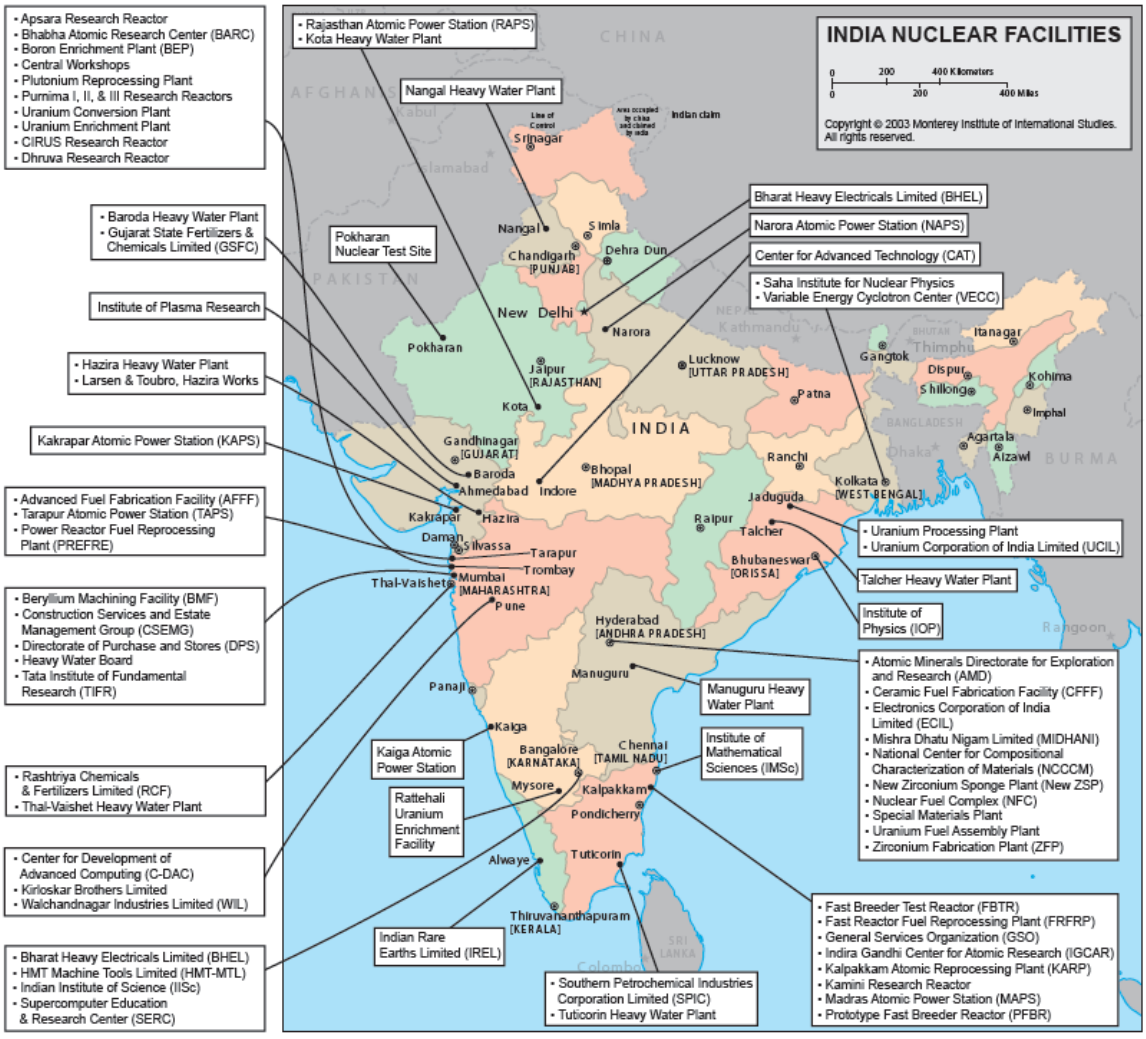
eighteen Indian PHWRs, 320 MW from two American BWRs, and 2000 MW from two Russian VVERs. DAE plans to construct six uprated 700 MW PHWRs; preliminary geological investigations have already begun for four of these units. Therefore, by 2020 the total power produced by these stage 1 reactors will be 10, 980 MW; with 8,660 MW from its indigenous PHWRs, that program is near the 10,000 MW limit originally envisioned for India's modest uranium reserves.

Since stage 2 is in the test demonstration phase and stage 3 is in the technology development phase, their contributions will be relatively minor. In addition to the initial 500 MW PFBR expected to be commissioned by 2010, DAE plans to have four more 500 MW units in operation by 2020. All five PFBRs are to be loaded with oxide fuel however, IGCAR states that one "will be operated with metallic fuel ([since the] design will have the flexibility to accept metallic fuel, although the reference design is for oxide fuel)."[93] The initial 300 MW AHWR of stage 3 is also anticipated to be in operation by 2020. Hence, the total power generated by stage 2 and stage 3 reactors will be 2800 MW.

Even given its demonstrated success in the PHWR program and significant progress in PFBR development, India's research, development, and construction efforts are not sufficient to attain the 20,000 MW goal. The planned additional reactors raise the total installed electrical capacity to only 13,780 MW. Hence, to accelerate its nuclear power program, India has opted to import technology by purchasing another six 1000 MW LWRs; reports in 2004 indicated that these reactors would be more Russian VVERs.[94] Since the restrictions on India's international trade in nuclear technology have been lifted, it is entirely possible for India to obtain these foreign LWRs; therefore, the country will essentially be able to meet its 2020 goal.

India's plans for the expansion of nuclear power beyond 2020 are even more ambitious. Expecting that by that date metal fuel technology will be fully developed and industrial constructional capabilities for large fast breeder reactors will available, the DAE intends to begin installing as many metal-fueled, 1000 MW FBRs as the supply of plutonium from thermal reactors allows. Based on this approach, nuclear power could contribute 25% of India's total electrical power by 2052.

Appendix A. India's Nuclear Facilities[95]



Appendix B. Issues with the U-233 / Th-232 Fuel Cycle [96,97,98,99,100,101,102,103]

As was stated in the Introduction, this report documents India's baseline nuclear energy strategy. Although this strategy has existed since 1958, India, between 1974 and 2008, was constrained to this path since nuclear commerce was closed to it by the NSG. However, significant drawbacks, not entirely clear in 1958, exist with the U-233 / Th-232 fuel cycle as compared to the Pu-239 / U-238 fuel cycle. The recent opening of nuclear commerce for India provides an opportunity for India to shift its baseline nuclear strategy to avoid those drawbacks. This appendix, therefore, highlights those drawbacks since they may form the impetus for such a strategy change.

B.1 Naturally Occurring Thorium

Natural occurring thorium consists of nearly 100% Th-232. Th-228 is a decay product of Th-232 and exists in secular equilibrium with it in proportion to their half-lives. Th-234 and Th-230 exist as members of the U-238 decay chain. Similarly, Th-231 and Th-227 exist as members of the U-235 decay chain. Due to their relative short half-lives, Th-234, Th-231, and Th-227 are inconsequential in separated thorium. Th-230, on the other hand, has a much longer half-life and its contribution to the activity of separated thorium is proportional to the atomic ratio of U to Th in the ore.

Because of the relatively short half-life (1.91-yr) of Th-228, a situation occurs with separated thorium that does not occur with separated uranium. Th-228 decays rapidly as compared to the uranium isotopes and since its daughters have very short half-lives, the activity of separated thorium increases rapidly from that immediately after separation. The activity reaches a steady-state value of three times the initial activity after about one-month. The activity then decays with a half-life of 1.91-yr as the Th-228 in the separated thorium ore decays. After about 4-yr, the activity again increases as the Ra-228 daughter of Th-232 produces new Th-228.The activity levels off at about four times the original after about 40-yr.

The most significant hazards from separated thorium are caused by gaseous Rn-220 and high-energy gammas from Bi-212 and Tl-208. It has been stated[xxi] that the activity of separated thorium is about 90 times as hazardous as separated uranium and, because of the situation described above, increases to about 270 times as hazardous within one-month.

B.2 Physics of Basic Conversion Processes

The transmutation of fertile[xxii] isotopes into fissile[xxiii] materials occurs (mostly) through a sequence of neutron capture(s) and beta decay(s). A neutron capture, in which the nucleus absorbs a neutron, increases the isotope's atomic weight by one but leaves its atomic number unchanged; for example,

[xxi] Domke, Bill, private communication.
[xxii] A fertile isotope is one that is not fissile but one that can be converted into fissile material.
[xxiii] A fissile isotope is one that can be split into two nominally equal fragments upon the absorption of a thermal neutron. A fissile material is capable of sustaining a nuclear fission chain reaction.

U-238 + n → U-239.

A beta decay, in which the nucleus emits a negatively-charged beta particle (β^-),[xxiv] increases the atomic number by one but leaves the atomic weight unchanged; for example,

U-239 → Np-239 + β^-,

where the atomic number of uranium, U, is 92 and of neptunium, Np, is 93.

In the transmutations of nuclear reactor fuels, the neutrons captured are (essentially all) fission neutrons from the nuclear chain reaction and the beta emissions are spontaneous radioactive decays with various half lives ($t_{½}$).

B.2.1 Basic Conversion Sequences for U-238 and Th-232

The basic transmutation sequences for the two fertile isotopes in India's nuclear power program, U-238 and Th-232,[xxv] are straightforward – a single neutron capture followed by two successive beta decays.

$$\text{U-238} \xrightarrow{\text{(n capture)}} \text{U-239} \xrightarrow[t_{½} = 23\ \text{m}]{(\beta^- \text{ decay})} \text{Np-239} \xrightarrow[t_{½} = 2.4\ \text{d}]{(\beta^- \text{ decay})} \text{Pu-239}$$

$$\text{Th-232} \xrightarrow{\text{(n capture)}} \text{Th-233} \xrightarrow[t_{½} = 22\ \text{m}]{(\beta^- \text{ decay})} \text{Pa-233} \xrightarrow[t_{½} = 27\ \text{d}]{(\beta^- \text{ decay})} \text{U-233}$$

By comparing the half-lives of Np-239 to Pa-233, one can see a problem that exists with the thorium fuel cycle that does not exist in the uranium fuel cycle. If one waits a negligible amount of time between irradiation and reprocessing, a portion of the potentially recoverable U-233 will be lost. Simultaneously, some of the Pa-233 will remain during reprocessing; rapid reprocessing, therefore, will require separation of protactinium which is one of the more difficult elements to separate from uranium. This suggests that a significant delay between irradiation and reprocessing would be appropriate.

B.2.2 More Comprehensive Conversion Sequences for U-238

Since nuclear fuel is subject to continuous neutron irradiation, the fissile material product is not the pure Pu-239 and U-233 indicated above. As Pu-239 is produced, subsequent neutron captures cause the creation of heavier isotopes of plutonium, such as, Pu-240, Pu-241, etc. These "higher" plutonium isotopes are subject to various types of decay, including beta decay. The result is the simultaneous production of isotopes of americium (atomic number 95) and, with further neutron capture and beta decay, curium (atomic number 96);

[xxiv] A neutron in the nucleus decays into a (positive) proton and a (negative) beta particle (an electron), which is emitted from the nucleus; that is, n → $p^+ + \beta^-$.

[xxv] These two fertile isotopes are actually fissionable; there is a small probability of fission (fission cross sections on the order of several tenths of a barn) upon the absorption of a fast neutron (neutron energy greater than approximately 1.5 MeV).

in nuclear reactor terminology, these elements are referred to as the minor actinides.[xxvi] Hence, the conversion of U-238 yields isotopes of plutonium, americium, and curium. The relative concentration of each isotope is dependent on the total neutron irradiation experienced – the greater the irradiation, the greater the relative concentration of higher plutonium isotopes and minor actinides.

B.2.3 More Comprehensive Conversion Sequences for Th-232

In the production of U-233, there are several transmutation sequences which cause the product to become contaminated with a small amount U-232. This leads to another problem with the thorium fuel cycle. Two of the U-232 production sequences involve an (n,2n) reaction in which the nucleus absorbs a single neutron but immediately emits two neutrons; hence, the atomic weight decreases by one and the atomic number remains the same. For isotopes considered in this context, the (n,2n) reaction is initiated only by fast neutrons with energies greater than 5 MeV.[xxvii]

One of these two sequences is simply the (n,2n) reaction of U-233 which immediately produces U-232, as follows:

U-233 → (n,2n) → U-232

The other is a sequence that is initiated by the (n,2n) reaction of Th-232 and contains two beta decays and a neutron capture, as follows:

$$\text{Th-232} \xrightarrow{(n,2n)} \text{Th-231} \xrightarrow[t_{1/2} = 26\ \text{h}]{(\beta^- \text{ decay})} \text{Pa-231} \xrightarrow{(n \text{ capture})} \text{Pa-232} \xrightarrow[t_{1/2} = 1.3\ \text{d}]{(\beta^- \text{ decay})} \text{U-232}$$

The fast neutrons that initiate either of these (n,2n) reactions are neutrons from the fission of either the U-233 contained in the original nuclear fuel or the fission of the U-233 that was bred from the Th-232. In the first instance, if the fuel is removed from the vicinity of the breeding material, allowing the fission neutrons to lose energy before reaching this material, the probability of an (n,2n) reaction and, hence, the production of U-232, is reduced. In the second instance, since the fissioning of U-233 is not significant until appreciable quantities of U-233 have been bred, the concentration of U-232 is determined by the quantity of U-233 produced[xxviii] – smaller production quantities have lower U-232 contamination.

A third sequence for the production of U-232 begins with a neutron capture in the minor thorium isotope Th-230 and progresses from Th-231 as does the (n,2n) reaction of Th-232, as follows:

[xxvi] In general, the minor actinides are fissionable, but at thermal energies, neutron capture is more likely; consequently, these actinide elements are more readily consumed (fissioned or transmuted) in a fast reactor.

[xxvii] Only a small number of neutrons in a fast or thermal reactor have energies greater than 5 MeV; these neutrons are the first generation neutrons released during the fission process.

[xxviii] The quantity of U-233 produced depends on the neutron flux (number and energy of the neutrons) at the Th-232 and on the irradiation time.

Th-230 → (n capture) → Th-231 → (β^- decay, $t_{½}$ = 26 h) → Pa-231 → (n capture) → Pa-232 → (β^- decay, $t_{½}$ = 1.3 d) → U-232

The quantity of U-232 produced by this sequence obviously depends on the fraction of Th-230 in the thorium. All thorium ores contain a small quantity of Th-230 since it is created by the alpha decay of U-234. Therefore, the higher the uranium to thorium ratio in the ore deposit, the higher the Th-230 concentration. The relatively pure thorium ores found in the monazite sands of India have a very low Th-230 concentration (0.5 ppm). This may allow India to design a fuel cycle[xxix] with fewer handling problems associated with higher levels of U-232 contamination resulting from the utilization of other world resources of thorium ore. If the Th-230 concentration in an ore was appreciable (20-50 ppm), then this mechanism would contribute the majority of the U-232 production in a similarly designed fuel cycle.

U-232 is also formed by the chain initiating with U-235:

U-235 → (n capture) → U-236 → (n capture) → U-237 → (β^- decay, $t_{½}$ = 6.75 d) → Np-237 → (n,2n) →

Np-236 → (β^- decay, $t_{½}$ = 22 h) → Pu-236 → (α decay, $t_{½}$ = 2.85 yr) → U-232

B.3 Consequences of U-232, Th-224, and Th-228 in Irradiated Thorium Fuel

U-232, with a relatively short half life of 69 years, decays by the emission of an alpha particle to Th-228; note that Th-228 is also a daughter product in the decay chain of Th-232.[xxx] From this point, the decay proceeds more rapidly, one branch leading to Tl-208, which beta decays, and also emits a high energy (2.6 MeV) gamma ray, to the stable isotope Pb-208. Bi-212, with its 1.8 MeV gamma, is also part of this chain.

Even though the U-232 is present in only small concentrations, the high energy gammas produced in its decay chain present a significant handling problem in the separation and fabrication of fuel containing the bred U-233. For example, ten days after separation, the gamma dose at one meter from 1-kg of U-233 containing just 100-ppm of U-232 is 10-mRem/hr; this dose increases to a peak value of approximately 1-Rem/hr after about 10-yr. At Oak Ridge national Laboratory, where more than 1200 containers of U-233 containing as much as 220-ppm U-232 have been stored for more than 30-yr, the gamma dose at 1-cm from some containers measures 300-Rem/hr.

[xxix] The fuel cycle must be based on core designs and irradiation schemes that minimize U-232 production from the (n,2n) reactions of U-233 and Th-232.

[xxx] Since the half life of Th-232 is 14 billion years, the vast majority of the Th-228 present is from the decay of U-232.

The U-232 makes fabricating fuel containing U-233 very difficult. It will either have to be fabricated very soon after reprocessing or it will have to be done behind very heavy shielding, such as within a hot cell. Furthermore, the freshly fabricated fuel will have to be handled and transported the way spent fuel is handled and transported today.

The Th-224 and Th-228 in the irradiated thorium fuel presents appreciable activity in the separated thorium. At Savannah River in the 1960s[xxxi], recovered thorium was put in railroad tank cars that were then put in an isolated yard for several years while these isotopes decayed sufficiently for the radiation levels to drop to a level where the thorium could be treated like natural thorium.Because of this, it may not be practical in a commercial fuel cycle to recycle the recovered thorium until it has been stored for a period of 5 to 20-yr.

B.4 Increased Doubling Time

The fast fission cross-section of U-238 is 4 to 5 times greater than that of Th-232. Partially because of this, use of Th-232 in a fast spectrum reactor leads to a lower breeding ratio. Consequently, the difference in doubling time, the amount of time to double the amount of fissile material assuming a growing number of breeder reactors, is significant. For one fast breeder reactor scenario, it has been estimated that the doubling time when utilizing the U-235-Pu-239 fuel cycle is 17.8-yr while the doubling time when utilizing the U-233-Th-232 fuel cycle is 108-yr.

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Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue, Bldg. 208
Argonne, IL 60439-4842

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