NUCLEAR FUEL CYCLE ASSESSMENT OF INDIA:

A TECHNICAL STUDY FOR U.S.–INDIA COOPERATION

A Dissertation

by

TARAKNATH WODDI VENKAT KRISHNA

Submitted to the Office of Graduate Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

DOCTOR OF PHILOSOPHY

December 2007

Major Subject: Nuclear Engineering

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ABSTRACT

Nuclear Fuel Cycle Assessment of India:

A Technical Study for U.S.–India Cooperation. (December 2007)

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The recent civil nuclear cooperation proposed by the Bush Administration and the Government of India has heightened the necessity of assessing India's nuclear fuel cycle inclusive of nuclear materials and facilities. This agreement proposes to change the long-standing U.S. policy of preventing the spread of nuclear weapons by denying nuclear technology transfer to non-NPT signatory states. The nuclear tests in 1998 have convinced the world community that India would never relinquish its nuclear arsenal. This has driven the desire to engage India through civilian nuclear cooperation. The cornerstone of any civilian nuclear technological support necessitates the separation of military and civilian facilities. A complete nuclear fuel cycle assessment of India emphasizes the entwinment of the military and civilian facilities and would aid in moving forward with the separation plan. To estimate the existing uranium reserves in India, a complete historical assessment of ore production, conversion, and processing capabilities was performed using open source information and compared to independent reports. Nuclear energy and plutonium production (reactor- and weapons-grade) was

simulated using declared capacity factors and modern simulation tools. The three-stage nuclear power program entities and all the components of civilian and military significance were assembled into a flowsheet to allow for a macroscopic vision of the Indian fuel cycle.

A detailed view of the nuclear fuel cycle opens avenues for technological collaboration. The fuel cycle that grows from this study exploits domestic thorium reserves with advanced international technology and optimized for the existing system. To utilize any appreciable fraction of the world's supply of thorium, nuclear breeding is necessary. The two known possibilities for production of more fissionable material in the reactor than is consumed as fuel are fast breeders or thermal breeders. This dissertation analyzes a thermal breeder core concept involving the CANDU core design. The end-of-life fuel characteristics evolved from the designed fuel composition is proliferation resistant and economical in integrating this technology into the Indian nuclear fuel cycle. Furthermore, it is shown that the separation of the military and civilian components of the Indian fuel cycle can be facilitated through the implementation of such a system.

ACKNOWLEDGEMENTS

For the provision of such an interesting research topic, the opportunity to work on an internationally important current issue and for the constant support I would like to thank my advisor Dr. William S. Charlton. In particular, I gratefully appreciate the trust he had in me and that he introduced me so successfully to the world of research.

I am indebted to Dr. Paul Nelson for the academic supervision of my research and for keeping me focused. Further more I am grateful to Dr. Jean Ragusa for stopping me from overlooking the minute details and for carrying out comparative studies of various systems.

Especially I appreciate Dr. Marvin L. Adams willingness to share good ideas as time permitted him from his busy schedule. I would like to thank Dr. Pavel Tsvetkov for his indispensable advice and help in matters of core analysis and approach to system modeling.

It would not have been possible without the continued support and help on SCALE code by David Ames and Ayodeji. I cannot fail to mention all the PhD students working in the department for cheering me up when I was losing ground.

There are hardly enough words to thank my parents Basanti and Pattabhiram for all they have done for me. I am grateful for your constant support and trust in me in all my endeavors. And thanks to my wife Divya who always brightened my day and still didn't take me seriously. Last, but not least thanks to my sister Jeaniee and brother-in-law Tanuj. You listened uncomplainingly to every piglet story and did not lose patience even when I was in challenging times.

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I. INTRODUCTION

I.A. Motivation for this Study

The recent civil nuclear cooperation proposed by the Bush Administration and the Government of India has highlighted the necessity for an accurate assessment of India's nuclear fuel cycle. This agreement, which proposes significant changes to long standing U.S. nuclear policy, has expectedly created considerable controversy due to tensions between two widely held American foreign policy objectives: (1) strengthening bilateral relationships with emerging powers and (2) preventing the proliferation of nuclear weapons. The U.S. has long sought to build a relationship with India, a rising power and ambitious nuclear state, since it first exploded an atomic bomb in 1974. India reiterated its resolve to possess nuclear weapons with its second test in 1998. Since 1974, successive U.S. administrations pursued a policy of technology denial and nuclear trade isolation until India relinquished its nuclear arsenals. American policy laid important and long lasting impressions. Isolation has likely slowed down arms buildup in the sub-continent. However, no state can ignore the 32 years (1974-2006) of penalty that India has endured because of its decision to develop and possess nuclear weapons. But during this period it has become clear that denuclearizing India was an unachievable objective perhaps even with a cohesive effort by the U.S., the International Atomic Energy Agency (IAEA), and the Nuclear Suppliers Group (NSG) [1].

This dissertation study follows the style of the *Journal of Nuclear Science and Engineering*.

The Pokhran-II nuclear tests in 1998 convinced the U.S. that India would never formally and unilaterally restrict or relinquish its nuclear arsenals. This scenario drove the desire to work with India on a broader, strategic level through civilian nuclear cooperation. The cornerstone of any civilian nuclear technological support necessitates the separation of military and civilian facilities [2, 3]. The degree of entanglement of the military and civilian facilities can only be assessed by a full-scale microscopic view of the complete nuclear fuel cycle of India. Assessment of uranium reserves, plutonium production, energy generation, technology available and projects attempted facilitates the dual objectives of strengthening bilateral ties and preventing the spread of nuclear weapons through stronger cooperation.

I.B. Objectives of the Study

A great deal of speculation has recently occurred with regards to the Indian fuel cycle (both military and civil) since the initial agreement for nuclear cooperation between the U.S. and India was made on July 18, 2005. Much of this speculation is moved by a misunderstanding of the technical details of the Indian fuel cycle and Indian nuclear facilities; however, some speculation is also a product of uncertainties in the status and disposition of various Indian facilities. The overall objective of the work presented here was to analyze and document the Indian fuel cycle especially with respect to its relevance to the U.S.-India Nuclear Cooperation Agreement of 2005. This study would provide an unbiased and complete resource to policy-makers and decision-makers in regard to that agreement as well as in future cooperation with India. It may also

educate the public on the status of the Indian fuel cycle. The specific goals of this work were as follows:

1. Provide a complete description of all Indian nuclear facilities (including decommissioned facilities, currently operating facilities, facilities under construction, and facilities planned).

2. Provide an historical assessment of the Indian fuel cycle and material production in India since inception to the present day (specifically December 31, 2006). This includes a determination of the present reserves of all nuclear materials in India.

3. Provide a detailed technical analysis of the future status of the Indian fuel cycle and material production if the U.S.-India Cooperation Agreement is not placed in effect.

4. Provide a detailed technical analysis of the future status of the Indian fuel cycle and material production if the U.S.-India Cooperation Agreement is placed in effect.

5. Perform a feasibility study of possible reactor systems that could facilitate the separation of the military and civilian nuclear facilities in India while increasing proliferation resistance and optimizing the use of domestic and international resources.

This work is focused on technical assessments of the Indian fuel cycle based on open source information of the Indian nuclear facilities and the usage of those facilities. Assumptions and uncertainties included in any of the models used here are explicitly declared, whenever possible.

II. HISTORY OF THE INDIAN NUCLEAR PROGRAM

In this section, a brief history of the development of the nuclear power program, nuclear weapons program, and advanced research initiatives of India are given. This history was derived from numerous sources and an effort was taken to try to supply the most accurate details possible. Historical reporting is provided to both increase the awareness of the reader of the complexity of the Indian program as well as to set the stage for the technical assessments of the Indian fuel cycle. Assumptions and inferences derived have been explicitly stated whenever possible. Much of the subsequent material in this section is adapted from reference 17.

II.A. Beginning of Indian Nuclear Program: 1944 - 1960

The Indian nuclear program started even before India became an independent nation. Dr. Homi Jehangir Bhabha submitted a proposal in March 1944 to the Sir Dorab Tata Trust for the establishment of a nuclear research institute [4]. This led to the creation of the Tata Institute of Fundamental Research (TIFR) in April 1945. Soon TIFR started its initial work in Bangalore in June 1945 with Bhabha serving as the first director [5]. In December 1945, Bhabha moved TIFR to Bombay where it continues to serve as a research institution today [6]. On August 15, 1947, India attained independence from British rule. A year after that the Indian Atomic Energy Act was passed. This led to the establishment of the Indian Atomic Energy Commission (AEC) [7]. The AEC would pursue in-depth studies of nuclear energy and was comprised of three members: Dr. Bhabha, Dr. K.S. Krishnan, and Dr. S.S. Bhatnagar [8].

At a press conference in Madras (presently called Chennai), Prime Minister Nehru spoke about the virtues of atomic energy for national development by stating “We are interested in atomic energy for social purposes. Atomic energy represents a tremendous power. If this power can be utilized to produce electricity, it will be a tremendous boon to mankind, because it is likely to be more available and cheaper than the building of huge hydroelectric projects. Therefore, we are interested in the development from the social point of view.” A four-year plan was unveiled to develop India's nuclear infrastructure for nuclear material exploration and the application of nuclear energy in medicine. During the period Dr. Bhabha began seeking technical information on reactor theory, design, and technology from the U.S., Canada, and the U.K. while negotiating the sale or trade of raw materials such as monazite and beryllium-containing ore [9].

In August 1950, Indian Rare Earths Limited (IRE) was established to recover minerals and process rare earths compounds and thorium-uranium concentrates. Later in April 1951, uranium deposits were discovered at Jaduguda and drilling operations commenced in December 1951 [10]. The Jaduguda mine was the main source of uranium for the entire Indian nuclear program until the present day.

In 1954, significant changes occurred which led to a definite path for establishing a nuclear weapons capability. On January 3, 1954, the Atomic Energy Establishment at Trombay (AEET) was created by the AEC. AEET led research on nuclear weapons technology and has been referred to as the “Indian Los Alamos”. The AEET was formally inaugurated on January 20, 1957 and was followed by the creation of the

Department of Atomic Energy (DAE) on August 3, 1954 with Dr. Bhabha as Secretary [11]. The DAE was not under the regular control of the cabinet but reported directly to the Prime Minister. On January 12, 1967 in tribute to Dr. Bhabha, who died in an airplane crash on January 24, 1966, the AEET was renamed as the Bhabha Atomic Research Center (BARC). During his tenure, Dr. Bhabha also transferred all the scientific initiatives from TIFR to AEET.

On May 10, 1954, Prime Minister Nehru sharply reacted to President Eisenhower's Atoms for Peace plan. His view was that "atomic energy for peaceful purposes is far more important for a country like India, whose power resources are limited, than for a country like France, an industrially advanced country" [12]. Following this ideology, Bhabha and Bhatnagar looked to the British for assistance in constructing a nuclear reactor and in converting uranium ores into metal for fabrication into fuel. Five tons of heavy water for use in a planned Indian research reactor was requested. The British encouraged Bhabha to approach the Atomic Energy of Canada Limited (AECL) for the supply, since the U.K. had a deficit of heavy water available for domestic use.

On November 26, 1954, Prime Minister Nehru stated at the "Development of Atomic Energy for Peaceful Purposes" conference that "atomic energy is a tremendous tool for the benefit of humanity, whether it is disease or poverty. It therefore becomes necessary for us to try not to lag behind in this, although we may not have the great resources that some other countries have". At the same conference, Bhabha presented the three-stage nuclear energy plan for national development (shown in Figure 1). Under

this plan, India would start its nuclear power in the first-stage with natural uranium-fueled, heavy-water moderated Pressurized Heavy Water Reactors (PHWRs) reactors to produce power and plutonium. The first-stage reactors would be based on the CANDU technology and would be built with Canadian assistance. These reactors would produce 420 GWe-yrs of electricity [13]. In the second-stage, plutonium would be separated from the spent fuel from the first-stage reactors and used to power fast breeder reactors (FBRs). The FBRs would then generate an additional 54,000 GWe-yrs of electricity. Thorium would be irradiated in the blanket of the FBRs to breed ^{233}U. In the third-stage, the ^{233}U bred from the second-stage would serve as fuel for the ^{233}U breeder reactors [14, 15]. These ^{233}U breeder reactors would provide 358,000 GWe-yrs of electricity and breed more fissile materials. Breeder reactors are an advanced reactor design in which the reactor breeds additional fuel during operation. Technically, a breeder reactor produces more fuel during operation than it consumes. The breeder reactor achieves this through a design which highly conserves neutrons in the system. These reactors would produce enough excess material to fuel themselves and produce excess for weapons use [16]. The Indian government formally adopted this three-stage plan in 1958 recognizing the importance of nuclear power as a sustainable energy source.

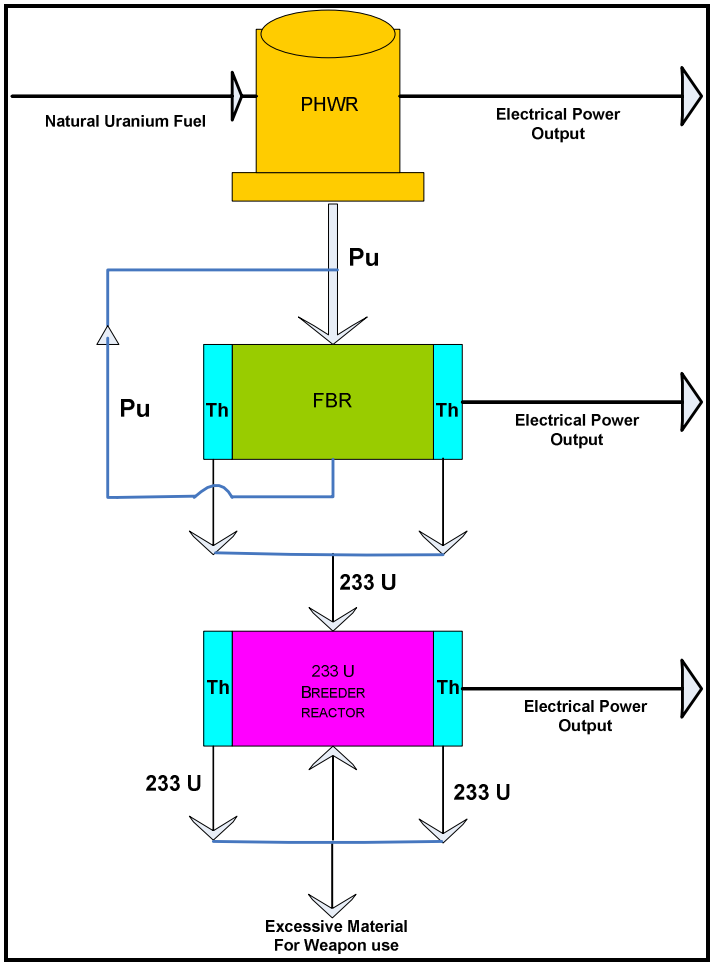

Fig. 1. India's three-stage nuclear power production strategy

The basis of the three-stage-program was indigenously available technology for production of natural uranium, vast reserves of thorium, and the mastering of heavy water production and spent fuel reprocessing technology. When this program was devised, India did not have any existing power reactors and there were no commercial fast breeder reactor systems anywhere in the world.

On the heels of Bhabha's nuclear power strategy, the atomic energy program in India grew rapidly. The budget for research in atomic energy grew by a factor of twelve

from 1954 to 1956 [17]. By 1959, the DAE consumed almost one third of India's research budget and the AEET employed over 1,000 scientists and engineers.

In 1955, the 1 MW_{th} APSARA research reactor was built with British assistance. APSARA went critical on August 4, 1956. APSARA was a light-water cooled and moderated swimming pool-type research reactor. It uses low-enriched uranium, plate-type fuel. The British-origin fuel for the reactor is safeguarded but the reactor itself is not under IAEA safeguards [18].

During most of 1955, India was an active negotiator with the U.S. and Canada to acquire a research reactor with plutonium production capability. These negotiations began when members of the U.S. Joint Committee on Atomic Energy (a part of the Atoms for Peace Program) visited India in early 1955 to promote the expansion of the peaceful applications of atomic energy. In September 1955, Canada agreed to supply India with a 40 MW_{th} research reactor. Canada also offered to pay all the foreign exchange costs of building the reactor. No strict safeguards on the use of the plutonium produced by the reactor were made (other than the commitment by India, via an annex to the agreement, that the reactor and fissile materials it produced would be used only for peaceful purposes). The reactor was named the Canada-India Reactor (CIR) [19].

On March 16, 1956, a contract was signed by the U.S. and India for supply of 18.9 tons of heavy water for the reactor. This agreement was based on the possibility that India's own Nangal heavy water plant may fail to operate at the required capacity factor [19, 20]. The agreement stated that "the heavy water sold here under shall be for use only in India by the government in connection with research and the use of atomic

energy for peaceful purpose…" [21]. Following this, the reactor was dubbed the Canada-India Reactor, United States (CIRUS).

The CIRUS reactor was based on the Canadian NRX reactor design and uses a heavy water moderator and natural uranium fuel. The establishment of CIRUS was a landmark event for both nuclear energy initiatives in India as well as nuclear proliferation. Although the agreement required that the reactor only be used for peaceful purposes, it occurred before any international policy regulations were in place for atomic technology transfers and desirably no provision for inspections were made. India refused to accept fuel from Canada and carefully avoided effective regulation for the reactor. India established a program to manufacture the natural uranium fuel for CIRUS indigenously so as to keep complete control of the plutonium produced from it. This program eventually succeeded in developing the techniques for producing the precisely manufactured, high-purity material demanded by the reactor. On February 19, 1960, ten fuel elements for the first load of CIRUS at Trombay were fabricated. CIRUS achieved criticality at AEET on July 10, 1960 [22].

CIRUS was well designed for producing weapons-grade plutonium. Weapons-grade plutonium contains a low percentage of ^{240}Pu and a high percentage of ^{239}Pu. Generally speaking, weapons-grade plutonium has a ^{240}Pu / ^{239}Pu ratio of approximately 6%. Since the PHWR uses natural uranium fuel, it achieves relatively low-burnup on the fuel. This low-burnup leads to plutonium production of a fairly high grade [22]. Using simulations of CIRUS with ORIGEN2, it is estimated that the reactor is capable of

producing enough plutonium for one to two weapons per year (approximately 5-10 kg of plutonium per year) [23].

In July 1958, the PHOENIX project was authorized in India. The project was to build a reprocessing facility with a capacity for processing 20 tons of spent fuel per year. Reprocessing facilities are used to chemically separate the plutonium from the uranium and fission products in spent fuel. Their capacity is generally listed in terms of the mass of spent fuel that can be processed annually in the plant. The quantity of plutonium then produced would depend upon the quantity of plutonium in each mass of spent fuel. The capacity of PHOENIX was designed to match the production capacity of CIRUS. The PHOENIX plant was based on the U.S. developed PUREX process and was commissioned in mid-1964 [19].

As a nation, India has always placed a premium on self-sufficiency. Due to its vast domestic resources of thorium but limited supplies of uranium, India has always placed strong emphasis on the development of breeder reactor fuel cycles. Breeder reactors require highly concentrated fissionable material for reactor fuel: either highly enriched uranium or plutonium. This provided a peaceful rationale for developing a plutonium separation capability, but the principal impetus for India's first spent fuel reprocessing plant was to obtain a nuclear weapons option [17].

II.B. Early Weapons Development Effort: 1960 - 1966

The Indian political and scientific community became aware of China's nuclear weapons program by 1961. This reinforced India's effort towards weaponization.

Bhabha initiated preliminary studies on weapon physics using the material support by the CIRUS reactor and the Trombay reprocessing plant. A physics group was established in January 1962 to study the design of implosion weapons. This group worked in secrecy and reported directly to Bhabha [17]. The humiliating defeat by China in a border-dispute war in 1962 led to a formal demand for the development of nuclear weapons in Parliament. Bhabha lobbied hard for matching the Chinese power and even asked Prime Minister Nehru to authorize a nuclear test in Ladhak on the Chinese border. This ambitious demand however was not supported by the existing nuclear infrastructure. CIRUS did not reach full power until October 16, 1963 and the PHOENIX plant was officially inaugurated on January 22, 1963 but produced very little plutonium over the years to come. Thus, India did not have the requisite plutonium for their first device until 1969 [22].

The Indian nuclear weapons program was advertised as a “peaceful nuclear explosives” (or PNE program). Bhabha and associated scientists pursued the PNE path as a means towards the acquisition of nuclear weapons. In the past, Bhabha acknowledged the fact that there is no difference between a PNE and a weapon. The nuclear weapons program continued under the guise of a PNE through the Pokhran–I test in 1974 and up until the Pokhran–II test in 1998. At that time, India finally acknowledged the actual objective of the weapon [22].

II.C. Negotiations and Aftermath of the Non-Proliferation Treaty: 1966 - 1974

India began as a strong advocate of the Non-Proliferation Treaty (NPT) but a turn of events made it a non-signatory of the final treaty. India insisted on universal disarmament with the non-existence of permanent nuclear powers. This fundamental stand lead India to vote against it on June 12, 1968.

During December 1968 to January 1969, a team of Indian scientists visited the Soviet Union. They were impressed by the plutonium-fueled, pulsed fast reactor as an excellent laboratory model for the study of the fission process. Following this, the PURNIMA (Plutonium Reactor for Neutron Investigation in Multiplying Assemblies) was designed. It was an hexagonal core of 177 stainless steel pencil-shaped rods containing 18 kg of plutonium in 21.6 kg of plutonium oxide pellets. PURNIMA went critical on May 18, 1972 after sufficient separated plutonium finally became available [24]. PURNIMA provided the test bed for understanding nuclear physics experiments. PURNIMA helped scientists develope facilities and gain experience in the handling of plutonium [17].

The nuclear weapons development program continued in 1970 and 1971. The basic design for India's first nuclear device was complete by early 1972. The important decision of assembling the device and preparing for the test was made later in 1972. The decision was heavily influenced by internal momentum and domestic politics. In order to not alert proliferation observers, Homi Sethna, head of BARC, separated the Indian space program from the DAE. This eliminated the DAE from developing both nuclear explosives and missile technology [22].

In 1970, the PHOENIX plant developed a serious leak and had to be shut down for maintenance. This limited the material available for a weapon. To provide the material for a test device, the PURNIMA reactor was shutdown in January 1973 and dismantled. PURNIMA contained 18 kg of plutonium and it is assumed that the test device required about 6 kgs of plutonium. Therefore, in 1974 there was only enough material for three devices. The test was named the "Smiling Buddha" and was successfully conducted on May 18, 1974. The material used in the device was produced originally by CIRUS but was first used in PURNIMA. In the aftermath of the test, the civilian nuclear power program struggled for the next three decades due to lack of domestic resources and its dependence on international technology which was unavailable because of India's nuclear isolation.

II.D. India's Isolation Following Operation Smiling Buddha: 1975 – 1998

International reaction to the "Smiling Buddha" was varied. The 1974 test sharply escalated international attention to nuclear proliferation, and international support for India's civilian nuclear program disappeared. Canada cut-off virtually all nuclear assistance four days after the test, bringing two nuclear power projects (the RAPS-II reactor and the Kota heavy water plant) to a halt. Following 1974 test, the U.S. congress passed amendments to the U.S. Atomic Energy Act of 1954 that prohibited U.S. involvement with India's nuclear sector. India had sufficient natural uranium to complete the RAPS-II reactor and meet startup and refueling needs; however, all future PHWR projects were seriously hindered.

In 1977 India started work on a larger 100 MW_{th} plutonium production reactor at Trombay. This reactor was named "DHRUVA". There were concerns over heavy water supplies for DHRUVA and these concerns also existed for the civilian reactors. MAPS-I suffered due to a lack of heavy water availability. On August 8, 1985, DHRUVA went critical but was soon crippled by serious problems requiring a shutdown. It was restarted in December 1986 and operated at 25% capacity factor from then until spring of 1987. DHRUVA finally achieved full power on January 17, 1988. After that time DHRUVA became the main supplier of plutonium for the Indian nuclear weapons program [25].

The arguments for developing the weapons can be assessed from a statement by then Prime Minister I.K. Gujral in a meeting with President Bill Clinton at the U.N. General Assembly session as "without it India could not expect to be admitted to the corridors of global power, nor enjoy the status of the dominant regional power; that the bomb might quicken the process of normalizing relations with China; that it would proclaim India's independence of the Soviet Union and compel the United States to change its attitude of hostility or benign neglect" [26].

II.E. The Pokhran-II Tests: 1998

India conducted a second series of nuclear weapon tests in May 1998. These tests were also conducted at the Pokhran facility and were referred to as the Pokhran-II tests. The Pokhran-II tests were conducted on the same day on which the 1974 test was conducted. There are conflicting reports on the claimed yields and design of these

devices [19]. Some reports claim that one of the devices used reactor-grade material which created a furor as this is presently and abundantly produced by CANDU reactors.

II.F. Following Pokhran-II Tests: 1999 - 2005

By 1998, India had 8 PHWR's of 220 MWe rating. The share of electricity generation that came from nuclear power had increased gradually but was well below expectations. It was nowhere close to the planned generation of 20,000 MWe by 2000. By March of 2006 India had an installed capacity of 3900 MWe from nuclear power plants. Nuclear energy comprised of 3% of the total installed capacity [27].

The plutonium from the spent fuel of production reactors was reprocessed for use in the Fast Breeder Test Reactor (FBTR) of 40 MW_{th}. This technological demonstration opened up the doors for establishment of Prototype Fast Breeder Reactor (PFBR) of 500 MWe [28].

A research reactor of 30 kW_{th} rating having ^{233}U as driver fuel and a BeO reflector was built and operated to demonstrate the feasibility of the third-stage. Furthermore, with this reactor a foundation to carry on research for development of commercial scale ^{233}U fueled reactors was established.

Immediately after the Pokhran-II tests, India was facing dual challenges of both international sanctions and diminishing uranium reserves from the flagship mine in Jaduguda. Prior to that date, the constraints on uranium fuel production were milling capabilities. After the Pokhran-II tests, the focal point shifted to the domestic uranium

ore reserves. Mining activities at many other sites were rigorously pursued. This was due the lower ore yield from these sites as well as due to political and social reasons.

II.G. Current State of Indian Nuclear Program

There are no official figures available of weapon stockpiles at any developmental stage of India's arsenal. Conclusions can only be inferred from considerations of India's ability to produce critical raw materials and production plants. India is likely to have an arsenal in excess of 100 weapons.

India's nuclear power program proceeds almost entirely without fuel or technological assistance from other countries. Partly as a result of this, its power reactors have been among the worst-performing in the world (at least with regard to capacity factors). This reflects the difficulties of technological isolation. However, the capacity factors are apparently now improving significantly and newer projects are constructed at a faster rate.

III. INDIAN NUCLEAR FACILITIES

The first nuclear power project of India started with General Electric constructing and commissioning two units of Boiling Water Reactor (BWR) power plants at Tarapur in 1969. Soon India realized the difficulty in acquiring enriched uranium for these reactor types and believed that BWR's would ensure lifetime dependence on the U.S. for fuel needs. India received the power plants at a meager cost of 80 million U.S. dollars. In 1969, the GE-built BWR's were the biggest electricity producing units in India and played a primary role in the development of Bombay (now called Mumbai), the economic capital of India. Even before India's first power plant at Tarapur, Homi Bhabha and his team were suggesting the three-stage-program discussed in section II and looked into the potential of CANDU-type reactors. This carried the burden of acquiring heavy water for moderation and as a coolant but using indigenous natural uranium fuel. Apparently the technology for production of this fuel existed in India.

Prior to 1998, there existed 12 nuclear power plants with 8 of them outside safeguards. The technology to build 220 MWe rated units was mastered and plans were laid to further the progress of nuclear power plants. Building of nuclear power reactors continued after the Pokhran-II tests with six more including two of 540 MWe ratings. There currently is a strong initiative to construct two power reactors per year. Capacity factors of the order of 80% were achieved in 2003 from newly commissioned power plants. Given the experience from the FBTR, construction started on two units of 500 MWe PFBR's. Old reprocessing plants were scaled up and new ones were built to meet the fuel needs of the breeder reactors.

India currently posses 15 operational PHWR, 3 PHWR are under construction and 4 planned PHWR power projects. India's nuclear power provides less than 2.5% of its electricity needs. Until the 540 MWe Tarapur-4 in 2005, India did not have a nuclear power producing unit greater than 220 MWe capacity. Table I lists all the existing Indian nuclear power plants, their capacity, type and date of commencement into service [29]. Table II lists the capacity factors for the existing nuclear power plants until 2003 and Table III extends the data to 2006 [30].

TABLE I List of Nuclear Power Plants of India

POWER PLANT	TYPE	RATED POWER	DATE OF COMMENCEMENT
Tarapur Atomic Power Station-1 (TAPS-1)	BWR	210 MWe	28 October 1969
Tarapur Atomic Power Station-2 (TAPS-2)	BWR	210 MWe	28 October 1969
Tarapur Atomic Power Station-3 (TAPS–3)	PHWR	540 MWe	January 2006
Tarapur Atomic Power Station-4 (TAPS-4)	PHWR	540 MWe	September 2005
Rajasthan Atomic Power Station-1 (RAPS-1)	PHWR	220 MWe	16 December 1973
Rajasthan Atomic Power Station-2 (RAPS-2)	PHWR	220 MWe	1 April 1981
Rajasthan Atomic Power Station-3 (RAPS–3)	PHWR	220 MWe	2000
Rajasthan Atomic Power Station-4 (RAPS-4)	PHWR	220 MWe	2000

TABLE I Continued			
Rajasthan Atomic Power Station-5 (RAPS–5)	PHWR	450 MWe	Aug 2007
Rajasthan Atomic Power Station-6 (RAPS-6)	PHWR	450 MWe	Sept 2008
Rajasthan Atomic Power Station-7 (RAPS–7)	PHWR	500 MWe	Con. Planned for 2011
Rajasthan Atomic Power Station-8 (RAPS–8)	PHWR	500 MWe	Con. Planned for 2011
Madras Atomic Power Station-1 (MAPS-1)	PHWR	220 MWe	27 January 1984
Madras Atomic Power Station-2 (MAPS-2)	PHWR	220 MWe	21 March 1986
Narora Atomic Power Station-1 (NAPS-1)	PHWR	220 MWe	1 January 1991
Narora Atomic Power Station-2 (NAPS-2)	PHWR	220 MWe	1 July 1992
Kakrapar Atomic Power Station-1 (KAPS-1)	PHWR	220 MWe	6 May 1993
Kakrapar Atomic Power Station-2 (KAPS-2)	PHWR	220 MWe	1 September 1995
Kaiga Atomic Power Station-1 (KAIGA-1)	PHWR	220 MWe	2000
Kaiga Atomic Power Station-2 (KAIGA-2)	PHWR	220 MWe	1999
Kaiga Atomic Power Station-3 (KAIGA–3)	PHWR	220 MWe	Mar 2007
Kaiga Atomic Power Station-4 (KAIGA-4)	PHWR	220 MWe	Sept 2007
Kaiga Atomic Power Station-5 (KAIGA–5)	PHWR	220 MWe	Con. Planned for 2007
Kaiga Atomic Power Station-6 (KAIGA–6)	PHWR	220 MWe	Con. Planned for 2007

TABLE I Continued			
Kundankulam Power Plant-1 (KK–1)	VVER	1 GWe	Dec 2007
Kundankulam Power Plant-2 (KK-2)	VVER	1 GWe	Dec 2008
Prototype Fast Breeder Reactor-1 (PFBR-1)	FBR	500 MWe	2010
Prototype Fast Breeder Reactor-1 (PFBR-2)	FBR	500 MWe	2010

India has a number of research and production reactors. No nuclear reactor, or for that matter any technological device, was more controversial in its lifetime than the CIRUS reactor. It provided the desired plutonium for the first nuclear test of India. India, being alienated from the advanced nuclear technology that it badly needed, was faced with technological hurdles in addition to the lower than desired domestic reserves of uranium. Moreover the indigenously built nuclear power plants were of lower rating compared to those being built by the developed nations and operated at low capacity factor. Over a period of time, India obtained self sufficiency in PHWR technology but until recently all the nuclear power plants were rated at 220 MWe and operated at a low capacity factor (52%) to maintain the fuel reserves.

To advance research on the development of the second-stage power reactor systems, a Fast Breeder Test Reactor (FBTR) was built based on the French Rapsodie design. This was a 40 MW_{th} fast reactor with a mix of plutonium and uranium carbide as fuel. The design and operation experience obtained from it enabled the construction of

the 500 MWe Prototype Fast Breeder Reactor (PFBR). Conversion of thorium to ^{233}U in the blanket of the fast breeder provides second-stage of the Bhabha's vision in section II and lays the foundation for the third-stage.

Technological demonstration of ^{233}U based reactor was done with commissioning and operation of the 30 kW KAMINI reactor. However commercial scale systems have yet to be demonstrated.

TABLE II Capacity Factors of PHWRs Until 2003

POWER PLANT	CAPACITY FACTOR	YEAR OF CRITICALITY
RAPS-1	23.31	1972
RAPS-2	52.65	1980
MAPS-1	52.82	1983
MAPS-2	52.92	1985
NAPS-1	60.62	1989
NAPS-2	67.82	1991
KAPS-1	70.91	1992
KAPS-2	84.14	1995
KAIGA-1	80.7	2000
KAIGA-2	80.91	1999
RAPS-3	77.98	2000
RAPS-4	79.2	2000

TABLE III Capacity Factors of PHWRs From 2004 to 2006

POWER PLANT	CAPACITY FACTOR / YEAR	YEAR OF CRITICALITY
All 12 Plants Stated in Table II	81% / 2004	Operating
TAPP–4 + 12 Plants	76% / 2005	TAPP-4 Critical on 09/2005
TAPP–3 & 4 + 12 Plants	52.4% / 2006	TAPP-3 Critical on 01/2006

TABLE IV Research Reactors

NAME	LOCATION	TYPE	START DATE	FUNCTION
CIRUS	Trombay	40 MW_{th} HWR Research Reactor	10 July 1960	Weapons-Grade Plutonium
DHRUVA	Trombay	100 MW_{th} HWR Research Reactor	10 August 1985	Weapons-Grade Plutonium
APSARA	Trombay	1 MW_{th} LWR	1956	Research
PURNIMA – 1	Trombay	Critical Assembly	1989	Decommissioned
PURNIMA – 2	Trombay	LWR	1984	Decommissioned

TABLE IV Continued

NAME	LOCATION	TYPE	START DATE	FUNCTION
PURNIMA – 3	Trombay	LWR	1994	Uses ^{233}U
Zerlina	Trombay	PHWR	1961	Decommissioned
Compact High Temperature Reactor	Trombay	0.1 MW_{th} Small Reactor	2010	Will use U-Th and ^{233}U to Produce Hydrogen
KAMINI	Kalpakkam	0.03 MW_{th} Test Reactor	1996	Uses ^{233}U
Andhra University	Vishakapatnam	0.1 MW_{th} Low Power Reactor	Unknown	Planned Research
FBTR	Kalpakkam	40 MW_{th} Fast Breeder Test Reactor	1998	Research and Development of Prototype Fast Breeder

India constructed additional weapons-grade-plutonium-producing reactors and numerous other research reactors for neutronic studies. Table IV lists all the research reactors, including their capacity, type, date of commencement into service and their function [29]. The 40 MW_{th} thermal FBTR at Indira Gandhi Center for Atomic Research (IGCAR) was made operational in 1985 with a plutonium-uranium mixed carbide fuel. This facility acted as the springboard for the fast reactor and breeder technology for India. On the basis of the research and experience from FBTR a unit of 500 MWe PFBR

is being constructed and expected to be in commercial operation by 2012 [28]. India has not declared these as civil facilities under the separation plan pursuant to the U.S.-India nuclear cooperation agreement. Not much is known about the core and fuel design. Over the years, India attained vast expertise in fuel fabrication, reactor operation, plutonium availability and thorium utilization.

India's large scale uranium enrichment endeavor started in the early 90's. The purpose of uranium enrichment in India is speculated to be for the core of a nuclear submarine, the primary of a thermonuclear device or to provide fuel for the fast breeder reactors. Table V lists the enrichment facilities, including their location, types of technology and date of commencement into service [29]. Of these facilities, only the Mysore facility is a large-scale facility. In estimating the enriched uranium accumulation for the Mysore facility, P1 centrifuges of 3 SWU/yr capacities with a total plant load of 2000 SWU/yr was assumed. There were reports claiming failure of this project because of its inability to produce weapons-grade enriched uranium [31].

TABLE V Enrichment Facilities

NAME	LOCATION	TYPE	START DATE	FUNCTION
Center for Advanced Technology	Indore	Laser Enrichment	1993	Research

TABLE V Continued				
NAME	**LOCATION**	**TYPE**	**START DATE**	**FUNCTION**
Rare Materials Project	Mysore	Centrifugal	1991	Uranium Enrichment
Laser Enrichment Plant	Trombay	Laser Enrichment	1993	Research
Uranium Enrichment Plant	Trombay	Pilot Scale Ultracentrifuge	1985	Research & Development

TABLE VI Heavy Water Production Plants

NAME / LOCATION	START DATE
Baroda	1980
Hazira	1991
Kota	1981
Manuguru	1991
Nangal	1962
Talchar	1985
Thal - Vaishet	1991
Trombay	Unknown
Tuticorin	1978

Table VI lists the heavy water production units with their location and start date for each facility [29]. The heavy water production technology of India is outdated and large scale renovation work, with international collaborations is needed.

All fuel fabrication facilities are listed in Table VII. The Nuclear Fuel Complex (NFC) at Hyderabad is the only large scale CANDU fuel fabrication facility in India. The NFC has an annual handling capacity of 250 tons of UF_6 and the estimated capacity as input to the plant is 216 tons of UF_6, after losses. In 2006, the NFC raised it's capacity from 250 to 600 tHM per year. The higher capacity though can only cater to the need of 14 PHWR's operating at 90% capacity factor. Any further addition of PHWR's would require additional capacity. Presently the NFC production capacity is not a restricton to operation because of the lower operational capacity factor of the power plants due to reduced uranium reserves. The scenario may change with additional mines or international uranium derived from the pending cooperation agreement.

TABLE VII Fuel Fabrication Facility

NAME	LOCATION	TYPE	CAPACITY tHM / YR	START DATE	FUNCTION
Enriched Fuel fabrication Plant	Hyderabad	BWR	25	1974	LWR Fuel Assemblies (Safeguarded)
Advanced Fuel fabrication facility (AFFF)	Tarapur	Unknown	20	1990	MOX Fuel for BWR, PFBR, PHWR & Research & Development

TABLE VII Continued					
NAME	**LOCATION**	**TYPE**	**CAPACITY tHM / YR**	**START DATE**	**FUNCTION**
Nuclear Fuel Complex (NFC)	Hyderabad	PHWR	250	1971	PHWR Fuel Bundles
			600	2006	
MOX Breeder Fuel Fabrication	Kalpakkam	Pilot Scale	Unknown	Unknown	MOX Fuel

Table VIII lists all of the reprocessing facilities in India [29]. These facilities are necessary for both weapons-grade material production and for the second stage of the Bhabha three-stage program. In 1992, two 100 tHM/yr reprocessing capabilities were added to the fuel cycle. This further closes the gap between the first and second stages of the Bhabha plan, by meeting the fuel needs of the PFBRs.

TABLE VIII Reprocessing Plants

NAME	**LOCATION**	**TYPE**	**tHM/YR**	**START DATE**	**FUNCTION**
Power reactor Fuel Reprocessing Plant (PREFRE)	Tarapur	PUREX	100	1977	Reprocess CIRUS, DHRUVA & PHWR Fuel. Provide Fuel for FBTR & AFFF
			150	1991	

TABLE VIII Continued

NAME	LOCATION	TYPE	tHM/YR	START DATE	FUNCTION
Kalpakkam reprocessing Plant (KARP)	Kalpakkam	PUREX	100	1997	Reprocess MAPS & FBTR Fuel. Provide Fuel for PFBR
Fast Reactor Fuel Reprocessing Plant (FRFRP)	Kalpakkam	Full Scale	Unknown	Future	Reprocess FBTR Fuel. May provide Fuel for PFBR
Lead Minicell Facility	Kalpakkam	Demonstration	Unknown	2003	Reprocess FBTR & PFBR Fuel in Future
Plutonium Reprocessing Plant	Trombay	PUREX	30	1964 – 74	Reprocess CIRUS, DHRUVA Fuel for weapon-grade Plutonium
			50	1984 Cont	

Exploration of uranium ores in India started as early as 1967. Around that time the Nuclear Non-Proliferation Treaty (NPT) was being formulated and steps were taken for non proliferation of nuclear materials and technology. Uranium mining was initiated with governmental backing. Beginning with Jaduguda (located in the eastern part of India), six to seven different locations were discovered over a period of time. To

estimate the ore capacity with grade, open source information, unclassified official data, IAEA red book data statements, budgetary allocations and news reports by independent study groups were referred to. Time periods that are better representative of ore exploration activities were given in Table IX [32].

TABLE IX Uranium Ore Details

TIME FRAME	LOCATION OF THE MINE	ORE CAPACITY (MEGATONS/ YEAR)	ORE GRADE (%)
1967 - 1998	Jaduguda	141.25	0.06
1986	Bhatin	5.475	0.04
1995 - 1998	Narwapahar	3.65	0.05
1999 Onwards		7.3	
1999 Onwards	Jaduguda	10	0.04

The Jaduguda mine located in the Singhbum East District of Bihar began operations in 1967. Although it has the capacity to deliver 200 megatons of ore yearly, its actual production has been 115 megatons per year averaged over a period of 40 years (1967-2006). For 32 years (1967-1998), the ore excavation was at the rate of 141.25 megatons of ore per year which dropped to an average of 10 megatons of ore per year in 1999-2006 [32]. Uranium Corporation of India Limited (UCIL) plans to extend the existing mine to exploit deposits below 555 meters and are estimated to extend to 905

meters. In November 1998, the U.S. imposed sanctions on the Jaduguda facility due to suspected nuclear weapons-related activities.

The Bhatin mine is much smaller than the Jaduguda mine. It shares the majority of its infrastructure with Jaduguda. The Bhatin mine has been in operation since 1986-87 and currently produces 5.475 megatons per year. The Narwapahar uranium mine and mill became fully operational in 1999-2000. Considered one of the most modern mines in India, it has the capacity to produce 7.3 megatons of ore per year [32].

Mining is followed by milling for recovering U_3O_8 from the ore. There is on average a 15% loss in processing activity (ore to U_3O_8) and 20% losses in conversion from U_3O_8 to UF_6. The UF_6 is sent to the NFC for fabricating into uranium oxide pellets for PHWR fuel bundles. UF_6 is also used as a feed for the uranium enrichment plant. As calculated using the assumptions, Mysore enrichment plant needs a feed of 2.15 tons of UF_6 per year for producing 10 kilograms of 90% ^{235}U and the Trombay plant uses 0.43 tons of UF_6 every year to produce 2 kilograms of 90% ^{235}U. The calculations of the feed capacity for the enrichment plant and product details over a year basis are given in section IV. The enriched uranium production with the actual feed requirements, SWU capacity, and centrifuges of a particular design are shown in Figures 19 and 20 of Appendix A. These are based on the assumption of full operational efficiency over an entire year.

There are prospective mining projects planned including the Mohuldih project to be commissioned in 2010, the Baghjanta project expected to be in service by 2008, the Bandugurang project (expected by end of 2007), the Lambapur-Peddagattu project

(which is speculated at 5900 tons of uranium at 0.044% ore-grade), the Pulivendula project, the Domiasiat project, the Gogi project and the Turamdih project [33].

IV. FUEL CYCLE ANALYSIS TO PRESENT DAY

A complete fuel-cycle assessment of India is presented in this section, using a flowsheet description of the materials, facilities and technologies in India. The results with explanations for reproducibility of the analysis for applications in assessment of fuel cycle of India on an extended period in future are emphasized in this study. The fuel cycle assessment accounts for the significant milestones in the Indian timeline of 1974 (first nuclear explosion, Pokhran-I test), 1998 (Pokhran-II tests), and 2006 (pursued U.S.-India Nuclear Cooperation Agreement). The data collected and synthesized for depiction in the flowsheet is representative of the consolidated contribution to the fuel cycle, within periods designated by those milestones. Through the method of timeline depiction, material accounting, electricity generation, and technologies adopted are depicted in the corresponding flowsheets. The policy approach, with regard to the operation of facilities and their priorities, is understood and interpreted by describing and analyzing the timeline data in sequential topical descriptions of materials produced and consumed. The topical descriptions of material produced and consumed have been arranged in this section on the basis of their relevance in the time frame of the flowsheet representations. The blocks in Figure 2 flowsheet with solely military applications have been represented in red, civilian in yellow and the dual use facilities were shown in blue. The dual-use nature of the nuclear program is clearly evident and analyzed in this section, from the mass flow in Figures 2, 3 and 4.

IV.A. Assessment of Plutonium Production and Use: Beginning till 1974

Plutonium production in India began with the CIRUS reactor, which has been discussed in detail in the preceeding two sections. In flowsheet assessment until 1974 (Figure 2, red blocks near top), the CIRUS reactor fuel was analyzed by burnup and depletion code ORIGEN2.2. The input deck was simulated at a specific power of 32 watts/gm for 1 ton of natural uranium fuel in CANDU library. The equally placed burnup steps reached defined limit of weapons-grade plutonium (^{240}Pu / ^{239}Pu <= 6%) at 1066 MWd/tU for CIRUS core. The CIRUS reactor was the source of depleted uranium fuel for RAPS-I reactor and also provided weapons-grade plutonium to the first nuclear weapon test of India. In this mode the commitment to peaceful use and intention for weapons development clearly evident, by the dual nature of use of the nuclear technology that India pursued.

In the beginning (lower left of Figure 2) the 220 MWe rated RAPS-I reactor was made critical with 40 tons of natural uranium and 17.65 tons of depleted uranium fuel of low burnup (1066 MWd/tU), as obtained from the spent fuel of the CIRUS reactor and reprocessed in PHOENIX plant. The spent fuel reprocessing for use in RAPS-I core only involved removal of fission product. The use of the depleted uranium instead of all fresh natural uranium fuel was to diminish the effect of plutonium peak.

After the first criticality of the CANDU reactor with natural uranium fuel, plutonium production from ^{238}U begins. Plutonium is produced at a higher rate than that being fissioned. This excessive plutonium accumulation decreases the delayed neutron fraction of the fuel (delayed neutron fraction of ^{235}U fuel ~0.0065 and plutonium fuel

~0.0026). To respond to the lowered reaction time available for the reactor regulation system in case of a transient the core power is reduced and so operated until plutonium levels stabilize. Now by having depleted uranium bundles in selective channels the fissile content is not excessive due to the plutonium production so the reactor regulation system can control the core power transients with the reactor operating at full power.

The quantity of the depleted uranium required in RAPS-I was computed by 2D lattice code HELIOS-1.4. To compute the exact quantity of depleted uranium fuel required, a basis of 92 mk excess reactivity was assumed. This was concluded after multiple iteration of the HELIOS input deck with incremental changes in fuel composition from fresh to a combination of depleted and fresh fuel. Any amount of excess reactivity less than that of 92 mk led to higher use of depleted uranium fuel in the core. If the plan of 20,000 MWe by 1986 as foreseen by Bhabha was to be achieved, then savings in the use of depleted uranium was an uncompromising necessity. If the plan were to be realized, then in 12 years (1974-1986) 9 power plants of 220 MWe rated were supposed to be built. The depleted uranium requirement for these reactors at the rate of 17.65 tons would be 158.85 tons. This need could have been barely met with reserves and annual production along with supplying the weapons initiative. Having more absorber rods is a challenging task in CANDU reactors given the compact size of the core. The neutron absorption can also be dealt with liquid poison but that necessitates higher reshuffling sequences to redistribute the fuel for avoiding local peaking due to plutonium production. The increase in reshuffling sequences is limited by the refueling

machine utilization to achieve the above. The assumptions and conclusions have been reached on the basis of reactor operator experience and knowledge of design details.

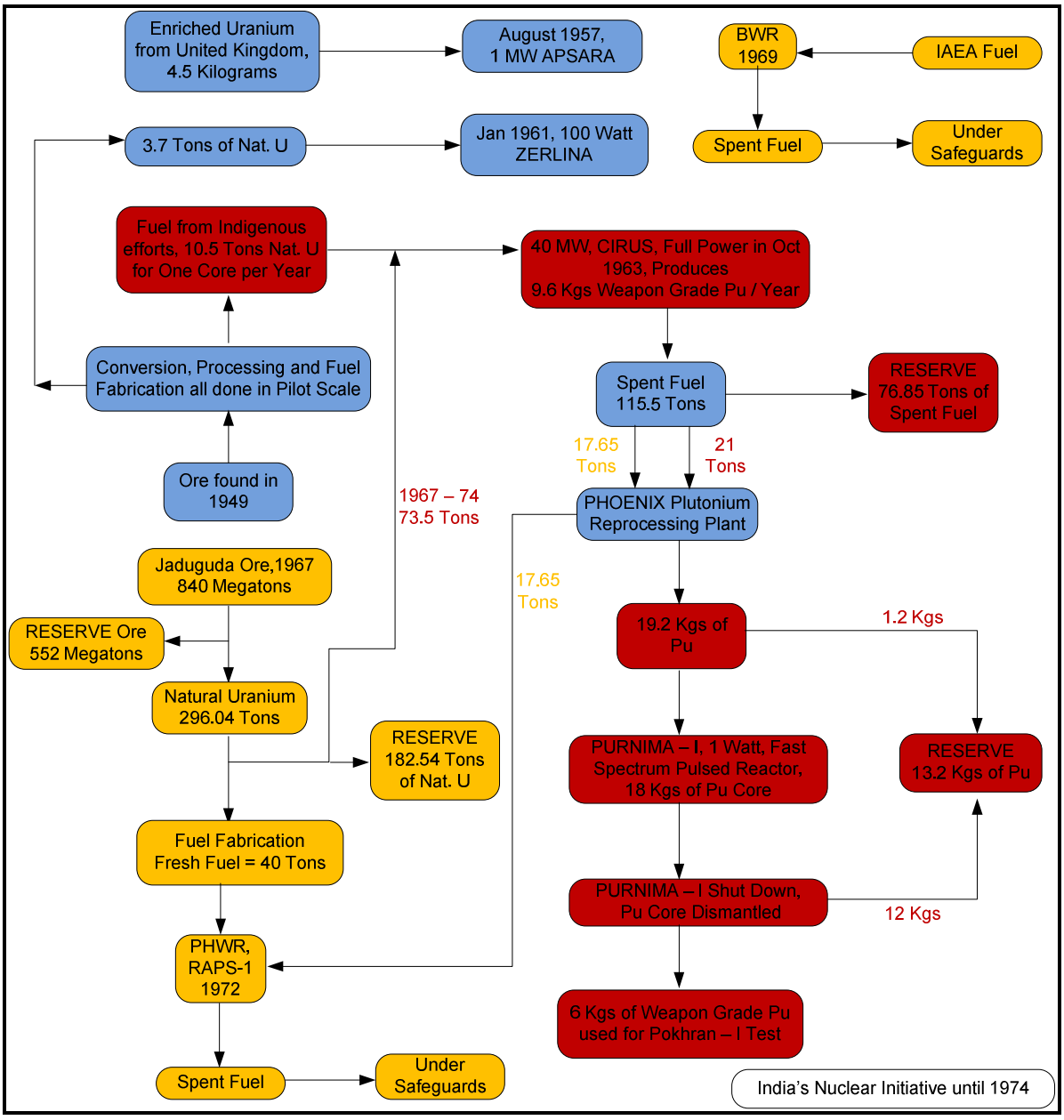

Fig. 2. Nuclear fuel cycle flowsheet until 1974

The flowsheet study shows that by 1974 only 13.2 kgs reserve of weapons-grade plutonium (shown in Figure 2 lower right) existed in India. Thus the analysis concludes that by the time Pokhran-I test was conducted India had plutonium to build only two more weapons. To calculate the quantity of plutonium produced in the time period an estimate is made of the amount of fuel irradiated. Usually the amount of fuel irradiated is based on the figures published for the amount of electricity generated in the case of power reactors. Since electricity production is not of relevance in these reactors, the capacity factor of 50% was calculated from the 10.5 tons of UO_2 being irradiated at 1066 MWd/tU burnup in CIRUS, to compute weapon-grade plutonium estimates. Weapons-grade plutonium is defined as that with a ratio of 6% or less for ^{240}Pu / ^{239}Pu. The lower percentage is considered because the spontaneous fission capability of ^{240}Pu would set of a chain reaction prematurely resulting in a fizzle, with perhaps only a few tons of energy being released as against its nominal value. The calculations that led to the capacity factor of 50% for CIRUS are stated in Appendix G.

CIRUS became critical on 10th of July 1960, but it started operating at full power only in October of 1963. By the end of September 1997, the reactor was shut down for refurbishment. In October 2000, it was still undergoing refurbishment. It is an old reactor with pneumatic control systems and requires an extensive period for refueling. As concluded from calculations in Appendix G an annual capacity factor of 50% is the maximum that can be reached in a year at full power operation of the core. From the output of ORIGEN2.2 depletion code for CIRUS reactor fuel, production of weapons-grade plutonium was computed at 280 days at full power which corresponds to the

desired burnup. This allows for 85 days in a year for shutdown to do maintenance, any refueling and make-up for lost hours of low capacity factor operation. As per the reference (24) CIRUS can irradiate 10.5 tons of natural uranium oxide fuel per year, which results in accumulation of 115.5 tons (center right of Figure 2) of depleted uranium in the 11 year period 1963-1974. It is thus computed by ORIGEN2 that annual production of weapons-grade plutonium would be 9.6 kilograms, with due allowance for unplanned shutdowns and regular maintenance. In order to be conservative on estimates, the fuel irradiation during the period when the two productions reactors had just been commissioned at the respective dates and were operating at very low capacity factor was ignored.

IV.B. Assessment of Plutonium Production and Use: Beginning till 1998

Following the pattern of study by simulating the depletion using the ORIGEN2.2 code it was found that DHRUVA has a much shorter cycle of 67 days with 6.35 tons of in-core fuel for producing 5.53 kilograms of weapons-grade plutonium. In a pragmatic assumption of five core changes per year, DHRUVA can produce 27.63 kilograms of weapons-grade plutonium per year.

DHRUVA became critical in August 1985, but had various operating problems, including vibration in fuel and oil leakage from the coolant pump. In December 1986, it began operating at 25 MW_{th}, increasing gradually to reach 100 MW_{th} in January 1988. Since then it is calculated that DHRUVA was operating at an average capacity factor of 75% (calculations shown in Appendix G). Burnup computations on the irradiated fuel

(central red text in Figure 3) show that total plutonium production of India by 1997 was 393 kilograms (center bottom in Figure 3 and Table X), after accounting for losses in reprocessing, corrected capacity factors and actual years of operation.

TABLE X Plutonium Produced and Uranium Used in Production Reactors

TIME FRAME	QUANTITY OF WEAPONS-GRADE PLUTONIUM PRODUCED (KILOGRAMS)	QUANTITY OF NATURAL URANIUM IRRADIATED (TONS)	
1964 – 1974	48.01	52.5	
1975 – 1997	345.30	120.75	
1964 – 1997	393.31	CIRUS	173.25
		DHRUVA	269.875
1964 – 2006	**633.50**	**CIRUS**	**204.75**
		DHRUVA	**485.775**
2006 – 2011	140.93	DHRUVA	107.95

In Figure 3 a consolidated assessment of India's fuel cycle from the beginning till Pokhran-II tests in 1998 is stated. DHRUVA's production capacity of approximately 28 kilograms of plutonium annually was determined using ORIGEN2.2 fuel burnup and depletion code [23]. Similarly to the preceding section the fuel was irradiated at a

specific power of 32 W/gU until burnup (1055 MWd/tU) required for qualifying as weapons-grade plutonium is attained. By 1998 two more new entrants the FBTR and Kalpakkam Mini (KAMINI) reactor were also demonstrated. As stated earlier FBTR laid the foundation for studies and experiments on fast breeders and paved the path for construction of PFBR. Pursuance of ^{233}U reactor KAMINI would materialize the third stage of the Bhabha's nuclear power program.

Using the facility and material data from section III, the spent fuel accumulated along with the fuel's characteristics computed using ORIGEN2.2 and HELIOS-1.4 lattice code for a 37-pincell CANDU fuel bundle was analyzed [34]. Multiplication factor over burnup, fuel isotopics, cross-sections and delayed neutron fraction was executed with one, two and 34 energy group neutrons. Adjusted cross-section library of 34 neutron and 18 photon energy groups was universally used over all the input decks.

The fuel characteristics were computed for fuelmaps of a pincell, a fuel bundle and for the complete assembly comprising of pressure tube (PT), calandria tube (CT), moderator and coolant in the control volume. The fuel isotopics data of 2 energy groups with cut-off at 0.623 eV had acceptable deviations from values of the full 34 energy group simulations. The final isotopics presented are of the assembly fuelmap, comprised of structural materials, moderator and coolant.

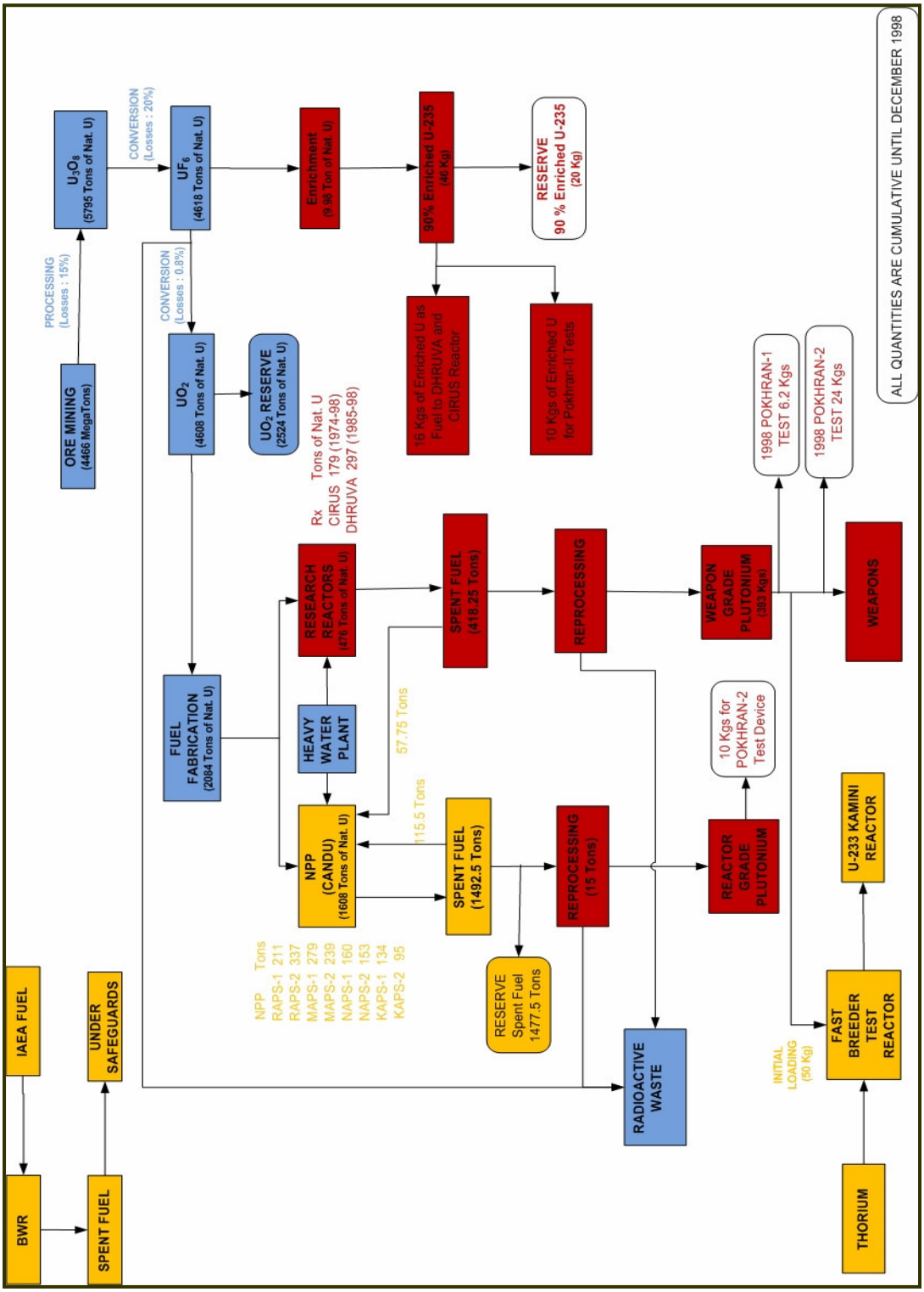

Fig. 3. Nuclear fuel cycle flowsheet beginning of fuel-cycle until 1998

Given the limited reprocessing capabilities of India, it is assumed that most of the spent fuel is in storage for future use. Assuming 6 kilograms of weapons-grade plutonium was used for Pokhran-I tests, 24 kilograms of weapons-grade plutonium was used for Pokhran-II tests, and 50 kilograms for FBTR core, India would have had enough weapons-grade plutonium in 1998 (center bottom in Figure 3) for at least 44 implosion devices (assuming 6 kilograms of weapons-grade plutonium required for a weapon).

IV.C. Assessment of Plutonium Production and Use: Beginning till 2006

In Table X time frames ending in 1974, 1997 and 2006 were chosen because of first implosion device test in 1974, one year prior to Pokhran-II tests and nuclear fuel cycle assessment and cooperation agreement initiation being 2006. The fourth row of Table X states the uranium used along with the 393 kilograms of plutonium produced by CIRUS and DHRUVA from 1963 till 1997 to show a comparison of the calculations against a press statement by Department of Energy of India, of 400 kilograms of plutonium being produced by 1997 in India. A projection until 2011 has also been shown for plutonium accumulation if CIRUS is decommissioned as per the separation plan and DHRUVA continues to operate at a capacity factor of 75%.

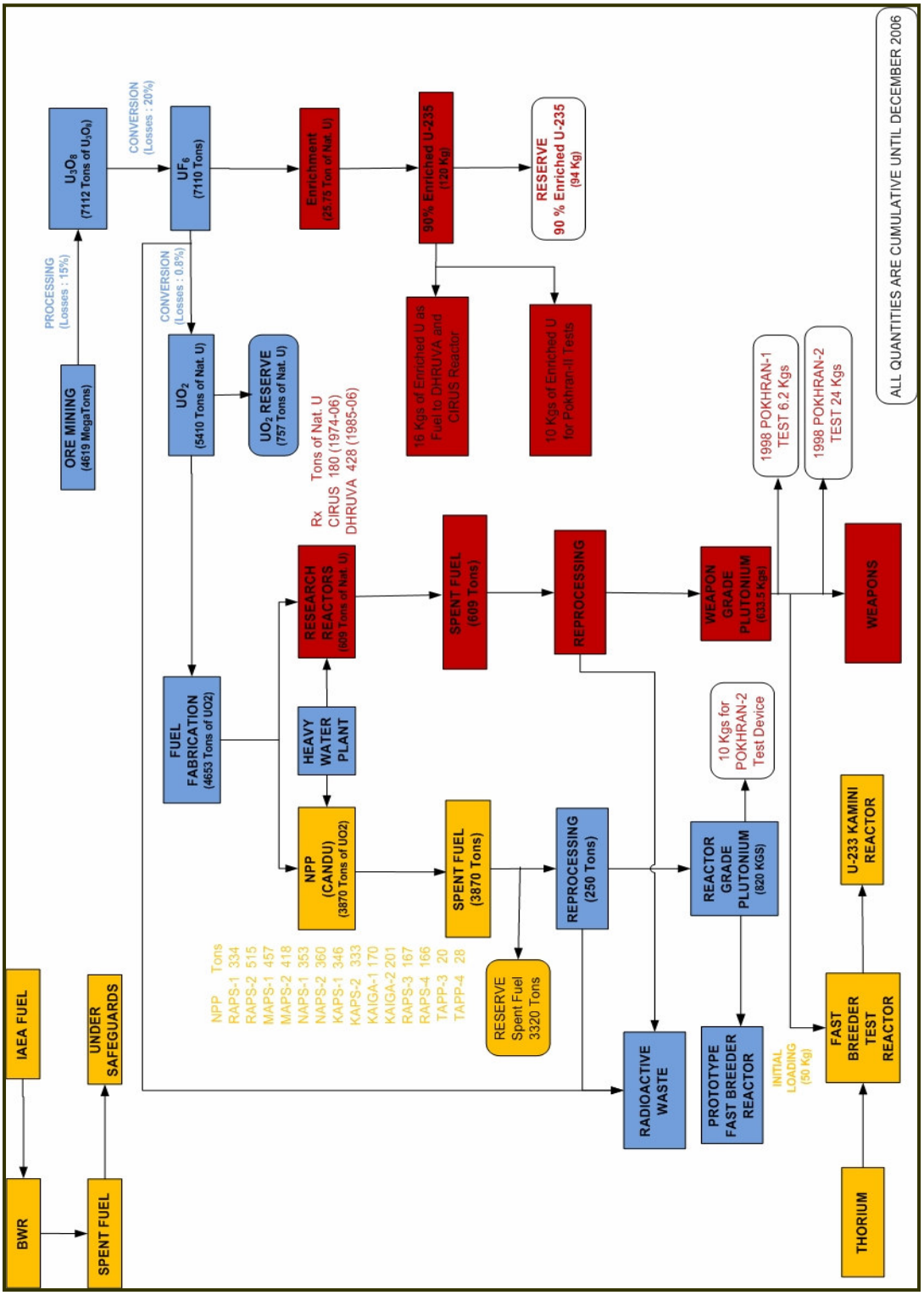

Fig. 4. Nuclear fuel cycle flowsheet until 2006

A consolidated fuel cycle flowsheet assessment for India was also performed from beginning of fuel cycle until 2006, using facility information from section III. The details of the assessment are shown in Figure 4. By December 2006 (center bottom in Figure 4), it is estimated that India would have accumulated 633.5 kilograms of plutonium. The reprocessing plant capacity of 250 tHM/yr (stated in Table VIII), operational for five years, is substantial for separating reactor-grade plutonium, to be used as driver fuel for the PFBR with an assumption of 850 kilograms of plutonium being required to fuel each core. With the assumption that the reprocessing plants are operational at full capacity factor since 2002 it can be concluded that by 2012 the plutonium would also be available for another PFBR as and when constructed. Thus at this rate of operation of the reprocessing plant and the availability of spent fuel, a fast breeder reactor (FBR) core can be added every five years, without taking the operational FBR doubling time into consideration, i.e., with the plutonium derived largely from PHWRs.

IV.D. Assessment of Natural Uranium Production

A flowsheet assessment of historical production of natural uranium in India was performed. These are shown in left center in Figure 2, upper right in Figure 3, and upper right in Figure 4 for the stated consolidated time frames. The calculations are specifically stated with assumed process losses in each stage of the materials. Recovery of U_3O_8 from the geological ore involves massive mechanical and chemical processes. Roasting and quenching after crushing of the ore is done to remove organics, followed

by leaching to scrap out the solid waste. Approximately 15% loss incurred in pulverizing the ore to finally recovering U_3O_8 has been assumed. U_3O_8 is later converted to UF_6 in a mill by hydro-flouration process. This chemical process leads to tailings containing heavy metal being dumped in to waste ponds. The product derived from the whole process is subject to 20% loss. UF_6 is then converted to UO_2 in a fuel fabrication plant to be used as reactor fuel. The fabrication losses are assumed to be of 0.8%.

TABLE XI Calculated Values for Annual Production of U_3O_8, UF_6 and UO_2

Time Frame	**Total Ore (MEGATONS / YEAR)**	**U_3O_8 (TONS / YEAR)**	**U_3O_8 [a] (TONS / YEAR)**	**UF_6 [b] (TONS / YEAR)**	**UO_2 [c] (TONS / YEAR)**
1967-86	141.25	254.25	216.11	216.83	164.99
1986-95	146.73	264.11	224.49	225.24	171.38
1995-98	154.03	277.25	235.66	236.44	179.91
1998-06	22.78	41.00	34.85	34.96	26.60

a. AFTER 15% LOSSES
b. AFTER 20% LOSSES
c. AFTER 0.8% FABRICATION LOSSES

Tables XI and XII state the quantity of materials produced per year over the time frames ending in 2006. The ore grades, historical uranium production rate and intermediate changes incurred in the production rates are explained in the text following

Table IX in section III. Actual uranium production at the dates corresponding to flowsheets were calculated and stated in Figures 2, 3 and 4. The ranges of the time periods shown in Tables XI and XII were chosen on the basis of new mine additions and changes in mine capacity of the present ones [32]. Table XII illustrates the historical production of all the three compounds of uranium. The scarcity of natural uranium reserves can also be concluded as one of the catalyst for the U.S.-India nuclear cooperation. Given the numerous prospective mining projects ongoing, the uranium production scenario might be completely different in the future. India's ability to sustain nuclear power projects using domestic reserves may need to be reconsidered if ore prospects are not realized.

TABLE XII Cumulative U_3O_8, UF_6 and UO_2 Production until 1998 and 2006

ENDING PERIOD	**U_3O_8 (TONS)** [d]	**UF_6 (TONS)** [e]	**UO_2 (TONS)** [f]
Until 1998	6834	6830 *	5197
Until 2006	7112	7110	5410

d. AFTER 15% LOSSES
e. AFTER 20% LOSSES
f. AFTER 0.8% FABRICATION LOSSES

* Feed to enrichment facilities was subtracted from the total UF_6 production as on 1998. Thus it can be assumed that the calculated quantity of the UO_2 is solely used for CANDU fuel fabrication.

IV.E. Assessment of Enriched Uranium Production

Enrichment facility was modeled with the assumptions stated in section II. It is concluded from the model that India could have accumulated 20 kgs of 90% ^{235}U (central right on Figure 3) by 1998 and 94 kilograms of 90% ^{235}U (central right on Figure 4) by the end of 2006 (assuming 10 kgs used in the Pokhran-II tests and 16 kgs used as experimental fuel in DHRUVA reactor). This amount of enriched uranium can fuel a nuclear submarine core if India continues in that program.

To describe the basis of these results, theory of enrichment, terminology used and equations required is discussed in the following paragraph. Performance of any enrichment is scaled on the basis of Separative Work Units (SWU) capacity that it can perform to. SWU describes the effort required to produce a unit mass in the separation devices using different techniques (Electromagnetic Separation, Gaseous-Diffusion, Centrifugal, Aerodynamic, Laser, Chemical and etc.) for relative enrichment of ^{235}U to that of ^{238}U. As shown below a SWU of approximately 192 is required to produce 1 kilogram of 90% ^{235}U from natural uranium of 0.72% ^{235}U producing a residual of 0.3% ^{235}U as depleted uranium waste to the enrichment process [31].

The feed "F" in kilograms required for a desired quality and quantity of product "P" in kilograms along with the waste "W" in kilograms is calculation follows. To calculate the feed 'F', fractional weights of the product, waste, feed and natural uranium are represented by N_p, $N_{w,}$ N_f and N_0 respectively. The value function $V(N)$ for the feed 'F' is given by

$$(2N - 1) * \ln\left[\frac{N}{(1-N)}\right] + b * N + a * P + W = F \quad (1)$$

where “a” is

$$- \ln\left[\frac{(1- N_0)}{N_0}\right] - \left[\frac{(1 - 2*N_0)}{(1 - N_0)}\right] \quad (2)$$

and “b” being

$$2* \ln\left[\frac{(1- N_0)}{N_0}\right] + \left[\frac{(1 - 2 *N_0)}{(N_0 * (1 - N_0))}\right] \quad (3)$$

Here N_0 is equal to 0.0072 for natural uranium. Using the value of N_0 in equations (2) and (3) “a” and “b” are -5.933 and 149.72, respectively. Thus the value function $V(N)$ can be stated as

$$(2N - 1) * \ln\left[\frac{N}{(1-N)}\right] + 149.72 * N - 5.933 \quad (4)$$

Applying conservation of mass equation

$$P + W = F \quad (5)$$

and conservation equation for the enriched isotope is given by

$$N_p * P + N_w * W = N_f * F \quad (6)$$

To produce 1 kilogram of 90% enriched uranium with $N_p = 0.9$ (90% Enriched U), $N_w = 0.003$ (Depleted U) and $N_f = 0.0072$ (Natural U) the feed ‘F’ required is calculated to be 213.57 kilograms of natural uranium (0.72% ^{235}U) and waste is 212.57 kilograms.

Solving the equation for separative work units,

$$SWU = V(N_p) * P + V(N_w) * W - V(N_f) * F \quad (7)$$

it is calculated that 192 SWU achieves the desired product.

Little is known about the nuclear submarine design and the number of them being planned for strategic deployment. Going by the standard designs, it is assumed that India would need 100 kilograms of 90% ^{235}U for each submarine core. The basis of the assumption is explicitly calculated and shown in Appendix H.

The uranium enrichment activity at the Rattehali plant in Mysore (Table V in section III) began in 1990. An assumption of 2000 SWU/year of installed capacity was done to analyze the enriched uranium production. With the given assumption, over a period of 10 years the facility would have produced 20,000 SWU. It requires 192 SWU and 213 kilograms of natural uranium to produce 1 kilogram of 90% ^{235}U. Therefore, the total expected 90% ^{235}U accumulation over 10 years period is 104.4 kgs. This is approximately the quantity of fuel required for one core of the Indian nuclear submarine, under the assumption that India pursues the submarine program. Thus the fuel for the first nuclear submarine of India, as well as an additional core in every 10 years, would be available, irrespective of the status of U.S.-India nuclear cooperation agreement. The production of uranium for future submarine cores however may require additional resources.

IV.F. Assessment of Power Production and Uranium Consumed

India's nuclear power plant analysis involves computing the quantity of fuel used along with the accumulation of the spent fuel. The capacity factors listed in Tables II and III were used to compute the UO_2 consumed by each facility. Fresh UO_2 fuel bundles are

not used exclusively in initial CANDU core loading, to avoid flux peaking due to plutonium production with the consequent delayed neutron fraction. Selective positioning of depleted uranium fuel bundles flattens the flux and improves controllability during fuel reshuffling stage [35].

TABLE XIII Fuel Consumed by PHWRs Until 2003

POWER PLANT	**UO_2 USED (TONS)**
RAPS-1	255
RAPS-2	436
MAPS-1	378
MAPS-2	339
NAPS-1	274
NAPS-2	281
KAPS-1	267
KAPS-2	254
KAIGA-1	91
KAIGA-2	122
RAPS-3	88
RAPS-4	90

TABLE XIV Fuel Consumed by PHWRs from 2004 to 2006

POWER PLANT	**UO_2 USED (TONS)**
All 12 Plants Stated in Table XIV	366
TAPP–4 + All 12 Plants Stated	352
TAPP–3 & 4 All 12 Plants Stated	257

As stated earlier the fuel quantities, refueling sequence and reshuffling period have all been studied with detailed computations on 37-pincell CANDU fuel bundle design. For the 220 MWe and 540 MWe rated power reactors the initial core loading is of 57.65 tons and 141.5 tons of UO_2 respectively. The first criticality of the core can either be achieved with a combination of fresh and depleted uranium fuel or fresh uranium fuel with a few bundles of thorium. RAPS and MAPS received their initial loads of 17.65 tons of spent fuel from CIRUS reactor. Spent fuel discharged from the RAPS and the MAPS was used for the initial fuel loading at the Narora Power Station (NAPS). But from Kakrapar Atomic Power Station (KAPS) onwards thorium bundles were used along with fresh UO_2 fuel bundles in the first core loading [the details of the initial loading of the core are based on my knowledge from experience as reactor operator in KAPS]. During the pre-refueling period of 140 days, there is excess reactivity to meet the power demand. So flux flattening is carried out by reshuffling of

fuel bundles. During this period there is a saving of 17.6 tons of uranium in 220 MWe reactors and 43.2 tons in 540 MWe reactors.

The total amount of UO_2 consumed until 2006 is determined through the sum of quantities in the right-hand-side of Tables XIII and XIV plus 40 tons each for the fresh core loading of the 12 plants. The total amount of UO_2 produced is 5410 tons (Table XII) and the total amount of UO_2 consumed is 4330 tons. In addition the production reactors (CIRUS and DHRUVA), consumed 609 tons of UO_2 by the end of 2006. Thus, the total UO_2 reserve available at the end of 2006 was 471 tons. Assuming no additional mining activity is added, the reserves and production for the power reactors can last for only a few more years. Table XV shows the fuel consumption for 2007 and 2008 from present operating plants and newer additions. All the NPPs in India are presently operating at 60% or lower capacity factor. The same is assumed for all the power plants under construction that may line up at the projected dates.

TABLE XV Fuel to be Consumed by PHWRs at 60% Capacity Factor

POWER PLANT	**UO_2 CONSUMPTION 2007 (TONS)**	**UO_2 CONSUMPTION 2008 (TONS)**
TAPP-4	55	55
TAPP-3	55	55
KAIGA-3	17	23
KAIGA-4	6	23
RAPP-5	6	23

TABLE XV Continued		
POWER PLANT	**UO_2 CONSUMPTION 2007 (TONS)**	**UO_2 CONSUMPTION 2008 (TONS)**
RAPP-6	0	7
RAPS-2	23	23
MAPS-1	23	23
MAPS-2	23	23
NAPS-1	23	23
NAPS-2	23	23
KAPS-1	23	23
KAPS-2	23	23
KAIGA-1	23	23
KAIGA-2	23	23
RAPS-3	23	23
RAPS-4	23	23

Summing up the consumption for 2007, as stated in Table XV, shows that by December 2007 India would consume an additional 392 tons of UO_2. If the operating capacity factors remain at 60% for 2008 then according to summation of the last column of Table XV an additional 449 tons of UO_2 will be consumed in the period from January 2007 to December 2008. Thus 841 tons of UO_2 will be consumed in the period from Jan 2007 to Dec 2008. The UO_2 reserves at the end of 2006 were 471 tons and the current production (see Table XI) is 26.6 tons per year. So the total UO_2 available during the stated period is only 524 tons. Thus consumption would exceed the production and reserve by December of 2008 even if operational capacity factors are further dropped to 50%.

V. FUEL CYCLE ANALYSIS FUTURE PROJECTIONS

The future of the Indian nuclear fuel cycle was assessed and possible options that a nation with receding resources would adopt were projected in this section. Fuel cycle possibilities resulting from scaled down domestic reserves along with established nuclear facilities were studied. The Indian fuel cycle is discussed in the light of U.S.–India nuclear cooperation and also under sustenance of long-standing international isolation following 1974 Pokhran-I test. The projections of facility utilization and material distribution were done utilizing the resources and capabilities reported in previous sections. The future of the existing nuclear power program inclusive of planned breeder program and weapons initiative was studied, under options that could evolve under given circumstances. Details of the U.S.–India nuclear cooperation agreement and its implications on the planned power projects were analyzed. The future of weapon program with and without the cooperation agreement was also analyzed. As earlier all the calculations were concluded with simulations on input decks for 37-pincell CANDU fuel bundle in 2D lattice code HELIOS-1.4 using 34 neutron and 18 photon energy groups. A complete fuel assembly was considered as a fuel map, and fuel characteristics were reported for a combination of 67 elements and isotopes.

V.A. Projections under NO U.S.-India Civilian Nuclear Cooperation

V.A.1. Future of Existing Nuclear Power Program

As concluded in section IV the uranium ore reserves in tandem with annual production would reach a null point by December of 2008. The date has been obtained

by calculating the actual uranium consumed (as stated in Table XIII, XIV and XV) by power plants and research reactors compared against past and present production levels of uranium. The projections of the uranium consumption and operating plants capacity factors for the future were made on the basis of a pragmatic approach a nation would take, which foresees an end to its resources.

From December of 2008 India would continue to have an annual production of 26.6 tons of UO_2. This quantity cannot meet the needs of nuclear power plants, which ironically contribute to less than 2.5% of the electricity needs. If uranium is not available from international suppliers then all existing nuclear power plants would have to be shut down. Addressing the issue of losses opens the ray of hope for meeting the electrical needs. Transmission and distribution losses along with theft amount to a large percentage of the total electricity produced. If corrective measures can save on this front, then they could exceed the production of electricity by all the existing nuclear power plants.

If the scenario where all nuclear power plants are devoid of natural uranium fuel and only 26.6 tons of UO_2 is produced every year, as seen since 1998, is continued beyond 2008 then the available uranium can only meet the demand of DHRUVA reactor. DHRUVA would produce 23.16 kilograms of plutonium per year (as computed with burnup code ORIGEN2.2) with the given fuel. This state of affairs can go on till ores completely dry out. A turn around of the nuclear power industry could come with discovery of new ore sites or availability of unsafeguarded natural uranium from non-NSG suppliers. This situation, with minute aberrations mostly would continue till end of

2012, when India's uranium ore reserves are supposedly expected to get a boost from the present levels by newer mining sites [33]. By the end of 2012, India would then have two operational fast breeder reactors with fissile material being generated at faster rate than ever produced and eighteen PHWRs, even at a low capacity factor providing spent fuel for its second-stage program.

Two new factors that can fasten up the foreseen case are increase in production of old mines and discovery of new mines prior to 2012. From the open source information it can be said that a lot of effort to start newer mines is in progress. One of those is the Turamdih project; the total reserves of this mine have been assessed at 7.6 million tons of ore at a depth of 170 m. details of the assessed values is followed after the Table XVII. This mine was discovered in 1988 but was soon abandoned because of low ore-grade (0.02%). Work on it began later on 26th January 2004 and it is expected to in service by January of 2009. For the first three years the ore production is planned to be 550 tons per day and later increased to 750 tons per day. There is also a second site called Banduhurang mine where commissioning work was started on the same day (26th of Jan 2004). This can also be assumed to start supplying domestic uranium by January of 2009.

There are no mining details available so financial allocations were reverse calculated to relate to the rate of production. The Turamdih project has a budgetary commitment of $40.65 million (~2000 million rupees in 2003-04). This can used for 10 megatons of ore mining or slightly more (beneficial figure is 11.68 tons of UO_2

equivalent) considering a cost of $4 million for mining 1 megaton of ore. Data for compounds of uranium at various stages were calculated and are stated in Table XVI.

TABLE XVI Calculated Values of U_3O_8, UF_6 and UO_2 from Turamdih Mine

Time Frame	**Total Ore (MEGATONS / YEAR)**	**U_3O_8 (TONS / YEAR)**	**U_3O_8 [g] (TONS / YEAR)**	**UF_6 [h] (TONS / YEAR)**	**UO_2 [i] (TONS / YEAR)**
2009-11	20.075	36.14	30.71	30.82	23.45
2012 Onwards	27.375	49.28	41.88	42.02	31.98

g. AFTER 15% LOSSES
h. AFTER 20% LOSSES
i. AFTER 0.8% FABRICATION LOSSES

TABLE XVII Fuel Use, Reserve and Production in Tons for Eight Unsafeguarded PHWRs

260 DAYS REFUELING	**SCENARIO – 1 85% C.F.**		**SCENARIO – 2 70% C.F.**		**SCENARIO – 3 50% C.F.**	
YEAR	**PRODUCTION + RESERVE**	**USE**	**PRODUCTION + RESERVE**	**USE**	**PRODUCTION + RESERVE**	**USE**
2007	415.6	222	415.6	183	415.6	130.5
2008	220.2	222	259.2	183	311.7	130.5

TABLE XVII Continued

260 DAYS REFUELING	SCENARIO – 1 85% C.F.	SCENARIO – 2 70% C.F.	SCENARIO – 3 50% C.F.	
2009	Not possible to operate at this Capacity factor		242.9	130.5
2010			174.2	130.5
Jan 2011			8.78	10.88

From the calculated values of the production and reserve, the use of uranium in three different scenarios was analyzed in Table XVII. As per the envisioned scenarios, if the PHWRs are operated at 50% capacity factor they can stretch their operating period till 2010. At any higher capacity factor India will run out of fuel even for the reactors not safeguarded under the separation of civilian and military facilities. This plan is a calculated initiative wherein 8 PHWRs are kept as unsafeguarded to provide fuel for the second-stage of Bhabha's three-stage-program. The sole idea is to keep operating till 2011 so as to buy time to start the production from two big mines as found in Andhra Pradesh and Meghalaya. This scenario can also be realized with few of the power plants being shut-down for couple of years for coolant channel replacement, with only eight PHWRs kept operational at 50% capacity factor. A second possibility is to fuel most of the reactors with low and medium burnup fuel if the challenge of handling irradiated fuel for on-power refueling can be met.

V.A.2. Future of Fast Breeder Program

The basis of fast breeder program is extraction of reactor-grade plutonium from the PHWR spent fuel. Taking into account an average burn up of 6500 MWd / tU for CANDU fuel bundles, 4.108 kilograms of plutonium (ratio of ^{240}Pu / ^{239}Pu = 42%) can be extracted from one ton of PHWR spent fuel. Given the historical operational reprocessing capability of 50 tons of spent fuel per year (before 1991 only one reprocessing plant of 100 tHM/yr existed and a capacity factor of 50% is assumed for it), India could extract 205 kilograms of plutonium every year from the spent fuel and in total could produce 19.6 tons of plutonium. This quantity of plutonium is enough to support twenty fast breeder reactors of 500 MWe ratings backed up by desired reprocessing capabilities. The fast breeder program does not seem to be deterred by the failure of cooperation as fuel needs can be met from the vast reserves of the thermal reactor spent fuel and presence of required reprocessing capability. Whether the second-stage-power-program involving fast breeders is economical or not is unimportant for the scientific community, because of its ability to produce weapons-grade material. Each FBR further enhances the plutonium and ^{233}U accumulation beyond the levels of present production rates [36].

The present reprocessing capability of 205 kilograms of plutonium per year can support construction of a 500 MWe fast reactor power plant in every five years. The obstacle for additional expansion of the fast breeder program is financial strength and a lack of fast reactor power plant operational history but definitely not reprocessing capacity. Fast reactors in France, Russia and Japan have experienced difficulties in

handling of sodium. There is also a concern of debris clogging in the primary loop and core entry nozzles because of the compact core size and strict dimensions.

It all depends how far India goes to pursue its second-stage breeder program despite the economic strain from it. India is a growing economy with extended nuclear infrastructure. If India could sustain the establishment of more fast breeders a closed fuel cycle with fast breeder reactors can be realized. Judging on the basis of spent fuel reserves and reprocessing capabilities India can sustain a feasible breeder program with 20 FBRs (the fuel for which can be made available from the spent fuel reserve of PHWRs) at a doubling time of 20 years. A conservative doubling time of 20 years was assumed as the breeding ratio is not known and operational capabilities are questionable. If the doubling time is less than 20 years, then the cycle can be sustained with less number of reactors.

The idea in the second stage of the three-stage-program is to sustain the fast breeder program by recycling plutonium, and have ^{233}U produced in a thorium blanket in the third-stage. Proliferation concerns are raised by the fact that weapons-grade plutonium is produced and ^{233}U a second type of weapon fissile material is accumulated.

A non-proliferating fuel composition would have been one in which plutonium is produced at a higher ratio of ^{240}Pu / ^{239}Pu and ratio of (^{233}U + ^{235}U) / U is less than 20%. A reactor system which closely resembles these features is thermal breeder reactor (TBR). In the past TBRs of the likes of Shippingport Atomic Power Station operated for 5 years (1977-82) and fuel characteristics at the end (reported in 1987) showed a rise of 1.3% of fissile material content from that of initial loading. TBR systems can gain

importance under the present scenario of international collaboration and proliferation concerns. A study was carried out on a TBR design that can replace the second-stage of the power program and integrate second and third-stage. Details of the TBR are discussed in section VI, "Alternate Reactor Systems for the Proposed Fuel Cycle".

Assertion of India's commitment to the three-stage program is evident from the cumulative fuel cycle analysis in a flowsheet form until 2006 (Figure 4). There appears to have been a gross disregard to the reality of domestic uranium reserves in the process of adding new power plants presently under construction. A quick advent to breeder program with indigenous capabilities of reprocessing and design of fast core in a closed fuel cycle might have produced a different scenario altogether.

V.A.3. Effect on Weapons Program

According to the assessment of Figure 4, India has sufficient plutonium to produce 69 implosion weapons (considering 6 kgs of plutonium used for each weapon). If uranium production is solely diverted to DHRUVA, addition of four weapons a year can be maintained. PHWRs operating at lower capacity factor could produce weapons-grade plutonium, even if not operated deliberately for that cause. The same rate of success of enrichment plant assures the possibility of thermonuclear weapons and nuclear submarine core fuel.

If the collaboration efforts fail then most of the nuclear power plants could be closed down from power mode and plutonium production continues at the present level from the production reactors. The plutonium production rate may also rise because of low capacity factor operating power reactors. Contribution from breeders could be an

additional boost to the weapons program. Each fast core can produce approximately 140 kilograms of ^{233}U every year [37]. The total stockpile of nuclear weapons which has been added over four decades of struggle can then be repeated every year. This scenario would still have its doors open to cooperation but no effort could role back the clock to lower rate of weapons-grade fissile material production.

A possible case of all the 8 PHWRs not under safeguards being operated at 50% capacity, the fuel needs for next five years (2007-2011) would be (9.4 tons/plant/year * 8 plants * 5 years) 376 tons. This is definitely possible given the present uranium reserves along with the lowered rate of production. The alarming scenario is production of 460 kilograms of plutonium by each PHWR for next five years at a ^{240}Pu / ^{239}Pu ratio of 20.3%. India is presumed to have tested nuclear device with reactor-grade plutonium in Pokhran-II tests in 1998 [17].

V.B. Projections under YES U.S.-India Civilian Nuclear Cooperation

V.B.1. Details of U.S. – India Nuclear Cooperation Agreement

The intent of the U.S.–India nuclear agreement is to bring 14 of the 22 reactors under India-specific safeguards. These safeguards are somewhat different than that applied to Nuclear Weapon States (NWS) and Non-Nuclear Weapons States (NNWS) as defined under the NPT. Four such reactors are already under IAEA safeguards and two under construction also will be. This cooperation also brings all future PHWRs to be built, under the safeguards. The most controversial research reactor of the non-proliferation debate, CIRUS, would be shut down by 2010. CIRUS produced more or

less 1/3rd of the total weapons-grade plutonium that India possesses as of now (187.2 kilograms out of the total of 633.5 kilograms produced between 1964 and 2006). The implications of this cooperation can better be understood by analyzing the implications for the nuclear power program, fast breeder program and weapons initiative. In the process of material accounting and analysis of facility capabilities, the observations and conclusions of the CRS reports stated in references [38-50] were also addressed.

V.B.2. Future of Nuclear Power Program

Under the cooperation agreement India would have eight unsafeguarded commercial reactors. The spent uranium fuel rods used in this heavy-water cooled and moderated nuclear power plants can be reprocessed to extract plutonium for the fast breeders of the second-stage. While operating in power mode the uranium fuel remains in the reactor for 300 to 310 days to reach a burnup of 7000 MWd / tU resulting in 2600 grams of ^{239}Pu, 1108 grams or more of ^{240}Pu, 306 grams or more of ^{241}Pu, and about 90 grams of ^{242}Pu for every ton of natural uranium fuel. Plutonium extracted from commercial reactors is generally considered less desirable for use in nuclear weapon due to a high concentration of ^{240}Pu and plutonium isotopes of even higher isotopic number. Table XVIII details the required fueling days for CANDU cores to produce the desired amount of reactor-grade plutonium while operating at a specific capacity factor. It also details the refueling sequence days, total plutonium production with quality and requirement of natural uranium fuel. A prospective option of operating the reactors at 100% capacity factor with 300 refueling sequences per year of 8 bundles per day would produce 4.108 kilograms of reactor-grade plutonium from a ton of heavy metal as

computed from ORIGEN2.2. Accounting for the amount of plutonium required from each FBR core (assumed to be 850 kilograms) it is concluded that an optimized number of eight PHWRs (six of 220 MWe and two of 540 MWe) are desirable for pursuing the fast breeder program of one a year. Eight PHWR's (6 X 220 MWe and 2 X 540 MWe) operating with the stated refueling sequences would produce 217.152 tons of spent fuel, which would result in 892 kilograms of plutonium every year. This conclusion is based on the reprocessing capabilities of 250 tHM/yr presently available and fueling needs of the 500 MWe rated FBR (assumed 850 kilograms).

TABLE XVIII Plutonium Production, Uranium Use and Refueling Days at Different Capacity Factors

NPP of 220 MWe	(For 300 Days of Refuel-ing per year)	(Considering 300 Days of Refueling per year at the Burnup given in Column 2)							(At 7000 MWd/ tU burnu p per year)
C.F. of the NPP (%)	Burnup **MWDa ys / tU**	Pu-239 **Gms**	Pu-240 **Gms**	Pu-241 **Gms**	**Ratio** of Pu240 / Pu239	Rx grade Pu in **Grams / Ton of U**	Use of Nat. U in **Tons**	Pu prod-uced in **Kgs**	Days of Refuel ing requir ed in a year
1	70	68.92	0.280		0.4		0.004	3	3
5	350	324.6	6.536		2.0		0.094	12	15
10	700	605.4	24.16		4.0		0.377	24	30
11	770	657.1	28.8		4.4		0.456	26	33

TABLE XVIII Continued									
NPP of 220 MWe	(For 300 Days of Refuel-ing per year)	(Considering 300 Days of Refueling per year at the Burnup given in Column 2)							(At 7000 MWd/ tU burnu p per year)
C.F. of the NPP (%)	Burnup **MWDa ys / tU**	Pu-239 **Gms**	Pu-240 **Gms**	Pu-241 **Gms**	**Ratio** of Pu240 / Pu239	Rx grade Pu in **Grams / Ton of U**	Use of Nat. U in **Tons**	Pu prod-uced in **Kgs**	Days of Refuel ing requir ed in a year
12	840	707.5	33.78		4.8		0.543	28	36
13	910	756.6	39.08		5.2		0.637	30	39
14	980	804.5	44.69		5.6		0.739	32	42
15	1050	851.1	50.61		5.9		0.848	34	45
20	1400	1068	84.35	5.674	7.9	1158	1.5	44	60
25	1750	1262	124.4	10.24	9.9	1397	2.4	53	75
30	2100	1437	170.1	16.51	11.8	1624	3.4	61	90
35	2450	1594	221	24.66	13.9	1840	4.6	69	105
40	2800	1738	276.9	34.88	15.9	2050	6.0	77	120
45	3150	1869	337.7	47.36	18.1	2254	7.6	85	135
50	**3500**	**1988**	**403.3**	**62.31**	**20.3**	**2454**	**9.4**	**92**	**150**
55	3850	2096	473.4	79.9	22.6	2649	11.4	100	165
60	4200	2194	547.8	100.3	25.0	2842	13.6	107	180
65	4550	2282	626.3	123.7	27.4	3032	15.9	114	195
70	4900	2360	708.4	150	30.0	3218	18.5	121	210
75	5250	2429	793.5	179.3	32.7	3402	21.2	128	225
80	5600	2490	880.8	211.6	35.4	3582	24.1	135	240
85	5950	2541	969.6	246.4	38.2	3757	27.2	142	255
90	6300	2584	1059	283.6	41.0	3927	30.5	148	270
95	6650	2620	1147	322.6	43.8	4090	34.0	154	285
100	7000	2649	1235	362.9	46.6	4247	37.7	160	300

These eight PHWRs dedicated for power generation and would need 244 tons of natural uranium at the rate of one refueling a day. The requirement of 244 tons of fuel is close to the assumption of 250 tons of natural uranium as material constraint in the analysis of latency capability in Figure 5. If the unsafeguarded reactors operate at the assumed capacity factor of 90% then annually 1280 kgs (160 kilograms of reactor-grade plutonium can be produced by each PHWR annually) of reactor-grade plutonium would be at India's disposal.

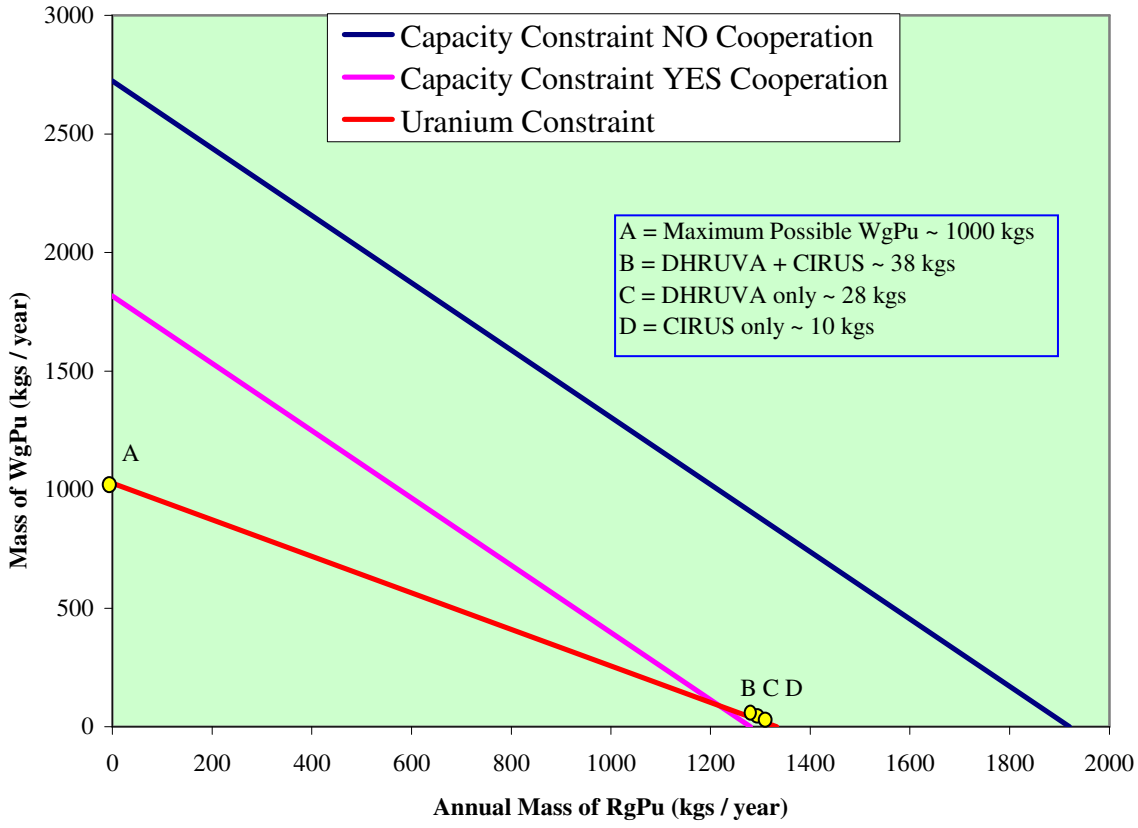

Fig. 5. Latency capability with/without cooperation and uranium constraint

This quantity of reactor grade plutonium can drive one and half fast reactor cores every year with an assumption of 850 kgs being required to fuel each FBR. Figure 5 concludes that the material constraint also brings down the latency capability of producing weapons-grade plutonium. Diverting 250 tons of natural uranium would lead to production of 1027 kgs of weapons-grade plutonium annually considering no restrictions on the refueling rate is described by point 'A' in Figure 5. The points B, C and D in Figure 5 state the weapons-grade plutonium production DHRUVA and CIRUS operational simultaneously, only DHRUVA and only CIRUS operational respectively.

V.B.3. Future of Fast Breeder Program

The nuclear cooperation treaty with the U.S. opens the option of all future commercial nuclear power plants to be kept under safeguards. If India decides to do so then to augment the supply of plutonium in the future, India would have to pursue the path of reprocessing of international nuclear fuel to further the progress of fast breeder reactors.

The spent fuel from the eight commercial reactors not under safeguards needs to be diverted to India's fast breeder program which will initially be driven by uranium-plutonium fuel. This fuel fed into a fast breeder reactor can eventually be recovered after a specified doubling time and reprocessing schedule. The process thus involves a time lag in producing its own fuel by breeding [51].

Consider the case where international fuel is made available to all safeguarded PHWRs from 1st of January 2007 and India has to fuel eight unsafeguarded PHWRs, DHRUVA and provide feed for enrichment plant. As per the calculations, by 1st of

January 2007 India would be left with 471 tons of UO_2 as reserves along with yearly production of 26.6 tons. The demand of fuel for eight PHWRs operating at 80%, with 240 days of refueling per year is 192.8 tons of UO_2 per year (8*24.1, from the 80% capacity factor row of Table XVIII). Actual need is of 300 days of refueling per year to have substantial positive reactivity (12 mk, as computed by the 37-pincell HELIOS1.4 input deck) for xenon build-up override during fast start up after an inadvertent shutdown. The PHWRs can be operated at lower capacity factors to stretch the operating days. Such a state of affairs can continue until the newer mines start to deliver uranium.

V.B.4. Future of Weapons Program

To accumulate 633.5 kilograms of weapons-grade plutonium it takes approximately 609 tons of natural uranium (stated in Table X). Including this in the total consumption of natural uranium, the reserves left as on end of 2006 would be meager 471 tons. It can barely fulfill the need for fuel till December of 2008 in the present CANDU power plants operating at 60% capacity factor. As calculated from 60% capacity factor row data of Table XVIII the natural uranium need is 462.4 tons (13.6 tons/plant * 17 plants).

If natural uranium supplies are ensured from international community and the natural uranium reserves are used for production of plutonium in the DHRUVA reactor, then it would result in accumulation of another 410 kilograms of weapons-grade plutonium over a period of fourteen years. The other use of natural uranium is feed to the enrichment plant. By enrichment of the available natural uranium feed a product of 2211 kilograms of 90% ^{235}U can be produced. This definitely surpasses the need of 417.6

kilograms of 90% ^{235}U for four nuclear submarines, and their refueling after 10 years. But to separate this amount of feed the enrichment plant needs to be of 424563 SWU capacities. Given the present 2000 SWU per year it would take 212 years to enrich the uranium in reserve.

TABLE XIX Weapons-Grade Plutonium and Refueling Days at Different Capacity Factors

Capacity factor (%) with burnup of 1050 MWd / tU	Weapon-grade-plutonium production in kgs per year	Number of Refueling required in a year to maintain burnup of 1050 MWd / tU
15	34	45
20	45	80
25	57	125
30	68	180
35	79	245
40	91	320
45	**102**	**405**
50	113	500
55	125	605
60	136	720
65	147	845
70	159	980
75	170	1125
80	181	1280
85	193	1445
90	204	1620
95	215	1805
100	227	2000

DHRUVA and enrichment activities can both share the meager domestic uranium along with the eight PHWRs operating at lower capacity factor. The scenario would change with availability of ore from new mines. If desired the power reactors can be operated on a faster refueling sequence at a lower capacity factor to produce weapons-grade plutonium. From Table XIX it can be seen that a 220 MWe rated power plant can produce 102 kilograms of weapons-grade plutonium by operating at 45% capacity factor with 405 refueling sequences in a year. Table XIX also describes other combinations of capacity factors, with associated refueling schedules, to produce weapons-grade plutonium.

For the production of weapons-grade plutonium with lower ^{240}Pu concentrations from power reactors, the fuel rods in a reactor have to be changed every 43 days or less (derived from ORIGEN2.2 input deck at a burn up of 1050 MWd/tU to have ^{240}Pu / ^{239}Pu ratio < 6%). Indian heavy water reactors do not have to be shut down in order to change fuel rods. So India has the option to produce weapons-grade plutonium from those of its 8 commercial nuclear power plants not under safeguards by changing some of the fuel rods at a fast rate. There is a limitation on the number of refueling that can be done in a single day. Mostly it is one refueling sequence per day or two to make up for the missed occasions. On very rare occasions three channels refueling has been attempted in a day in the CANDU reactors (statement based on reactor operator experience). The fuel movement inside the core is done in less than 18 minutes so that the peripheral bundles do not see the peak flux while passing through the center of the core. For the whole refueling process it takes close to 4 hours and counting for the power

reduction and raise via chemical shim, (depends on the channel), it is close to 6 hour schedule.

Given the limitations on the refueling rate it is doubtful that PHWRs may serve as a source of weapons-grade plutonium. Past records of plutonium production, which can be obtained from capacity factors of power plants, do not indicate PHWRs ever being used for weapons-grade material production.

VI. ALTERNATE REACTOR SYSTEM FOR THE PROPOSED FUEL CYCLE

An alternate reactor system that would further the cause of separation of military and civilian nuclear fuel cycle and enable efficient use of resources under safeguards and strengthen the non-proliferation cause has been developed. The alternate reactor system was optimized for efficient utilization of domestic thorium and international uranium resources. The proposed alternate reactor system is a breeder in thermal spectrum with CANDU-type core geometry. This Thermal Breeder Reactor (TBR) was designed and optimized to fit the proposed fuel cycle of India. The suitability of the alternate reactor system in the power program was synthesized on specific metrics that effectively represent the objectives. This system was also studied in comparison to the presently planned PHWRs. The objective of developing a TBR was to convert ^{232}Th to ^{233}U at an optimized breeding ratio and simultaneously achieve high burnup on the fuel. The end-of-cycle (EOC) fuel composition was ensured to be proliferation resistant by optimizing the system with the specified metrics. The metrics to measure the objectives are [53]:

- Mass (kilograms) of uranium used per unit (MWday) of electricity production,
- ^{240}Pu / ^{239}Pu ratio to establish the reactor-grade nature of plutonium,
- Total plutonium production and
- Presence of less than 20% fissile uranium (^{233}U + ^{235}U) in the total uranium mass.

VI.A. Thermal Breeder Reactor Design Concept

Thermal breeder reactors primarily operate on the basis of neutron absorption by fertile isotopes in a thermal spectrum, producing fissile fuel than they consume. Earlier studies on breeders have shown that the absorption cross-section is an important factor in choosing fertile material for the core [53]. The fact that ^{232}Th breeds ^{233}U through neutron absorption and successive beta decays because of higher neutron absorption cross-section (than that of ^{238}U) was an overriding factor that favored use of thorium in breeders [54]. This though brings in the concern of proliferation as ^{233}U is a directly weapon usable material.

If ^{233}U and ^{235}U both direct weapon usable materials, produced or present in a mix with various transuranic and transplutonic isotopes and are less than 20% of the total uranium content then it would be hard to separate the weapons-grade material from the spent fuel. A high-cost reprocessing facility followed by enrichment process would be required to separate the fissile content. The intermixing in the spent fuel is obtained with homogeneous mixture of thorium and low enriched uranium oxide as fuel in the fresh core. This in the process also limits the percentage of thorium that can be used in place of ^{238}U in the core. In a way it is an advantage, because the lower absorption cross-section for epithermal neutrons in ^{232}Th leads to lesser negative power co-efficient incase of power transient. But again savings on thorium can fail the system with higher concentration of plutonium produced by fertile isotope ^{238}U when replaced with ^{232}Th. In a thermal breeder reactor of the suggested design the low enriched fuel reaches high burnup and achieves higher ^{240}Pu / ^{239}Pu ratio for the EOC fuel. The blanket with half

the size of the driver also breeds lesser quantity and reactor-grade plutonium. Also stated in previous section, the increased ratio of plutonium isotopes constraints the proliferation risk as spontaneous fission neutron yield of ^{240}Pu can prematurely detonate a weapon (called preinitiation).

VI.B. The Theory of Breeder Reactors

A fissile production rate that exceeds the fissile consumption rate can be attained through an appropriate combination of fissile and fertile mass arranged in a suitable geometry, in coincidence with proper reprocessing schedules [55]. Where, the ratio of the mean rate of fissile material production to the mean rate of fissile material consumption is defined as the "Breeding Ratio" (BR). A BR > 1 is a self sustaining reactor system and the reactor is called a breeder. If BR < 1 then the reactor is Converter Reactor and the Breeding Ratio is usually referred to as "Conversion Ratio" (CR). BR can be shown to be equal to the ratio of the fissile material produced (FP) to fissile material destroyed (FD) during a fuel cycle (i.e., between periodic refueling or at the end of average burnup) [53]:

$$BR = \frac{FP}{FD} \tag{8}$$

The breeding gain is given by

$$G = BR - 1. \tag{9}$$

Substituting equation (8) into equation (9) yields:

$$G = \frac{FP}{FD} - 1 \quad \text{or} \quad G = \frac{FP - FD}{FD} \tag{10}$$

In the numerator of equation (10) the fissile material produced minus the fissile material destroyed is basically the total fissile content at the EOC after subtracting the initial loading. Thus it can also be inferred as ratio of the difference of fissile materials at the end and beginning of cycle to the fissile material destroyed. The breeding gain

$$G = \frac{FEOC - FBOC}{FD} \tag{11}$$

is normally accounted for at the end of the one year but since the suggested breeder design being an on-power refueling reactor, it is also computed at average burnup.

Like any other reactor the physics of thermal breeding reactors are driven by neutron economy. A nuclear reactor can breed over a broad energy spectrum but adequate breeding ratios can only be realized in a certain energy range. A high breeding gain is attained with a fast neutron spectrum, although a low breeding gain on a faster fueling cycle characteristic of some thermal breeder designs like that of on-power refueling CANDU-type cores is an overriding design as shown through this study. The terms universally known in the fission process are 'ν' is number of neutrons produced per fission, 'η' is number of neutrons produced per absorption and 'α' being capture to fission ratio (σ_c / σ_f).

These parameters are related by,

$$\eta = \frac{\upsilon\sigma_f}{\sigma_f + \sigma_c} = \frac{\upsilon}{1 + \sigma_c / \sigma_f} = \frac{\upsilon}{1 + \alpha}, \tag{12}$$

Where the parameters ν and α are measured quantities, while η is a derived quantity. In a thermal neutron energy spectrum ν is fairly constant up to the energy range

of 1 MeV for each of the primary fissile isotopes. While α varies considerably with energy and between isotopes. This behavior of ν and α leads to variations in η over energy range. In a thermal reactor the energy of the neutron increases with rise in moderator temperature. This results in capture of neutrons in resonance cross-sections leading to reduced η. Increases in fuel to moderator ratio also enhances the importance of the near thermal and epithermal neutrons, thus reducing the average value of η. Considering a simple neutron energy balance where one neutron is absorbed by fissile nucleus in order to continue chain reaction and 'L' neutrons lost unproductively, parasitic absorption (capture in structures, coolant, control rods, poisons and fission products) and also by leakage from the reactor. The number of neutrons left for capture by fertile nucleus is $[\eta-(1+L)]$. So to produce a fissile nucleus from fertile nucleus after a fission event we need to have $[\eta-(1+L)]\geq 1$ or $\eta \geq 2+L$. The quantity $[\eta-(1+L)]$ is basically fissile nuclei produced to fissile nuclei destroyed, thus making the breeding ratio (BR) equal to $[\eta-(1+L)]$. Taking into account the neutrons lost due to moderator and coolant expansion the breeding ratio actually achieved is

$$BR=[\eta\varepsilon-(1+L)], \tag{13}$$

where ε is the correction factor for the moderator and coolant temperatures.

From equation (13) it can be concludeded that maximum BR possible is $\eta\varepsilon-1$ or $\overline{\eta}-1$, where $\overline{\eta}=\eta\varepsilon$.

Thus the maximum breeding ratio

$$BR_{\max}=\overline{\eta}-1 \tag{14}$$

represents the quantity of fissile material produced in a certain time or at average burnup. Now to calculate the actual quantity of fissile material produced, a gain factor that is representative of the breeder's gain needs to be formulated.

A reactor gain factor (RGF) is a representative number for a particular breeder reactor which producing fissile material in excess of its own fissile inventory to fuel an identical reactor at the end of its average burnup cycle. Given the average burnup of fuel type, the required doubling time varies accordingly. In this study, RGF has been defined in terms of initial fissile inventory (M_0 kgs) used by the reactor and the fissile material (M_g kgs) gained, at the end of the average burnup.

$$RGF = \frac{M_0}{M_g} \tag{15}$$

Where, M_g kgs is the produced fissile material content (^{233}U, ^{237}Np and Pu)

$$M_g = FEOC - FBOC \tag{16}$$

From equations (11) and (16) we have

$$M_g = FD * G \tag{17}$$

Or, $M_g = G * (1-\alpha) *$ (Fissile mass fissioned till the average burnup) (18)

The fissile mass fissioned till the average burnup is computed as power in watts of the reactor core multiplied to 2.93 X 10^{10} fissions / watt-seconds and time to reach average burnup stated in seconds along with molar mass of 238 gms / gm-mole. The correction factor to the calculation for actual operation is accounted by multiplication of the

fraction of power. The final value obtained is finally divided by 6.023 X 10^{24} atoms / gm-mole to get the mass in grams [56].

The other measure to judge the fissile material production is the Reactor Doubling Time (RDT), which is the time required by the particular breeder reactor to produce fissile material in excess of its own fissile inventory. Hence, it is the time necessary to double the initial load of fissile material to fuel an identical reactor. Defining M_{gd} kgs as the time averaged difference between the fissile inventory at the beginning of the year and the fissile inventory at the end of the year. Thus

$$RDT = \frac{M_0}{M_{gd}} \tag{19}$$

and $M_{gd} = FEOC - FBOC$, computed at the end of one year (20)

From equations (11) and (20) we have

$$M_{gd} = FD * G \tag{21}$$

Or, $M_{gd} = G * (1 - \alpha) *$ (Fissile mass destroyed in a year) (22)

The fissile mass fissioned is calculated in the similar manner with time being considered as one year instead of time to reach average burnup.

VI.C. Reactor Physics Simulations

VI.C.1. The SCALE Code System

The whole core 3D model of TBR, in CANDU-type core geometry with various combinations of fuel compositions, was created with SCALE version 5.1 modular code systems. This SCALE modular code system is developed and maintained by Oak Ridge

National Laboratory (ORNL) and is widely accepted around the world for criticality safety analysis and depletion calculations. Figure 6 shows the modules in a sequential pattern for executing the desired input.

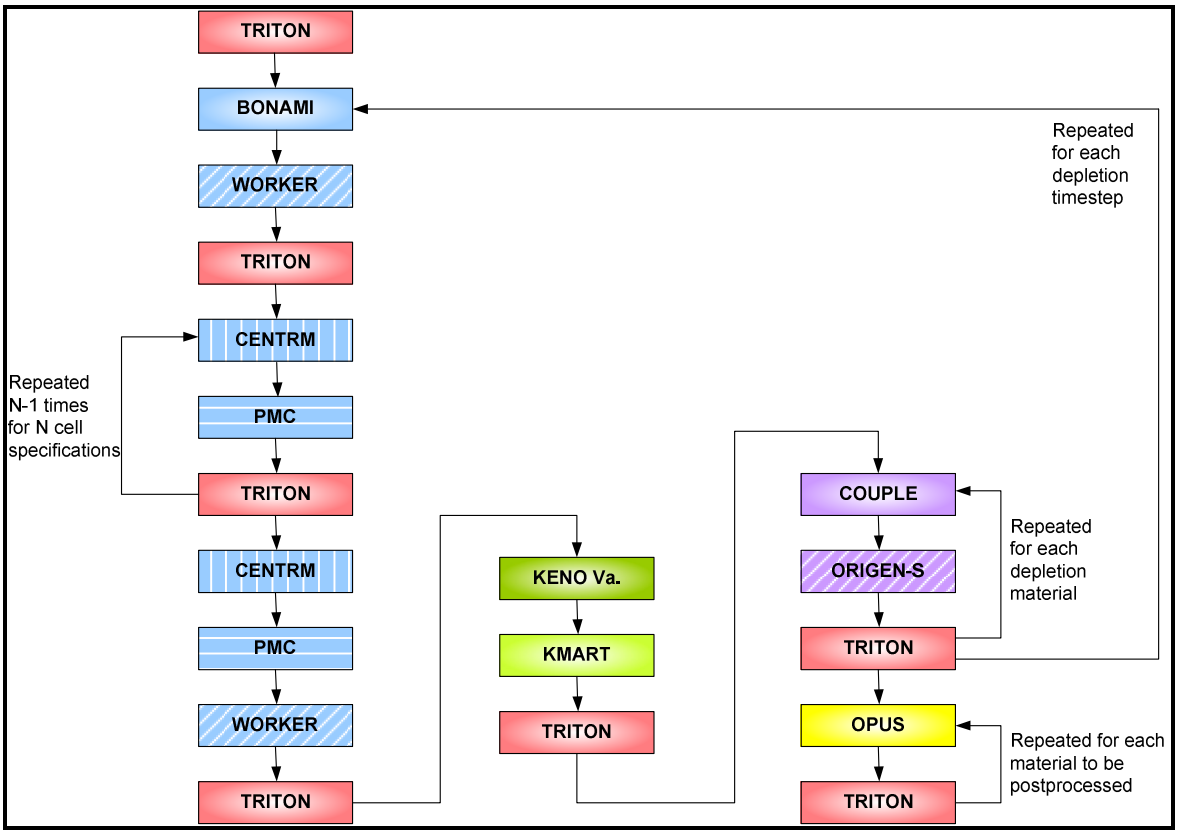

Fig. 6. Execution path for the TRITON analytical depletion sequence

The SCALE control module TRITON was used to perform depletion calculations. TRITON couples KENO V.a (3D Monte Carlo transport code) with the well known SCALE point depletion and decay module ORIGEN-S, which tracks more than 1500 nuclides. All TRITON models used the 238-group ENDF/B-VI cross section

library and the BONAMI, WORKER/CENTRM/PMC modules for cross section processing. The BONAMI module provides resonance corrected cross sections in the unresolved resonance range, and WORKER/CENTRM/PMC modules provide resonance corrected cross sections in the resolved resonance range. CENTRM is a one-dimensional discrete ordinates code that computes space-dependent, continuous-energy neutron spectra. PMC uses the spectra to collapse the continuous-energy cross section data to multi-group (238 group) data for use by KENO V.a.

As mentioned, KENO V.a is a multi-group 3D Monte Carlo criticality transport code that determines the effective multiplication factor. The KENO post-processing utility KMART is used to extract fluxes, determine power distributions, and collapse cross sections to the three-group form required by COUPLE and ORIGEN-S for depletion calculations. The OPUS module provides the ability to extract specific data from ORIGEN output libraries, perform unit conversions, and generate plot data for post-calculation analysis [55]. Illustrated in Figure 7 is the calculation flow path during TRITON depletion calculations. The KENO V.a models were run with 4050 neutron generations, skipping the first 50 generations. There were 5000 neutrons in each generation, resulting in a total of 2 million neutron histories.

VI.C.2. Simulation Details and Design Iterations

A series of stepwise iterations with different combinations of fuel configurations for driver and blanket fuel suitable to CANDU-type geometry was done to optimize TBR. The optimized breeder core TBR-1 finally decide on comprises of 80:20 uranium-thorium in driver fuel and 80:20 proportion of spent fuel from PHWRs and thorium in

blanket assemblies. The driver fuel portion of the core (i.e., 304 out of 456 channels) comprise of 3% of ^{235}U. The radial blanket assemblies surrounding the driver fuel are made of 80% spent fuel of CANDU-6 core after the fission products were removed and rest 20% is thorium. Tables XX and XXI details the fuel composition in the core. This combination of fuel achieves nearly 90% of the burnup of a CANDU-6 reactor with slightly higher rate of depletion from fresh core to the end-of-life (shown in Figure 7) in the first cycle. The average burnup achieved for the final two breeder core designs TBR-1, 2 and CANDU-6 fuels were 18.72, 19.68 and 22.08 GWd/MTHM respectively. The TBR-2 core KCODE and depletion modules are also reported for comparison purposes. TBR-2 driver fuel comprises of pure uranium metal and blanket being of 100% thorium fuel. As would further stand out in the study, spent fuel from the TBR-2 is not proliferation resistant and does not fulfill the necessity of burning the plutonium accumulated from thermal reactors. The blanket being made of 100% thorium also leads to accumulation of weapons-grade fissile material ^{233}U in an easily separable form.

TABLE XX Driver Fuel Configuration for the TBR-1 and 2

TBR – 1		**TBR – 2**	
MATERIALS	**WEIGHT PERCENT**	**MATERIALS**	**WEIGHT PERCENT**
O^{16}	9.6%	O^{16}	0%

TABLE XX Continued			
TBR – 1		**TBR – 2**	
U^{235}	3%	U^{235}	3%
U^{238}	67.4%	U^{238}	97%
Th^{232}	20%	Th^{232}	0%

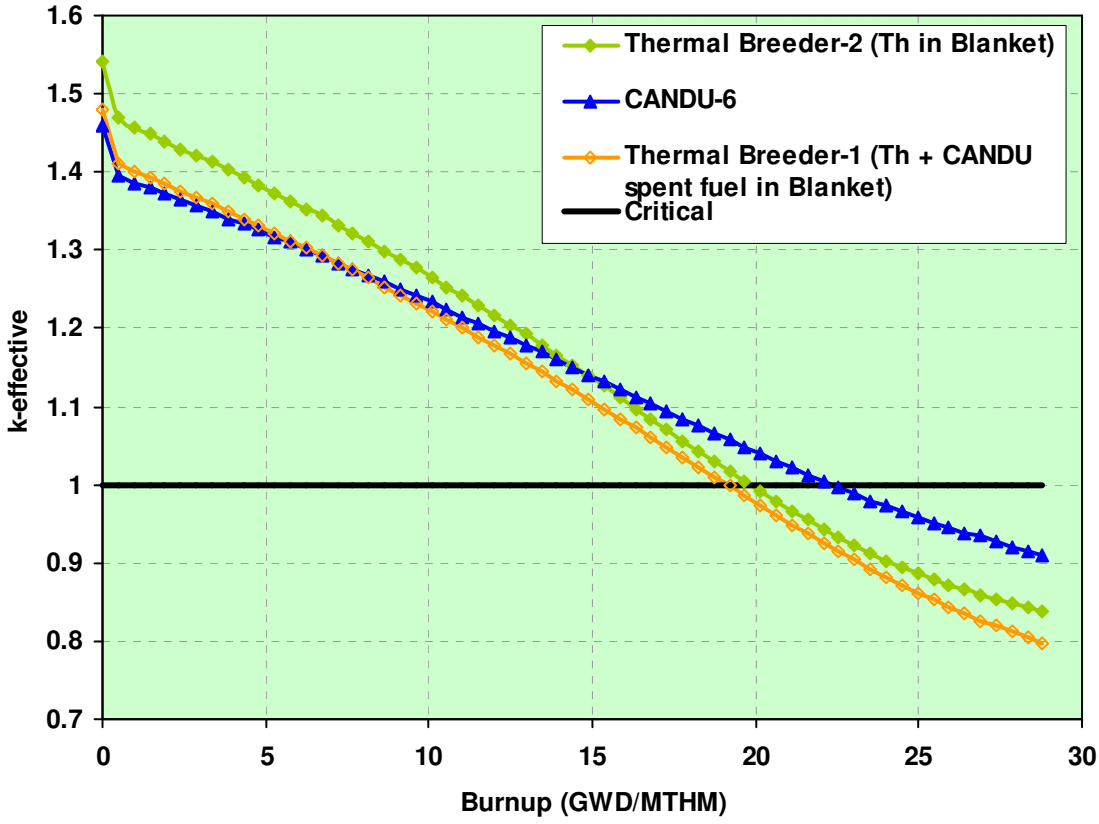

Fig. 7. K- effective versus burnup for thermal breeders and CANDU-6 cores

The steps of simulation, assumptions and modeling criteria involved in achieving the optimized design are as stated:

1. The total quantity of the heavy metal was kept same to that of the modeled CANDU-6 core because of similarity in material properties of the fuel and core size and structure.
2. To start with seven outer layers of the core were filled with blanket fuel assemblies comprising of 10% thorium and 90% spent fuel of PHWRs.
3. The enrichment of the driver fuel was increased from level of natural uranium to 20% ^{235}U in evenly distributed steps of 5%. The limit of 20% was set so as not to exceed the weapons-grade enriched uranium level.
4. The end of life (EOL) fuel composition was then compared against the initial fissile matter for all of the five cases.
5. In the next step thorium percentage was increased from 10% to 70% in steps of 10% for cases with 5% and higher enriched uranium. The minimum limit of 5% ^{235}U was chosen on the basis of achievable burnup being close to that of CANDU-6 core.
6. After comparing the EOC fuel composition for all the cases the region of uranium enrichment of less than 10% and thorium composition of 20% was chosen as the feasible point for the given geometry.
7. Later the driver fuel composition was varied from 3% to 10% ^{235}U with 20% thorium in the driver fuel. For each case the number of blanket assemblies was varied from seven layers down to one layer. The EOC fuel composition results

were found to be of high conversion ratio for two to three layers for all the cases of enriched uranium.

8. The full core model was then simulated with finer levels of uranium enrichment. Finally the fuel composition of 4.25% enrichment thus making it 3% of ^{235}U for the driver fuel core and 20% thorium was concluded to be appropriate for the first cycle of TBR.

TABLE XXI Fuel for Radial Blanket Assemblies of TBR–1

MATERIALS	WEIGHT PERCENT	MATERIALS	WEIGHT PERCENT
U^{234}	0.0000289%	Np^{237}	0.0083050%
U^{235}	0.2456285%	Am^{241}	0.0006869%
U^{236}	0.2085262%	Am^{243}	0.0016323%
U^{238}	79.077783%	Cm^{242}	0.0002466%
Pu^{238}	0.0019075%	Cm^{243}	0.0000032%
Pu^{239}	0.2470208%	Cm^{244}	0.0002342%
Pu^{240}	0.1533233%	Cm^{245}	0.0000033%
Pu^{241}	0.0368484%	Am^{242m}	0.0000078%
Pu^{242}	0.0178140%	Np^{237}	0.0083050%
U^{234}	0.0000289%	Th^{232}	20%

VI.C.3. Results from Simulations

The core model as shown in Figure 8 is designed to have higher moderator to fuel atom ratio to enhance neutron absorption in fuel that occurs at energies below 0.45 eV. Low absorption cross section material like D_2O is suitable for coolant and as moderator because of low fissile content of slightly enriched ^{235}U fuel. Limiting neutron loss due to leakage in thermal breeders is largely a matter of economics. Leakage has been reduced by surrounding the active core with blanket of appropriate thickness. The optimal thickness of the blanket (2-3 assemblies) was attained after balancing the gain of incorporating more assemblies against additional neutrons saved.

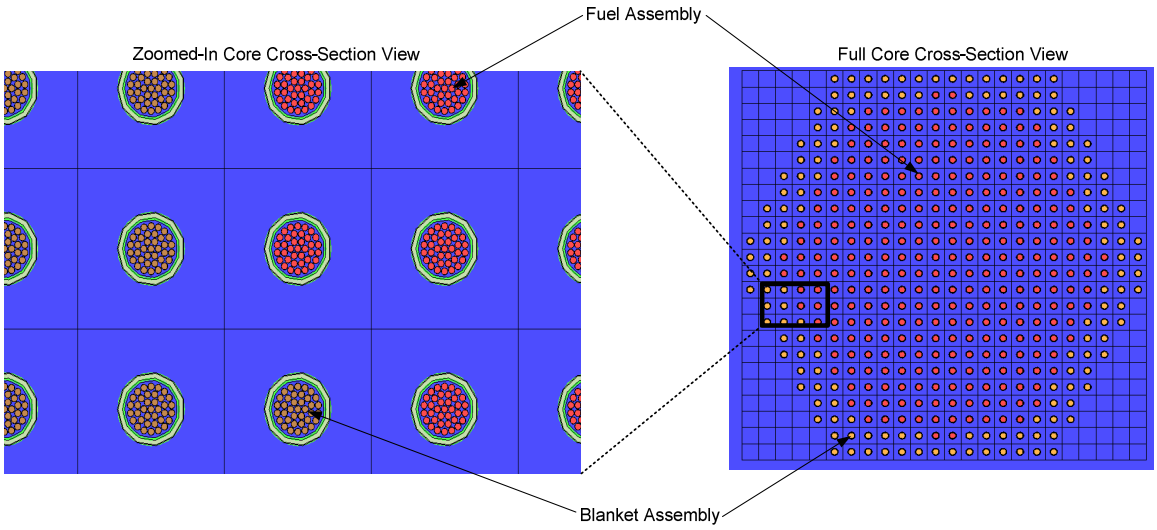

Fig. 8. Cross-section view of the TBR model (KENO 3D)

The blanket assemblies are cooled and moderated with D_2O. The driver fuel and blanket assemblies are surrounded with D_2O reflector both radially and axially. Each driver fuel and radial blanket assembly (as shown in Figure 9) comprises of calandria tube (CT) surrounding a pressure tube (PT) with twelve axially placed cylindrical fuel

bundles inside it. The annular space between the CT and PT is filled with CO_2 for tube failure detection. Each fuel bundle is made of 37 circular pincell, zircolay cladded fuel rods of 50 cm each. The gap between fuel and clad is filled with nitrogen to diminish fuel clad interaction and provide lubrication to the fuel pellets.

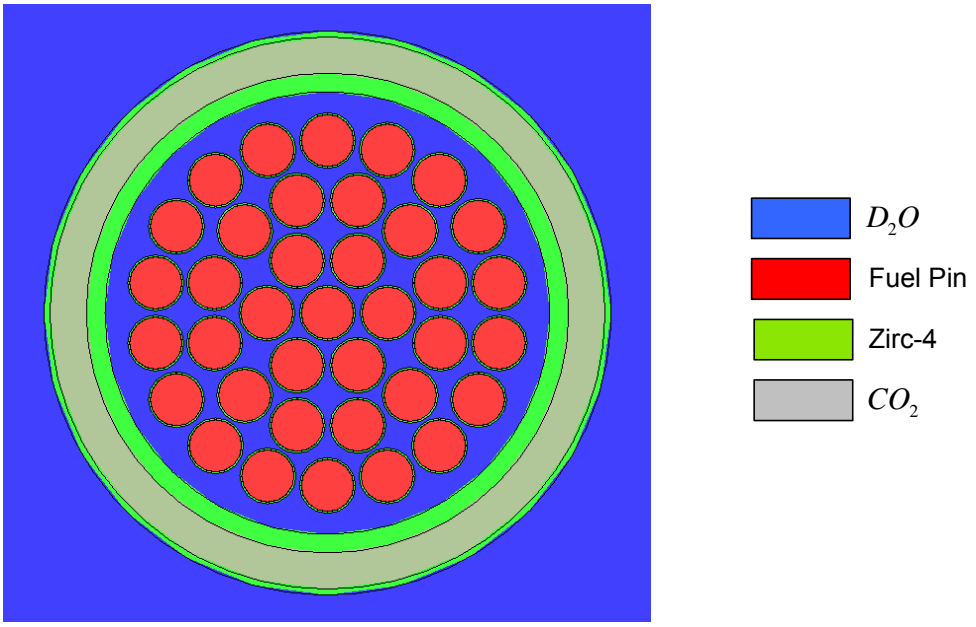

Fig. 9. Top view of the thermal breeder fuel assembly model (KENO 3D)

The modeled thermal breeder has an average reactivity drop of 0.8 mk per day. The core is kept operational by online refueling. To maintain the excess reactivity it needs to be refueled everyday with 10 fresh bundles (shown as in Figure 10). This is subject to the condition that core heat is generated only from the driver fuel assemblies. A detailed refueling sequence for online fueling and blanket bundles replacement would depend on the specific heat distribution of the core. The heat distribution of the core for

blanket and driver assemblies can be altered by incorporating heterogeneity in the core geometry. This change would need detailed analysis with respect to the blanket power, quench time and reflooding sequence during LOCA. Higher percentage of blanket power would also call for intricate reactivity regulation and protection system along with revised refueling sequence.

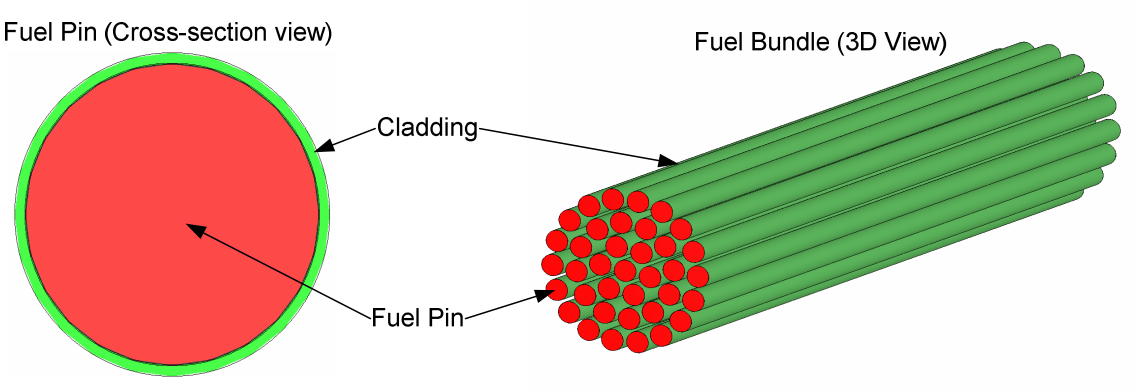

Fig. 10. Image of the thermal breeder fuel pin and 37-pincell bundle (KENO 3D)

Continuous burnup of the fuel results in production of fissile isotopes of the likes of ^{239}Pu, ^{241}Pu and ^{233}U, part of which is lost in the fissioning process and some gets accumulated at the end of average burnup. Material quantities for the driver and radial blanket fuel assemblies at the beginning and end of cycle of the first burnup cycle is given in Table XXII. The spent fuel of the driver and blanket are intermixed and ^{235}U was added to bring the fissile content in the driver fuel back to 3%. Because of the change in fuel isotopics the average burnup reached for the second cycle is different than the first cycle. Repeating the process of irradiation followed by making up for the loss in fissile content, it takes eleven cycles to reach a stage of breeding ratio that is greater than

one. The equilibrium state achieved has a breeding ratio of 1.04 with EOC fuel composition rich in fissionable plutonium isotopes and ^{233}U.

TABLE XXII Fissile Material Accounting for TBR–1

ISOTOPES	**TOTAL DRIVER FUEL CONFIGURATION (KILOGRAMS)**		**TOTAL RADIAL BLANKET FUEL CONFIGURATION (KILOGRAMS)**	
	BOC	EOC	BOC	EOC
^{235}U	2538	589	112	78
^{233}U	0	195	0	34
^{237}Np	0	10	3.77	4.38
^{232}Th	16922	16500	9083	9037
^{238}Pu	0	2	0.88	1.29
^{239}Pu	0	196	112	107
^{240}Pu	0	99	70	78
^{241}Pu	0	27	17	18
^{242}Pu	0	10	8	11
Total Pu	0	529	208	215
Fissile Content	2538	1118	320	327
Fissile Content Equivalent	2538	1238	332	358
Fissile Content Destroyed	--NA--	1948	--NA--	34

As the fuel for the blanket is made available from the accumulated spent fuel, the quantity of material can be judged as freely available to the cycle from its thermal reactor predecessors. This quantity of fissile material consumed by the blanket can also stay out of the calculation because it is has been already paid for. At the end of 585 days the power produced by the blanket is 7.7% and rest by the driver fuel. Fissile content equivalent takes into account the ^{237}Np mass and the fact that ^{241}Pu worth is 1.5 times more than ^{239}Pu because of primarily higher fission cross-section.

So, applying equation (8) for the data of first cycle from Table XXII we have,

FP = 1596 kilograms (1238 + 358) and FD = 1948 kilograms. Thus the conversion ratio for the first cycle of TBR-1 is 0.819. To reach the average burnup it takes 585 days of full-power-day operations and for fissile materials accumulation for one full core it needs to operate for 585 / 0.819 days or more in first stage. This is close to 715 days if operated at 100% capacity factor. Taking a realistic annual capacity factor of 75%, the first cycle of thermal breeder would produce its first initial load of fuel in every 2.6 years. This is though not the doubling time because it has been refueled for than once for each channel. The fuel produced, would be a mix of ^{233}U, ^{235}U and plutonium. After the first cycle of core operation the spent fuel is reprocessed and reloaded in the driver fuel assemblies along with required addition of ^{235}U to bring the fissile content to 3% in the driver part of the core. The process of irradiation and reprocessing reaches an equilibrium state after eleven cycles of reactor operation spanning over 11.14 years. The proportion of plutonium and uranium quantity in the EOC fuel is 2.4% and 1.69% respectively in the equilibrium cycle. The actual doubling

time for reactors in a fuel cycle is dependent on the number of reactors and the rate of reprocessing of the spent fuel. Thus it is concluded from the simulations and above calculations that s series of eleven TBRs built at a rate of one per year can realize an equilibrium breeding fuel cycle in eleven years. The next step in the evaluation of design is the fissile material data interpretation.

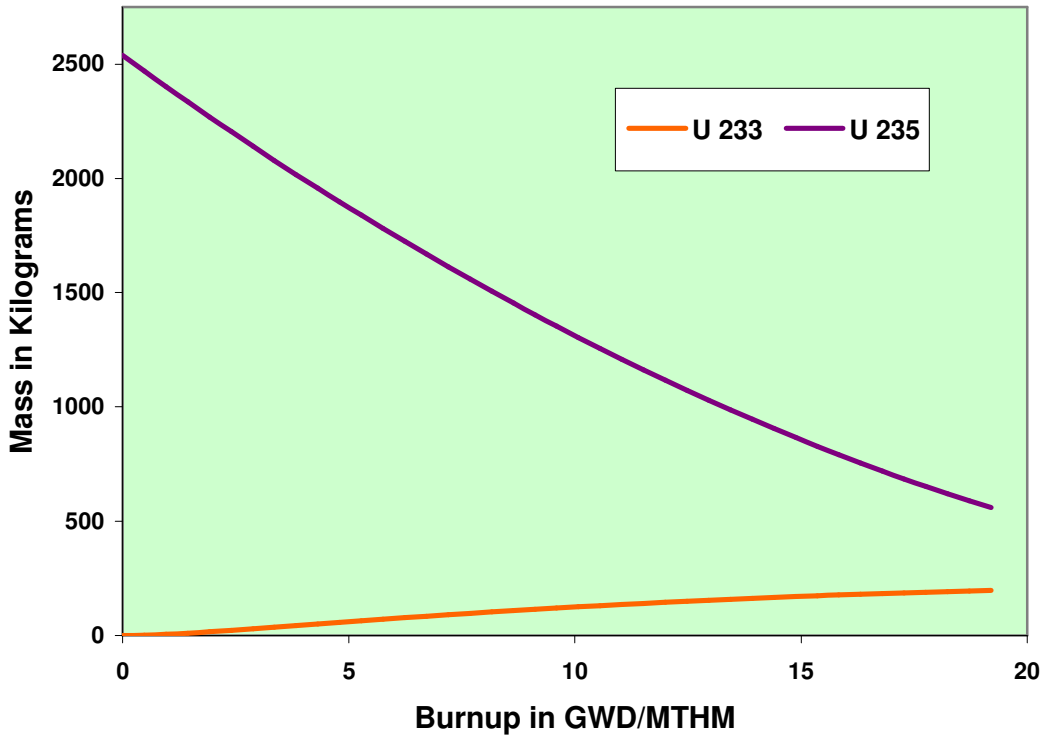

Fig. 11. Plot of uranium isotopes versus burnup for the driver fuel

Figure 11 infers that ^{233}U production is a continuously increasing process and thus there is no gain in lowering the fuel burnup of thermal breeder for maximizing ^{233}U production. The data has been plotted beyond the attained burnup to show the non-drooping trend of ^{233}U before average burnup is reached. But for ^{239}Pu (as shown in

Figure 12), peak is attained close to the average burnup of 18.72 GWd/MTHM. The increasing trend of ^{240}Pu ensures the reactor-grade nature of plutonium and accumulation of fissionable material primary to the establishment of equilibrium state. At the end of the desired average burnup of the driver and blanket fuel, the core comprises of 1445 kilograms of fissile material (^{235}U = 667 kgs, ^{233}U = 229 kgs, ^{239}Pu = 303 kgs & ^{241}Pu = 45 kgs). The plutonium produced, has a ^{240}Pu / ^{239}Pu ratio of 50.5% in the driver fuel. ^{233}U and ^{235}U are at a low percentage of total uranium and is in mix with plutonium, minor actinides and fission products thus making the removal process uneconomical and making the fuel composition proliferation resistant.

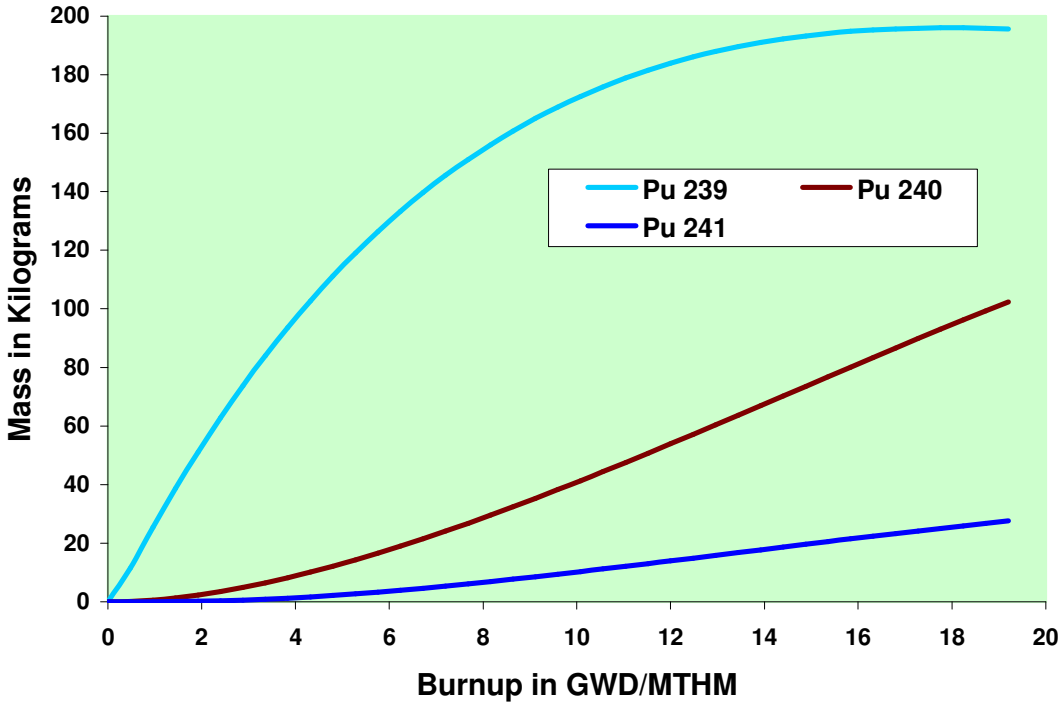

Fig. 12. Plot of plutonium isotopes versus burnup for the driver fuel

Figures 13 & 14 depict the uranium and plutonium isotopes for the fuel configuration irradiated in the blanket of 152 radial blanket assemblies similar in shape and size to that of the driver fuel assemblies. The ^{240}Pu / ^{239}Pu ratio reached is 72.8% in the radial blanket fuel, which makes it far more reactor-grade in nature than the driver fuel plutonium (~50.5%). EOC uranium composition for the equilibrium stage is a mixture of ^{233}U and ^{235}U in the ratio of 2:1.

At the start of first cycle the radial blanket assemblies produced 2.7% (Table XXIII) of the total core heat and they constituted 1/3rd of the total fuel assembly channels. This is due to the fact that the fissile content in the blanket at the beginning of cycle is nominal. The contribution of blanket heat slowly increases with burnup as production of fissile material from the fertile rises. At the end-of-life (585 days) in the first cycle the blanket heat is 7.7% of the total core heat generation. The blanket heat increases over following cycles and finally reaches to 15.6% for the equilibrium cycle.

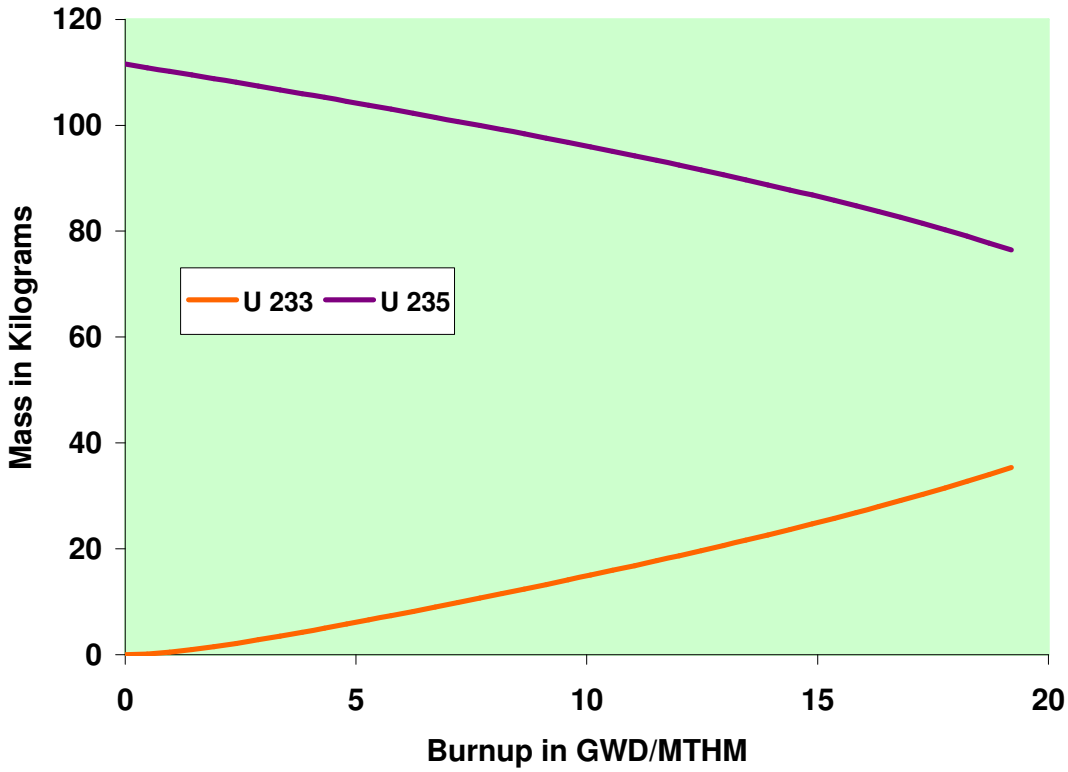

Fig. 13. Plot of uranium isotopes versus burnup for the blanket fuel

As observed there is an asymmetry in the power produced between driver and blanket fuel assemblies. To avoid large coolant temperature asymmetries, proper flow distribution among channels must be maintained. In case of PWR-type thermal breeder reactors the temperature variability is countered by cross flow, which results in excessive depletion of coolant from blanket assemblies and thus degrades the cooling of blanket fuel pins. The solution for this problem is to surround the PWR-type assemblies with can walls thus segregating the sub-assemblies. In case of CANDU reactors this segregation is inherent. And the flow in individual channels is controlled with orifices and ventures.

Coolant mixing of selective channels at the core outlet maintains the uniform temperature distribution across the channels and core as a whole.

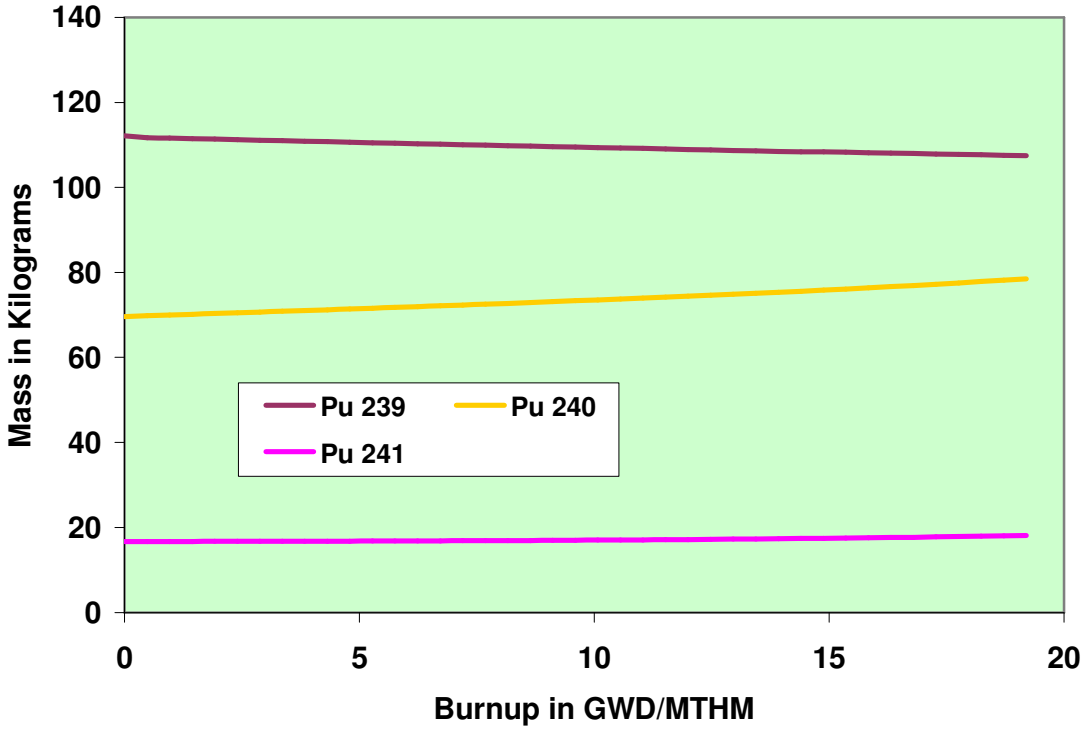

Fig. 14. Plot of plutonium isotopes versus burnup for the blanket fuel

As stated in Table XXIII there is reduction in the total power production for TBR-2 from that of the CANDU-6 core. Furthermore, this core uses more fissile content than CANDU-6 because of huge parasitic absorption by fertile material content that are not efficient fissile material producers. Thus the intermixing of ^{232}Th with the uranium in the driver fuel is not only required from non-proliferation point of view but also an argument for efficient power generating system.

TABLE XXIII Power in Zones for the First Cycle of TBR-1

	TBR - 1	TBR - 2	CANDU - 6
Core	3214 MW_{th}	2427 MW_{th}	2552 MW_{th}
Driver	3127 MW_{th}	2425.4 MW_{th}	--NA--
Blanket	87 MW_{th}	1.6 MW_{th}	--NA--
Percentage of Heat Generated in the Blanket	2.7% - BOC 7.7% - EOC	0.065% - BOC	--NA--

The rate of reactivity loss per day of TBR-1 core is close to that of CANDU-6 fuel, augmenting the fact that this core can be operated with similar regulation and protection control systems. Near equal initial multiplication factor for the TBR-1 and CANDU-6 fresh fuel core makes the reshuffling sequence primarily similar during the pre-refueling phase. In fact the online refueling sequence to maintain excess reactivity can be similar too.

VI.C.4. Optimized Design

In total 304 driver fuel assemblies are surrounded by 152 radial blanket assemblies is chosen as the optimized design with the assumed constraints. There is no heterogeneity introduced in the lattice pitch dimensions, fuel geometry and axial arrangement of fuel bundles (Stated in Table XXIV). The power density of the thermal breeder fuel is similar to the CANDU fuel at 32 watts of thermal power per gram of

heavy metal so as not to alter the pumping power of the coolant for heat removal. Because of the similarity in lattice pitch dimensions and power density of TBR with CANDU-type core, the quench time and reflooding sequence during LOCA are assumed to be of similar nature.

TABLE XXIV Dimensions, Initial Reactivity and Refueling Sequence

DESCRIPTIONS	**TBR - 1**	**CANDU-6**
Fuel Pin Radius	0.6077 cms	0.6077 cms
Fuel Clad Gap Thickness	0.0043 cms	0.0043 cms
Clad Thickness	0.042 cms	0.042 cms
Pincells in a Bundle	37	37
Fuel Bundle Length	50 cms	50 cms
Pressure Tube Inner Radius	5.1889 cms	5.1889 cms
Pressure Tube Outer Radius	5.6032 cms	5.6032 cms
Calandria Tube Inner Radius	6.4478 cms	6.4478 cms
Calandria Tube Outer Radius	6.5875 cms	6.5875 cms
Assembly Channel Pitch	28.575 cms	28.575 cms
Each Assembly Channel has	12 Fuel Bundles	12 Fuel Bundles

TABLE XXIVContinued		
DESCRIPTIONS	**TBR - 1**	**CANDU-6**
Number of Driver Fuel Assembly Channels	304	320
Number of Blanket Assembly Channels	152	--NA--
Number of Fuel Reshuffling Days	533 (first cycle)	628
Average Burn Up (GWD/MTHM)	18.72 (first cycle)	22.08
Days to Reach Average Burnup	585 (first cycle) 306 (equilibrium cycle)	690
Initial K-effective	1.47921 (first cycle)	1.45922
Power Density in Watts/Grams	32	32
Coolant & Moderator	D_2O	D_2O
Reactivity Loss per Day	0.803 milli-k (highest of all cycles)	0.649 milli-k

D_2O at low temperature and pressure is used as the reflector to surround the radial blanket assemblies. There is no physical boundary between the reflector D_2O and the moderator. The moderator cooling and purification system can be designed to meet the extra D_2O volume from the reflector. Larger pitch length, similar to that of a CANDU-6 core improves the moderator to fuel atom ratio compared to that of the fast

breeder reactors. The neutron leakage from the driver fuel assemblies has been taken advantage of by radial blanket assemblies and surrounding reflector, which improves the production of fissionable material and ^{233}U quantity. An important process following the fissile material production is spent fuel reprocessing. Handling of spent fuel is intricate because of the additional radiological hazards involved in the bred fuel comprising of ^{228}Th.

Generation of harmful nuclide ^{228}Th is of major concern because of its high radio-toxicity and decay to ^{212}Bi and ^{208}Tl nuclides which emits high energy gamma rays. ^{228}Th is generated by the following decay and burnup chain.

$$^{232}_{90}Th \xrightarrow[(n,2n)]{} {}^{231}_{90}Th \xrightarrow[(\beta^{-1})]{} {}^{231}_{91}Pa \xrightarrow[\sigma(n)]{} {}^{232}_{91}Pa \xrightarrow[(\beta^{-1})]{} {}^{232}_{92}U \xrightarrow[({}_2\alpha^4)]{} {}^{228}_{90}Th$$

This isotope of thorium (Th^{228}) can either be lost by decay to elements which are of high energy gamma emitters or by neutron capture of its predecessor ^{232}U. The two paths are as below.

$$^{228}_{90}Th \xrightarrow{(\lambda)} \underbrace{^{212}_{83}Bi}_{(\gamma emitter)} \xrightarrow[({}_2\alpha^4)]{} \underbrace{^{208}_{81}Tl}_{(\gamma emitter)}$$

$$^{232}_{92}U \xrightarrow[(\sigma(n))]{} {}^{233}_{92}U$$

The decay to ^{212}Bi and ^{208}Tl nuclides is radiation hazard problem because of hard gamma emitting daughter elements. And for absorption of neutron in ^{232}U to convert it to ^{233}U desires high neutron flux. In a fast breeder the fissile material content is large enough to provide high neutron flux and that would enhance to neutron absorption. But fast neutrons also enhance n, 2n reactions in ^{232}Th, which is the cause for ^{228}Th in the first place. In thermal breeders the large moderator to fuel volume ratio and small fuel

rod radius slows down the fast neutrons effectively and thus the n, 2n reactions of ^{232}Th drops drastically leading to decreased producing of ^{232}U. The modeled TBR-1 produces 1.51 grams of ^{228}Th in the driver fuel and meager 0.0266 grams in the blanket fuel. This fuel configuration would not raise the radiation hazard levels in reprocessing compared to that of the existing spent fuel from thermal reactors. The TBR-1 design with the optimized fuel configuration is economically beneficial and proliferation resistant because of its higher η value (as shown in Table XXV), reactor-grade nature of the plutonium, low fissile content uranium isotopes, high burnup and refueling sequence similar to the existing CANDU reactors.

TABLE XXV Beginning of Life α and η Coefficients

	TBR - 1	**TBR - 2**	**CANDU - 6**
α for Driver	0.067067	0.067434	0.0675844
α for Blanket	0.050618	Blanket does not have fissile content	Blanket does not exist
α for the Core	0.064526	0.067434	0.0675844
$\eta = \frac{\nu}{1+\alpha}$	2.309	2.295	2.295

Figure 15 shows the flux at the beginning and end-of-life in the first cycle for TBR-1 core. The flux increases when fuel assembly is considered as control volume instead of fuel rod because of increased moderator to fuel atom ratio.

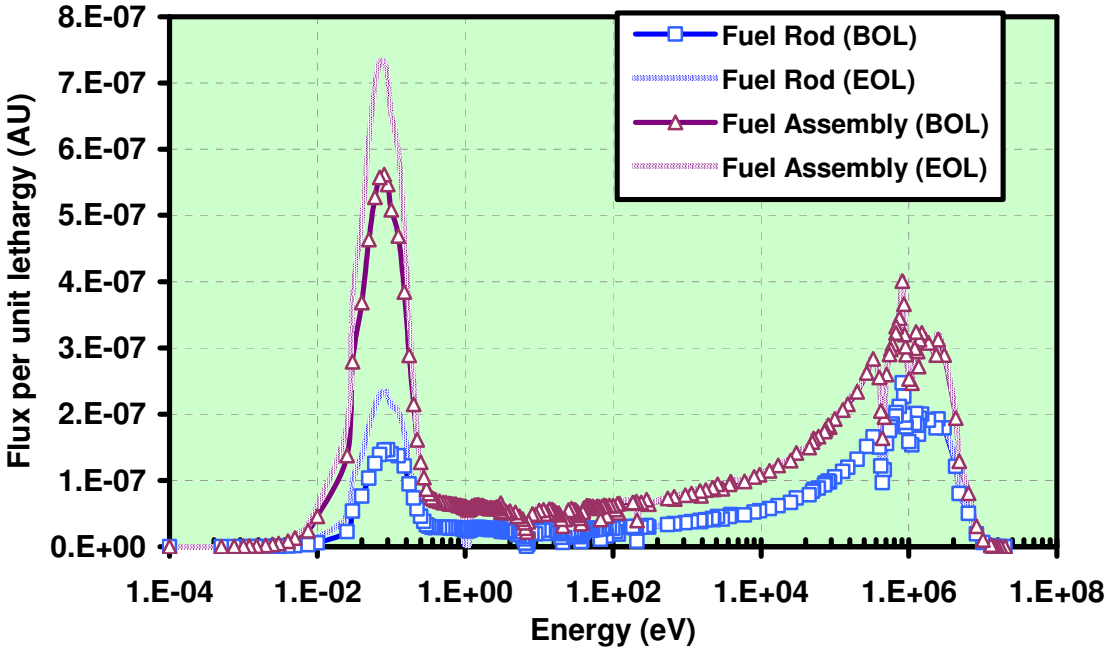

Fig. 15. Beginning and end of cycle flux pattern for TBR-1

VI.D. Implementation of Alternate Reactor System in Indian Fuel Cycle

In the outcome of U.S.-India nuclear agreement a nuclear power program that addresses the verification and monitoring requirements of an international collaboration needs to be formulated. This program is supposed to serve the purpose of effective use of resources under the realms of proliferation concerns as an alternate to the presently

pursued three-stage-power-program. The proposed alternate nuclear reactor system is based on the following objectives:

- Optimization of electricity production,
- Minimization of international uranium use,
- Maximizing proliferation resistance and
- Minimization of domestic uranium use

A comparative study of the alternate fuel cycle to that of the present three-stage-program is analyzed in this section. It was observed that the nuclear power program that better meets the objectives on the time frame of 2030 would serve the long term energy needs of India. International collaboration being an important feature in meeting the energy needs of India also simplifies the objectives of the alternate reactor system.

VI.D.1. Description of Alternate Future Fuel Cycle

The suggested model of nuclear power program with thermal breeders in next stage to PHWRs can appropriately fit to the nuclear fuel cycle of nations like India having CANDU spent fuel along with thorium reserves. The handling of end of cycle fissile content of TBRs is less of a challenge because of negligible presence of ^{228}Th in the bred fuel. As stated in earlier section ^{228}Th decays to two daughter elements that are hard gamma emitters and are nightmare for reprocessing. To start with, the driver fuel for this core is made from internationally supplied low enriched uranium and by default comes under monitoring and verification process. The reactor system under safeguards and verifications involving the international low enriched uranium and domestic thorium

reserves assures life time fuel supplies. The mixing of thorium in the driver fuel has been studied and concluded in previous section that the EOC spent fuel configuration constraints proliferation (because of low percentage of fissile uranium isotopes (^{233}U and ^{235}U) and reactor-grade nature of the produced plutonium) and economical (high burnup, uses existing reactor geometry and blanket fuel is in store). The alternate reactor system (shown in Figure 16) suitably engulfs the necessities of the fuel cycle that are looking for breeder systems for exploiting thorium and have tons of spent fuel from thermal reactors. By not incorporating any reprocessed plutonium in the fresh core driver fuel, the system undermines the necessity of reprocessing the thermal reactor spent fuel of the first-stage for extraction of plutonium. This reprocessing step between the PHWR & TBR is thus devoid of explicit plutonium production.

India's proposed nuclear power production strategy (shown in Figure 16) can draw parallels with the presently pursued three-stage-program in terms of meeting the energy demands of the country. The alternate reactor system is also a step forward with regard to safeguards of reprocessing facilities. There can be fuel handling and reprocessing facilities explicitly dedicated to the proposed alternate reactor systems. The two positive aspects in the power program are well accounted low enriched uranium from international suppliers and easily verifiable thorium supply to breeder cycle which makes the international cooperation feasible and sustainable on a long run.

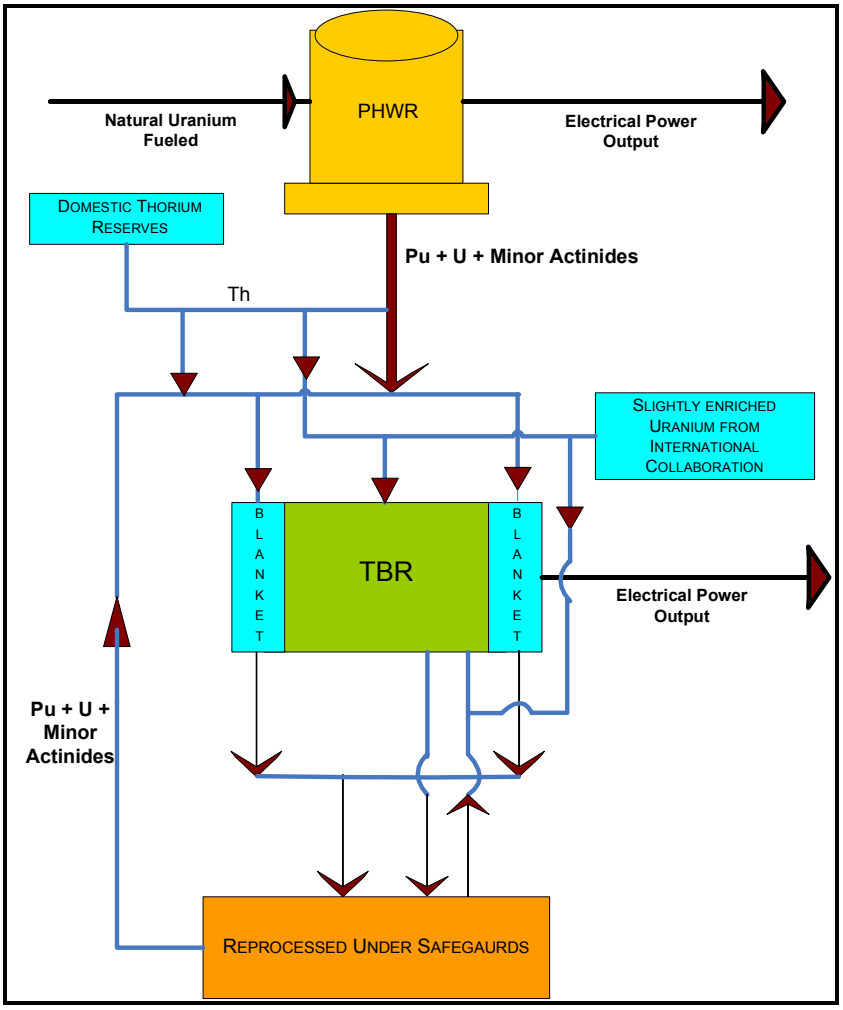

Fig. 16. India's proposed nuclear power production strategy

VI.D.2. Metrics of Interest

Developing a fuel composition in a given geometry (CANDU-type in this study) with optimized distribution of driver and blanket assemblies was the first step in realizing the metrics of objectives for the modeled alternate reactor system. The basis of the metrics is depicted in the preamble of the section. The first row of the metrics in Table XXVI shows the total uranium utilized per unit of heat generated. As can be seen

the blanket fuel EOC composition has lesser uranium utilization from that of the driver fuel. This is because of lower fissile content of uranium and contribution of fission heat from plutonium and minor actinides. Thus the substantial contribution of heat generated makes the blanket close to a waste transmuter.

The blanket fuel has high ^{240}Pu / ^{239}Pu ratio and irradiation in the core makes the ratio further higher. Total plutonium in driver and blanket fuel has a large share of ^{240}Pu thus decreasing the proliferation concerns. Another concern being fissile uranium content which is also far from the weapons-grade limit (WgU implies (^{233}U + ^{235}U) / U > 20%).

TABLE XXVI Metrics for Assessment of Proposed Nuclear Power Program

Metrics of Objectives	**Driver Fuel**		**Blanket Fuel**	
	BOL	EOL	BOL	EOL
Total U / GWd	--NA--	17.84 kgs/GWd	--NA--	11.5 kgs/GWd
^{240}Pu / ^{239}Pu	0%	50.5%	62.5%	72.8%
Total Pu	0 kgs	529 kgs	208 kgs	215 kgs
(^{233}U+^{235}U) / U	4.25%	1.31%	0.31%	0.31%
^{240}Pu	0 kgs	99 kgs	70 kgs	78 kgs

VI.D.3. Comparisons to Existing Indian Fuel Cycle

Energy output of thermal breeder reactor system was calculated with the assumption of 30% overall plant cycle efficiency and 75% operational capacity factor for each power plant. As modeled in SCALE5.1 the input deck for TBR-I core has thermal output of 3214 MWth. This thermal heat content in association with the above stated assumptions would produce approximately 725 MWe.

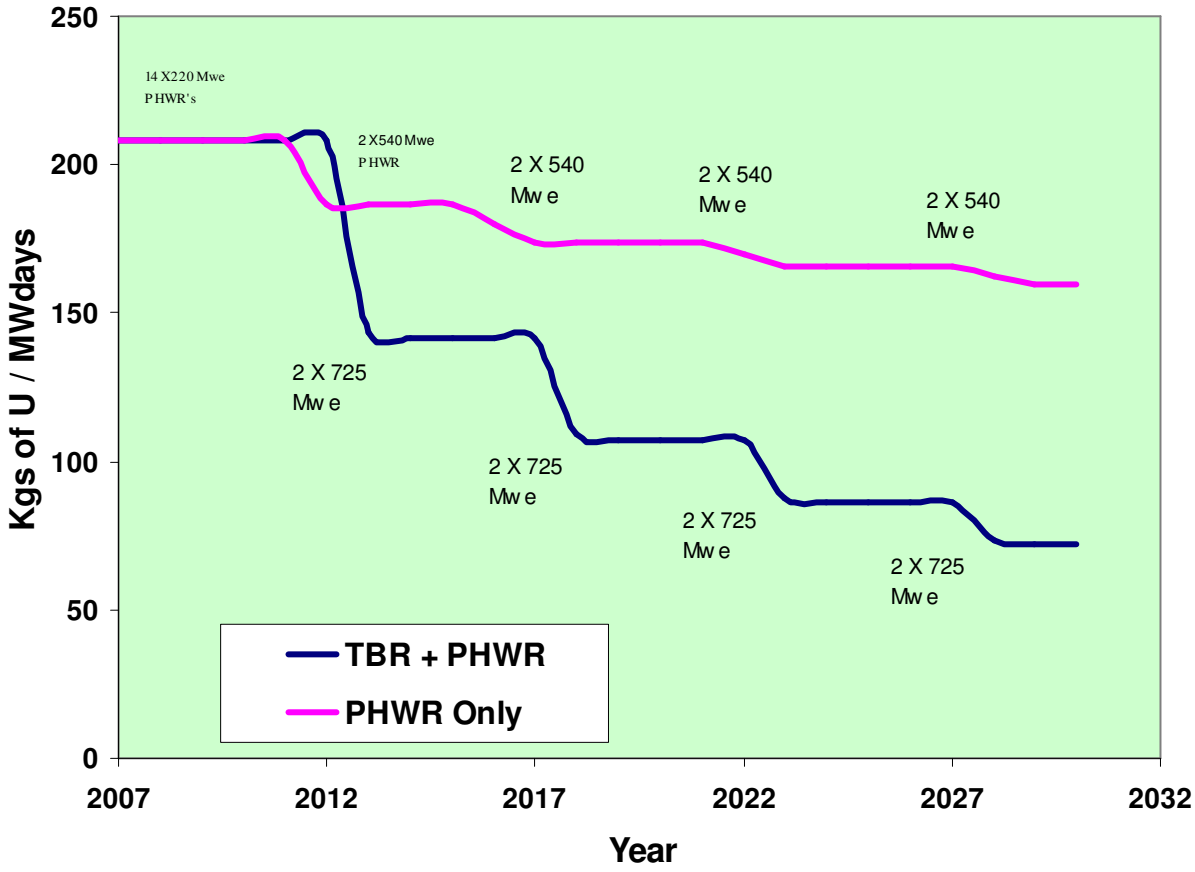

Fig. 17. Comparison for uranium utilization in electricity generation

On comparison with present PHWRs and under construction PFBR this output is double fold (500 MWe PHWR at 75% capacity factor = 375 MWe). Because of the similarity in design to CANDU-6 core, the TBR power plant would have coincident secondary and tertiary loops. The proposed power projects are suggested on the lines of the time frame of planned PHWR projects. Representing these projects on the basis of the metric of total uranium / GWd in Figure 17 suggests that there is a substantial gain in savings on the quantity of uranium required by the safeguarded TBR reactors.

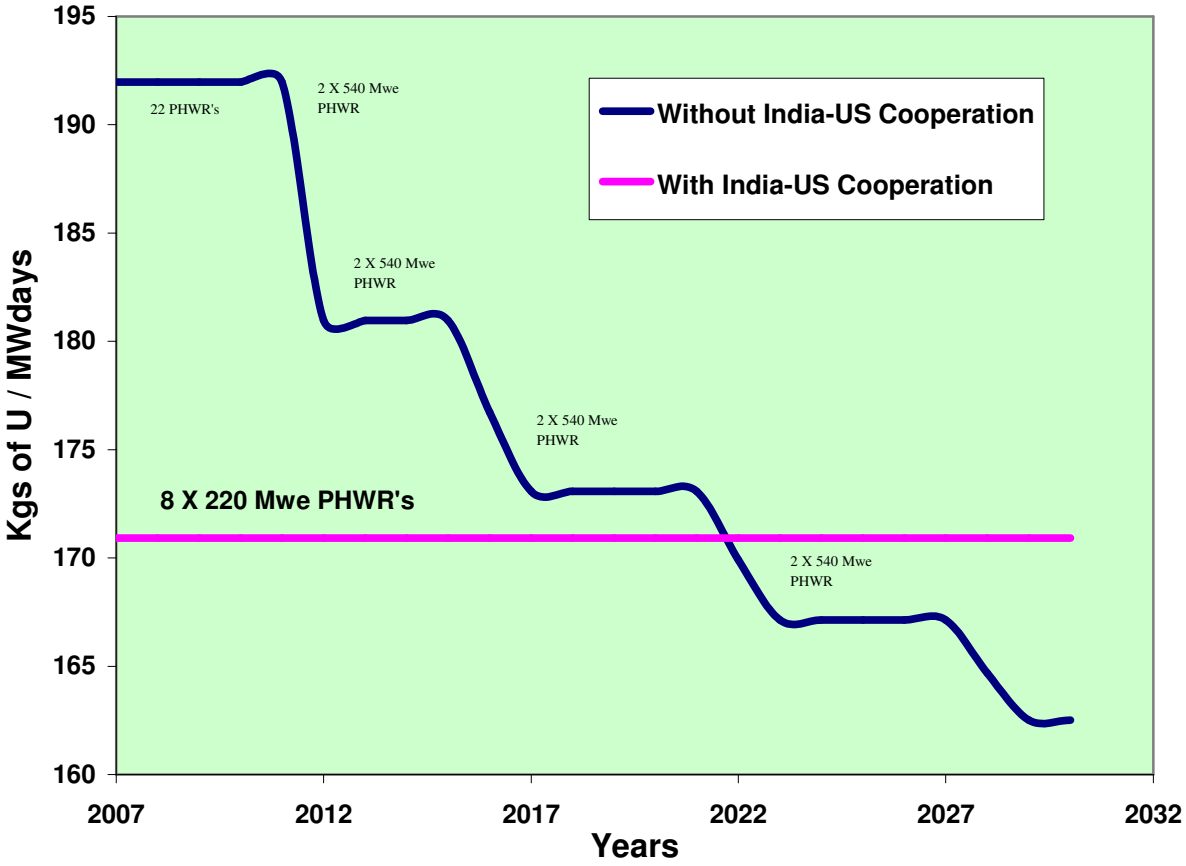

Fig. 18. Comparison of domestic uranium use with/without cooperation

Starting with fourteen PHWRs all the future reactors would be under safeguards. If the pertaining plan is replaced with TBR power program then beginning with early 2012 and extending till 2030 and beyond there is an advantage in electricity production from unit fissile content. Under the U.S.-India civilian nuclear cooperation India would have to supply fuel to 8 PHWRs, where as without the agreement the domestic uranium reserves are required for all the 22 reactors along with future PHWRs. Figure 18 shows the break even point of lowering the use of domestic uranium per unit of electricity production being 2021 against the planned construction of PHWRs. This phenomenon shifts to as early as 2012 with the proposed TBR power program (as shown in Figure 17).

VII. CONCLUSIONS AND RECOMMENDATIONS

India as an ambitious nuclear power seeking nation has to a large extent mastered the components of nuclear fuel cycle with far reaching consequences in both civilian and military sectors. It started as the loudest proponent of non-proliferation of nuclear technology but ended up as a non-signatory of NPT on the grounds of disparity. There were many occasions in the past when it was tried to bring Indian nuclear sector under safeguards. U.S.-India nuclear cooperation agreement was the closet to a credible and sustainable non-proliferation initiative in recent times.

With this dissertation study an effort has been made to study all the components of the nuclear fuel cycle and scale the detailed implications of international cooperation on the weapon and energy programs. The weapon program does not seem to be deterred by the U.S.-India civilian nuclear cooperation as the present rate of plutonium production can be sustained with the available uranium reserves and production reactors.

This agreement makes less number of reactors available for weapons-grade plutonium production. The U.S.-India cooperation agreement definitely decreases the latency capabilities for production of weapons-grade plutonium from that of the scenario of failed agreement. Furthermore the reactor constraint of only 8 unsafeguarded reactors even decreases the reactor-grade plutonium production by 4.1% from that of the natural uranium material constraint.

On the energy front, India would have to operate its commercial PHWRs at a lower capacity factor without the agreement. This phenomenon would increase the quantity of plutonium produced from the present levels. Further increase in plutonium

production by lower capacity factor of PHWRs or a newer facility leads to excessive accumulation and a concern for South-East Asia arms race.

A small but significant component of India's fuel cycle is enrichment facility. Need of enriched uranium has its use in nuclear submarine core, thermonuclear device and as reactor fuel. The constraints to the enrichment desires is not the natural uranium feed but the technological hurdles. Nuclear submarine reactor fuel requirement is not dependent on the U.S.-India cooperation agreement because of the meager amount of uranium required [also concluded in reference 56]. It's the technical expertise of modeling and building one is the call of the day.

The effort to develop and test thermonuclear device in 1998 Pokhran-II test was a motive as well as a compulsion to free itself from dependence on large piles of enriched uranium or plutonium to have a credible deterrence. There is an underlying urge to test again because of not so satisfactory results from the test, calling for improved and reliable design [also concluded in reference 57].

Breeders are the route which India sees as the building block for vibrant energy program and also for setting the stage for its third-stage of the nuclear power program [26]. The third-stage reactors are to be fuelled with ^{233}U obtained from the thorium breeding in the fast breeder reactors of second-stage. As evident there is strong opposition to include the breeder program in the U.S.-India civilian nuclear cooperation agreement. The inclusion of FBRs is seen as an effort to cap India's breeder program and eventually diminish it to levels of no significance. Facts suggest that fast breeder reactor design and operation are not new to the world [58]. Fast breeders do not exist in

large numbers, purely because of poor economic returns as compared to available nuclear energy alternatives. India's interest in fast breeder program is because of its vast reserves of thorium and availability of ample plutonium from spent fuel [59].

Through this dissertation an alternative to fast breeders has been suggested. A power program with PHWRs followed by thermal breeder reactors constraints proliferation risks, lowers uranium use and establishes thorium breeding cycle. Unlike the FBR the modeled thermal breeder makes reactor-grade plutonium, does not impart huge radiation risk from spent fuel because of lower quantities of ^{228}Th, uses existing core geometry and does not involve the sodium-water heat transfer phase.

The suggested thermal breeder design appropriately fits to the nuclear fuel cycle of the nations having CANDU reactor experience and thorium reserves. The EOC fissile content is easy to recover because of negligible production of ^{228}Th. The fuel configuration is proliferation resistant because of lower percentage of ^{233}U and ^{235}U in the fuel and reactor-grade nature of plutonium. The reactor can be brought under safeguards and international monitoring system by involving the international supply of low enriched uranium. Presence of thorium fuel fits the necessities of the fuel cycle that are looking for breeder systems for exploiting the domestic reserves of thorium. By not incorporating any plutonium in the driver fuel, this system undermines the need for reprocessing of thermal reactor spent fuel for extraction of plutonium.

On the long run the TBR cycle stands out to be better positioned with regards to the quantity of uranium used per unit of electricity generated. This cycle can partially be

implemented for enhancing the burnup on the EOC fissile content of the CANDU spent fuel and attaining breeder equilibrium state.

Through this study a feasible nuclear power program has been suggested to that of the presently pursued three-stage-power program. A viable international nuclear collaboration can be established on the basis of safeguards and verification methods. The closely guarded fuel cycle of India has large scope of improvement by integrating it to the international domain and the objective of military and civilian nuclear sector separation can be more explicitly achieved with proposed thermal breeder nuclear power program.

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APPENDIX A

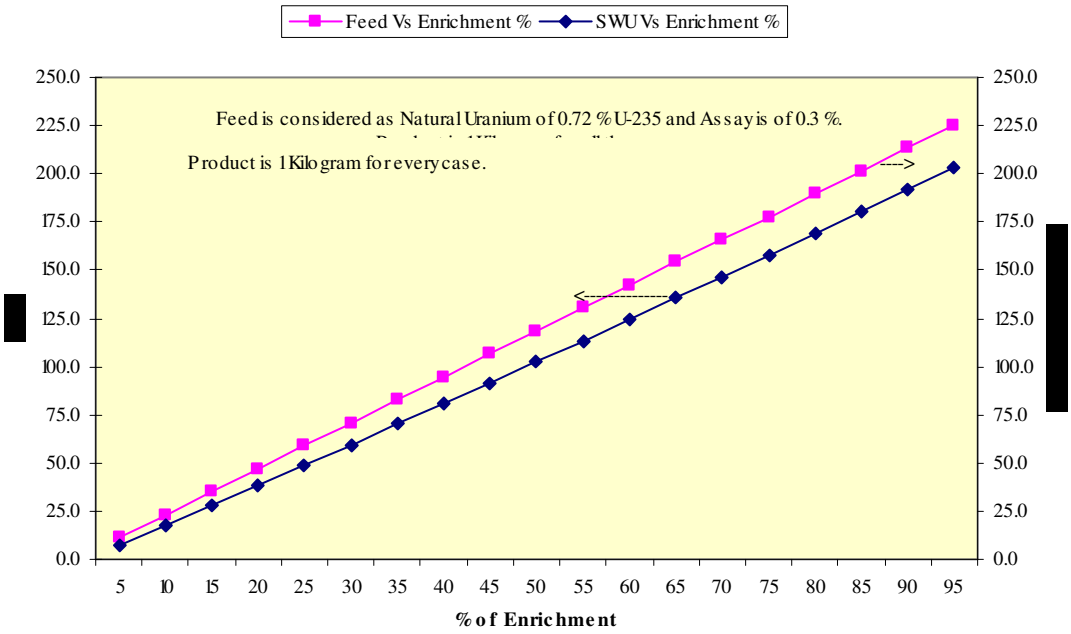

Fig. 19. Plot of feed and SWU capacity for different levels of enrichment

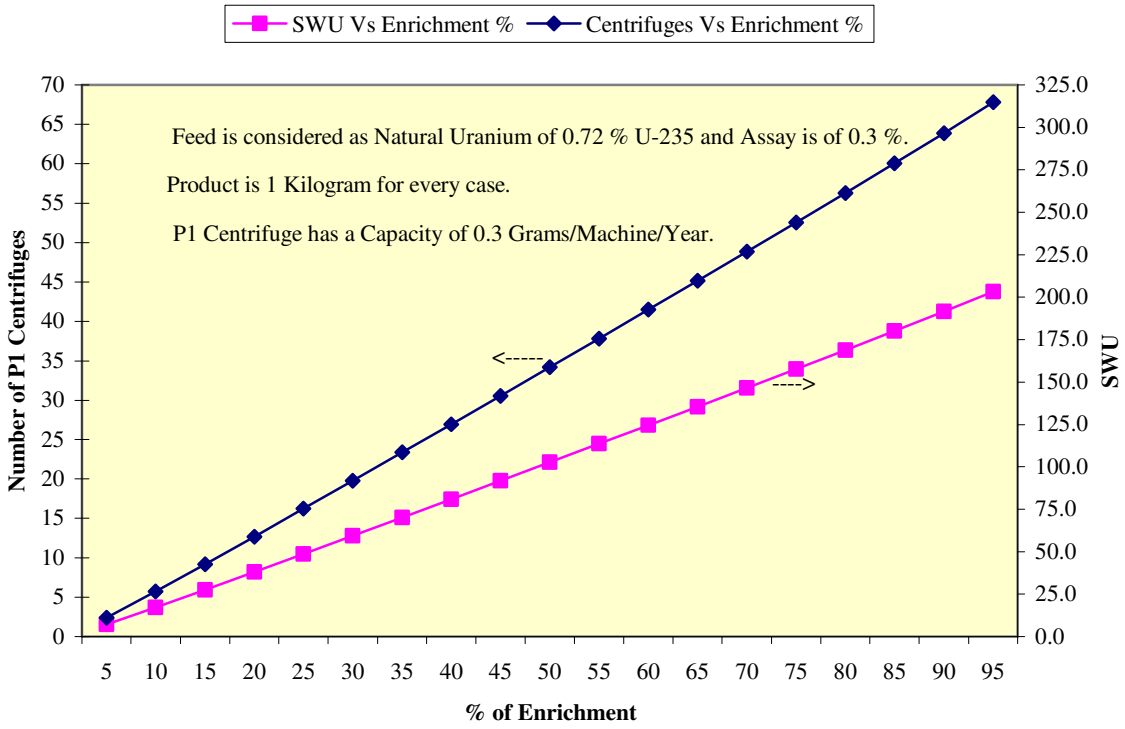

Fig. 20. Plot of centrifuges and SWU capacity at different levels of enrichment

APPENDIX B

CANDU-6 SCALE5.1 Criticality Input Deck

```
'Input generated by GeeWiz SCALE 5.1 Compiled on November 9, 2006
'batch_args \-x\-p\-m
=csas25 parm=(centrm)
thermal breeder
v6-238
read composition
 uo2         1 1 773
                                 92235 2
                                 92238 98   end
 d2o         2 1 523   end
 d2o         3 1 333   end
 zirc4       4 1 523   end
 zirc4       5 1 333   end
 d2o         6 1 523   end
 zirc4       7 1 523   end
 n           8 den=0.000129 1 523   end
 atomco2         9  0.000125  2
                                 6000 1
                                 8016 2
                         1 523   end
 d2o         11 1 303   end
 d2o         12 1 523   end
 zirc4       13 1 523   end
 d2o         14 1 523   end
 zirc4       15 1 523   end
 n           16 den=0.000129 1 523   end
end composition
read celldata
  latticecell triangpitch fuelr=0.6077 1 gapr=0.612 8 cladr=0.654 4
hpitch=1.408 12 end
    centrm data  iim=50  iup=20     end centrm
end celldata
read parameter
 gen=450
 npg=5000
 nsk=50
 flx=yes
 htm=yes
end parameter
read geometry
unit 1
com='u fuel rod'
 zcylinder 1 1  0.6077      600        0
 zcylinder 8 1   0.612      600        0
 zcylinder 4 1   0.654      600        0
unit 2
com='u fuel bundle'
```

```
 zcylinder 12 1  5.1689      700        100
  hole 1       0        0      100
  hole 1   1.408        0      100
  hole 1  -1.408        0      100
  hole 1   0.704  1.219364      100
  hole 1   0.704  -1.219364      100
  hole 1  -0.704  1.219364      100
  hole 1  -0.704  -1.219364      100
  hole 1 2.623455  0.702953      100
  hole 1 -2.623455  0.702953      100
  hole 1 2.623455  -0.702953      100
  hole 1 -2.623455  -0.702953      100
  hole 1 1.920502  1.920502      100
  hole 1 1.920502  -1.920502      100
  hole 1 -1.920502  1.920502      100
  hole 1 -1.920502  -1.920502      100
  hole 1 0.702953  2.623455      100
  hole 1 -0.702953  2.623455      100
  hole 1 0.702953  -2.623455      100
  hole 1 -0.702953  -2.623455      100
  hole 1 3.987487  0.703101      100
  hole 1 3.987487  -0.703101      100
  hole 1 -3.987487  0.703101      100
  hole 1 -3.987487  -0.703101      100
  hole 1       0    4.029      100
  hole 1       0   -4.029      100
  hole 1 3.506537   2.0245      100
  hole 1 3.506537  -2.0245      100
  hole 1 -3.506537   2.0245      100
  hole 1 -3.506537  -2.0245      100
  hole 1 2.602647  3.101714      100
  hole 1 2.602647  -3.101714      100
  hole 1 -2.602647  3.101714      100
  hole 1 -2.602647  -3.101714      100
  hole 1 1.38484  3.804815      100
  hole 1 1.38484  -3.804815      100
  hole 1 -1.38484  3.804815      100
  hole 1 -1.38484  -3.804815      100
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 5
com='outer shell h2o '
 cuboid 11 1  14.2875  -14.2875  14.2875  -14.2875      800        0
global unit 6
com='core shell'
  array 1    -324     -324        0
 zcylinder 11 1     550      850      -50
end geometry
read array
ara=1 nux=24 nuy=24 nuz=1
```

```
 com=''
 fill
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5 5
    5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5
    5 5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
end fill
end array
end data
end
```

APPENDIX C

CANDU-6 SCALE5.1 Depletion Input Deck

```
'Input generated by GeeWiz SCALE 5.1 Compiled on November 9, 2006
'batch_args \-x\-p\-m
=t5-depl parm=(centrm)
thermal breeder
v6-238
read composition
 uo2         1 1 773
                                92235 2
                                92238 98   end
 d2o         2 1 523   end
 d2o         3 1 333   end
 zirc4       4 1 523   end
 zirc4       5 1 333   end
 d2o         6 1 523   end
 zirc4       7 1 523   end
 n           8 den=0.000129 1 523   end
 atomco2         9  0.000125  2
                                6000 1
                                8016 2
                        1 523   end
 d2o         11 1 303   end
 d2o         12 1 523   end
 zirc4       13 1 523   end
 d2o         14 1 523   end
 zirc4       15 1 523   end
 n           16 den=0.000129 1 523   end
end composition
read celldata
  latticecell triangpitch fuelr=0.6077 1 gapr=0.612 8 cladr=0.654 4
hpitch=1.408 12 end
    centrm data  iim=50  iup=20     end centrm
end celldata
read depletion
  1
end depletion
read burndata
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
```

```
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
end burndata
read opus
  symnuc=u-233 u-234 u-235 u-236 u-238 pu-238 pu-239 pu-240 pu-241
pu-242 np-237 am-241 am-243 cm-242 cm-243 cs-134 cs-137 nd-143 nd-144
nd-145 nd-146 cm-244 cm-245 cm-246 cm-247 ru-106 am-242m th-232 end
  units=grams
  time=years
  matl=0 1 end
end opus
read model
read parameter
 gen=450
 npg=5000
 nsk=50
 flx=yes
 htm=yes
end parameter
read geometry
unit 1
com='u fuel rod'
 zcylinder 1 1  0.6077      600        0
 zcylinder 8 1   0.612      600        0
 zcylinder 4 1   0.654      600        0
unit 2
com='u fuel bundle'
 zcylinder 12 1  5.1689      700        100
  hole 1       0        0      100
  hole 1   1.408        0      100
  hole 1  -1.408        0      100
  hole 1   0.704  1.219364      100
  hole 1   0.704  -1.219364      100
  hole 1  -0.704  1.219364      100
  hole 1  -0.704  -1.219364      100
  hole 1 2.623455  0.702953      100
```

```
  hole 1 -2.623455  0.702953      100
  hole 1 2.623455  -0.702953      100
  hole 1 -2.623455  -0.702953      100
  hole 1 1.920502  1.920502      100
  hole 1 1.920502  -1.920502      100
  hole 1 -1.920502  1.920502      100
  hole 1 -1.920502  -1.920502      100
  hole 1 0.702953  2.623455      100
  hole 1 -0.702953  2.623455      100
  hole 1 0.702953  -2.623455      100
  hole 1 -0.702953  -2.623455      100
  hole 1 3.987487  0.703101      100
  hole 1 3.987487  -0.703101      100
  hole 1 -3.987487  0.703101      100
  hole 1 -3.987487  -0.703101      100
  hole 1       0    4.029      100
  hole 1       0   -4.029      100
  hole 1 3.506537   2.0245      100
  hole 1 3.506537  -2.0245      100
  hole 1 -3.506537   2.0245      100
  hole 1 -3.506537  -2.0245      100
  hole 1 2.602647  3.101714      100
  hole 1 2.602647  -3.101714      100
  hole 1 -2.602647  3.101714      100
  hole 1 -2.602647  -3.101714      100
  hole 1 1.38484  3.804815      100
  hole 1 1.38484  -3.804815      100
  hole 1 -1.38484  3.804815      100
  hole 1 -1.38484  -3.804815      100
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 5
com='outer shell h2o '
 cuboid 11 1  14.2875  -14.2875  14.2875  -14.2875      800        0
global unit 6
com='core shell'
  array 1    -324     -324        0
 zcylinder 11 1     550      850      -50
end geometry
read array
ara=1 nux=24 nuy=24 nuz=1
 com=''
 fill
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5 5
    5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
```

```
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5
    5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5
    5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5
    5 5 5 5 5 5 5 2 2 2 2 2 2 2 2 2 2 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
    5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5 5
 end fill
end array
end data
end model
end
```

APPENDIX D

Thermal Breeder Core-1 SCALE5.1 Criticality Input Deck

```
'Input generated by GeeWiz SCALE 5.1 Compiled on November 9, 2006
'batch_args \-x\-p\-m
=csas25 parm=(centrm)
thermal breeder
v6-238
read composition
wtptfuel        1  11.1  4
                                8016 9.6
                                92235 3
                                92238 67.4
                                90000 20
                        1 773   end
 d2o          2 1 523   end
 d2o          3 1 333   end
 zirc4        4 1 523   end
 zirc4        5 1 333   end
 d2o          6 1 523   end
 zirc4        7 1 523   end
 n            8 den=0.000129 1 523   end
 atomco2          9  0.000125  2
                                6000 1
                                8016 2
                        1 523   end
 d2o          11 1 303   end
 d2o          12 1 523   end
 zirc4        13 1 523   end
 d2o          14 1 523   end
 zirc4        15 1 523   end
 n            16 den=0.000129 1 523   end
 d2o          17 1 303   end
 wtptcandu        10  11.6  18
                                92235 0.2457388
                                92236 0.2085
                                92238 79.0778
                                94239 0.247
                                90000 20
                                92234 2.89e-05
                                94238 0.00191
                                94240 0.1533
                                94241 0.0368
                                94242 0.0178
                                93237 0.00831
                                95241 0.000687
                                95243 0.00163
                                95601 7.79e-06
                                96242 0.000247
                                96243 3.23e-06
                                96244 0.000234
```

```
                                96245 3.32e-06
                        1 523   end
end composition
read celldata
  latticecell triangpitch fuelr=0.6077 1 gapr=0.612 8 cladr=0.654 4
hpitch=1.408 12 end
    centrm data  isn=6  iim=50  iup=20     end centrm
  latticecell triangpitch fuelr=0.6077 10 gapr=0.612 16 cladr=0.654 15
hpitch=1.408 14 end
    centrm data  isn=6  iim=50  iup=20     end centrm
end celldata
read parameter
 gen=450
 npg=5000
 nsk=50
 flx=yes
 htm=yes
end parameter
read geometry
unit 1
com='u fuel rod'
 zcylinder 1 1  0.6077      600        0
 zcylinder 8 1   0.612      600        0
 zcylinder 4 1   0.654      600        0
unit 2
com='u fuel bundle'
 zcylinder 12 1  5.1689      700      100
  hole 1       0        0      100
  hole 1   1.408        0      100
  hole 1  -1.408        0      100
  hole 1   0.704  1.219364      100
  hole 1   0.704  -1.219364      100
  hole 1  -0.704  1.219364      100
  hole 1  -0.704  -1.219364      100
  hole 1 2.623455  0.702953      100
  hole 1 -2.623455  0.702953      100
  hole 1 2.623455  -0.702953      100
  hole 1 -2.623455  -0.702953      100
  hole 1 1.920502  1.920502      100
  hole 1 1.920502  -1.920502      100
  hole 1 -1.920502  1.920502      100
  hole 1 -1.920502  -1.920502      100
  hole 1 0.702953  2.623455      100
  hole 1 -0.702953  2.623455      100
  hole 1 0.702953  -2.623455      100
  hole 1 -0.702953  -2.623455      100
  hole 1 3.987487  0.703101      100
  hole 1 3.987487  -0.703101      100
  hole 1 -3.987487  0.703101      100
  hole 1 -3.987487  -0.703101      100
  hole 1       0    4.029      100
  hole 1       0   -4.029      100
  hole 1 3.506537   2.0245      100
```

```
  hole 1 3.506537  -2.0245      100
  hole 1 -3.506537   2.0245      100
  hole 1 -3.506537  -2.0245      100
  hole 1 2.602647  3.101714      100
  hole 1 2.602647  -3.101714      100
  hole 1 -2.602647  3.101714      100
  hole 1 -2.602647  -3.101714      100
  hole 1 1.38484  3.804815      100
  hole 1 1.38484  -3.804815      100
  hole 1 -1.38484  3.804815      100
  hole 1 -1.38484  -3.804815      100
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 3
com='th fuel rod'
 zcylinder 10 1  0.6077      600        0
 zcylinder 16 1   0.612      600        0
 zcylinder 15 1   0.654      600        0
unit 4
com='th fuel bundle'
 zcylinder 14 1  5.1689      700      100
  hole 3       0        0      100
  hole 3   1.408        0      100
  hole 3  -1.408        0      100
  hole 3   0.704  1.219364      100
  hole 3   0.704  -1.219364      100
  hole 3  -0.704  1.219364      100
  hole 3  -0.704  -1.219364      100
  hole 3 2.623455  0.702953      100
  hole 3 -2.623455  0.702953      100
  hole 3 2.623455  -0.702953      100
  hole 3 -2.623455  -0.702953      100
  hole 3 1.920502  1.920502      100
  hole 3 1.920502  -1.920502      100
  hole 3 -1.920502  1.920502      100
  hole 3 -1.920502  -1.920502      100
  hole 3 0.702953  2.623455      100
  hole 3 -0.702953  2.623455      100
  hole 3 0.702953  -2.623455      100
  hole 3 -0.702953  -2.623455      100
  hole 3 3.987487  0.703101      100
  hole 3 3.987487  -0.703101      100
  hole 3 -3.987487  0.703101      100
  hole 3 -3.987487  -0.703101      100
  hole 3       0    4.029      100
  hole 3       0   -4.029      100
  hole 3 3.506537   2.0245      100
  hole 3 3.506537  -2.0245      100
  hole 3 -3.506537   2.0245      100
  hole 3 -3.506537  -2.0245      100
```

```
  hole 3 2.602647  3.101714      100
  hole 3 2.602647  -3.101714      100
  hole 3 -2.602647  3.101714      100
  hole 3 -2.602647  -3.101714      100
  hole 3 1.38484  3.804815      100
  hole 3 1.38484  -3.804815      100
  hole 3 -1.38484  3.804815      100
  hole 3 -1.38484  -3.804815      100
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 5
com='outer shell h2o '
 cuboid 11 1  14.2875  -14.2875  14.2875  -14.2875      800        0
global unit 6
com='core shell'
  array 1    -324     -324        0
 zcylinder 17 1     550      850      -50
end geometry
read array
ara=1 nux=24 nuy=24 nuz=1
 com=''
 fill
    5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5
    5 5 5 5 5 4 4 4 4 4 4 2 2 4 4 4 4 4 4 5 5 5 5 5
    5 5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5 5
    5 5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5 5
    5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5
    5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5
    5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5
    5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5
    5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5
    4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4
    4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4
    4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4
    4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4
    5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5
    5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5
    5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5
    5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5
    5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5
    5 5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5 5
    5 5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5 5
    5 5 5 5 5 4 4 4 4 4 4 2 2 4 4 4 4 4 4 5 5 5 5 5
    5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5
 end fill
end array
end data
end
```

APPENDIX E

Thermal Breeder Core-1 SCALE5.1 Depletion Input Deck

```
'Input generated by GeeWiz SCALE 5.1 Compiled on November 9, 2006
'batch_args \-x\-p\-m
=t5-depl parm=(centrm)
thermal breeder
v6-238
read composition
wtptfuel        1  11.1  4
                                  8016 9.6
                                  92235 3.0
                                  92238 67.4
                                  90000 20
                        1 773   end
 d2o          2 1 523   end
 d2o          3 1 333   end
 zirc4        4 1 523   end
 zirc4        5 1 333   end
 d2o          6 1 523   end
 zirc4        7 1 523   end
 n            8 den=0.000129 1 523   end
 atomco2          9  0.000125  2
                                  6000 1
                                  8016 2
                        1 523   end
 wtptcandu       10  11.6  18
                                  92235 0.2457388
                                  92236 0.2085
                                  92238 79.0778
                                  94239 0.247
                                  90000 20
                                  92234 2.89e-05
                                  94238 0.00191
                                  94240 0.1533
                                  94241 0.0368
                                  94242 0.0178
                                  93237 0.00831
                                  95241 0.000687
                                  95243 0.00163
                                  95601 7.79e-06
                                  96242 0.000247
                                  96243 3.23e-06
                                  96244 0.000234
                                  96245 3.32e-06
                        1 523   end
 d2o          11 1 303   end
 d2o          12 1 523   end
 zirc4        13 1 523   end
 d2o          14 1 523   end
```

```
 zirc4       15 1 523   end
 n           16 den=0.000129 1 523   end
 d2o         17 1 303   end
end composition
read celldata
  latticecell triangpitch fuelr=0.6077 1 gapr=0.612 8 cladr=0.654 4
hpitch=1.408 12 end
    centrm data  isn=6  iim=50  iup=20     end centrm
  latticecell triangpitch fuelr=0.6077 10 gapr=0.612 16 cladr=0.654 15
hpitch=1.408 14 end
    centrm data  isn=6  iim=50  iup=20     end centrm
end celldata
read depletion
  1 10
end depletion
read burndata
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
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  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
  power=32 burn=30 nlib=2 end
end burndata
read opus
  symnuc=u-233 u-234 u-235 u-236 u-238 pu-238 pu-239 pu-240 pu-241
pu-242 np-237 am-241 am-243 cm-242 cm-243 cs-134 cs-137 nd-143 nd-144
nd-145 nd-146 cm-244 cm-245 cm-246 cm-247 ru-106 am-242m th-232 end
  units=grams
  time=years
```

```
  matl=0 1 10 end
end opus
read model
read parameter
 gen=450
 npg=5000
 nsk=50
 flx=yes
 htm=yes
end parameter
read geometry
unit 1
com='u fuel rod'
 zcylinder 1 1  0.6077      600        0
 zcylinder 8 1   0.612      600        0
 zcylinder 4 1   0.654      600        0
unit 2
com='u fuel bundle'
 zcylinder 12 1  5.1689      700      100
  hole 1       0        0      100
  hole 1   1.408        0      100
  hole 1  -1.408        0      100
  hole 1   0.704  1.219364      100
  hole 1   0.704  -1.219364      100
  hole 1  -0.704  1.219364      100
  hole 1  -0.704  -1.219364      100
  hole 1 2.623455  0.702953      100
  hole 1 -2.623455  0.702953      100
  hole 1 2.623455  -0.702953      100
  hole 1 -2.623455  -0.702953      100
  hole 1 1.920502  1.920502      100
  hole 1 1.920502  -1.920502      100
  hole 1 -1.920502  1.920502      100
  hole 1 -1.920502  -1.920502      100
  hole 1 0.702953  2.623455      100
  hole 1 -0.702953  2.623455      100
  hole 1 0.702953  -2.623455      100
  hole 1 -0.702953  -2.623455      100
  hole 1 3.987487  0.703101      100
  hole 1 3.987487  -0.703101      100
  hole 1 -3.987487  0.703101      100
  hole 1 -3.987487  -0.703101      100
  hole 1       0    4.029      100
  hole 1       0   -4.029      100
  hole 1 3.506537   2.0245      100
  hole 1 3.506537  -2.0245      100
  hole 1 -3.506537   2.0245      100
  hole 1 -3.506537  -2.0245      100
  hole 1 2.602647  3.101714      100
  hole 1 2.602647  -3.101714      100
  hole 1 -2.602647  3.101714      100
  hole 1 -2.602647  -3.101714      100
  hole 1 1.38484  3.804815      100
```

```
  hole 1 1.38484  -3.804815      100
  hole 1 -1.38484  3.804815      100
  hole 1 -1.38484  -3.804815      100
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 3
com='th fuel rod'
 zcylinder 10 1  0.6077      600        0
 zcylinder 16 1   0.612      600        0
 zcylinder 15 1   0.654      600        0
unit 4
com='th fuel bundle'
 zcylinder 14 1  5.1689      700      100
  hole 3       0        0      100
  hole 3   1.408        0      100
  hole 3  -1.408        0      100
  hole 3   0.704  1.219364      100
  hole 3   0.704  -1.219364      100
  hole 3  -0.704  1.219364      100
  hole 3  -0.704  -1.219364      100
  hole 3 2.623455  0.702953      100
  hole 3 -2.623455  0.702953      100
  hole 3 2.623455  -0.702953      100
  hole 3 -2.623455  -0.702953      100
  hole 3 1.920502  1.920502      100
  hole 3 1.920502  -1.920502      100
  hole 3 -1.920502  1.920502      100
  hole 3 -1.920502  -1.920502      100
  hole 3 0.702953  2.623455      100
  hole 3 -0.702953  2.623455      100
  hole 3 0.702953  -2.623455      100
  hole 3 -0.702953  -2.623455      100
  hole 3 3.987487  0.703101      100
  hole 3 3.987487  -0.703101      100
  hole 3 -3.987487  0.703101      100
  hole 3 -3.987487  -0.703101      100
  hole 3       0    4.029      100
  hole 3       0   -4.029      100
  hole 3 3.506537   2.0245      100
  hole 3 3.506537  -2.0245      100
  hole 3 -3.506537   2.0245      100
  hole 3 -3.506537  -2.0245      100
  hole 3 2.602647  3.101714      100
  hole 3 2.602647  -3.101714      100
  hole 3 -2.602647  3.101714      100
  hole 3 -2.602647  -3.101714      100
  hole 3 1.38484  3.804815      100
  hole 3 1.38484  -3.804815      100
  hole 3 -1.38484  3.804815      100
  hole 3 -1.38484  -3.804815      100
```

```
 zcylinder 6 1  5.1689      800        0
 zcylinder 13 1  5.6032      800        0
 zcylinder 9 1  6.4478      800        0
 zcylinder 5 1  6.5872      800        0
 cuboid 3 1  14.2875  -14.2875  14.2875  -14.2875      800        0
unit 5
com='outer shell h2o '
 cuboid 11 1  14.2875  -14.2875  14.2875  -14.2875      800        0
global unit 6
com='core shell'
  array 1    -324     -324        0
 zcylinder 17 1     550      850      -50
end geometry
read array
ara=1 nux=24 nuy=24 nuz=1
 com=''
 fill
    5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5
    5 5 5 5 5 4 4 4 4 4 4 2 2 4 4 4 4 4 4 5 5 5 5 5
    5 5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5 5
    5 5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5 5
    5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5 5
    5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5
    5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5
    5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5
    5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5
    4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4
    4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4
    4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4
    4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4
    5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5
    5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5
    5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5
    5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5
    5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 5 5 4 4 2 2 2 2 2 2 2 2 2 2 2 2 2 4 4 5 5 5
    5 5 5 5 4 4 4 2 2 2 2 2 2 2 2 2 2 2 2 4 4 4 5 5
    5 5 5 5 5 4 4 4 4 4 4 4 2 2 4 4 4 4 4 4 5 5 5 5
    5 5 5 5 5 4 4 4 4 4 4 4 4 4 4 4 4 4 4 5 5 5 5 5
end fill
end array
read mixt
 eps=1.0 sct=2
end mixt
end data
end model
end
```

APPENDIX F

```
CANDU-6 input deck in HELIOS-1.4 with natural uranium fuel

+THEL

Case8=CASE('hy3418-961a.dat'/'can8.hrf'/'DUPIC')

!materials database!
!moderator!
'D2O-68'=MAT(1.085743/1002,19.981999520432;
1001,0.010070999758296;8016,80.0079270798097004;
5000,0.0000024)

!coolant!
'D2O-288'=MAT(0.807642/1002,19.95;1001,0.027975;8016,80.022025)
!'D2O-288a'=MAT(0.7672599/1002,19.95;1001,0.027975;8016,80.022025)
'D2O-288b'=MAT(0.7268778/1002,19.95;1001,0.027975;8016,80.022025)
'D2O-288c'=MAT(0.6864957/1002,19.95;1001,0.027975;8016,80.022025)
'D2O-288d'=MAT(0.403821/1002,19.95;1001,0.027975;8016,80.022025)
'D2O-288e'=MAT(0.005/1002,19.95;1001,0.027975;8016,80.022025)!

!fuel!
'UO2-687'=MAT 9.19/92235,0.7204;92238,99.2742;92234,0.0054;8001,0)
'Zr-1'=MAT(6.55/40000,100)!clad!
'Zr-2'=MAT(7.68167/40000,100)!caltub!
'Zr-3'=MAT(8.1446/40000,100)!pretub!
nitr=MAT(0.808607/7014,100)!gap1!
cadi=MAT(0.0019/6000,27.27;8016,72.73)!gap2!
white=ALB(1/1/1) !diagonal albedo!

!------------------CANDU-6----------37-EL-------------------!

$p=PAR("28.575") !pitch length!
$co=PAR("1.308/2") !clad outer radius!
$inrr=PAR("2.97688/2") !inner ring radius!
$mrr=PAR("5.75056/2")  !middle ring radius!
$orr=PAR("8.66140/2")  !outer ring radius!
$ptir=PAR("10.3378/2") !pressure tube inner radius!
$ptor=PAR("11.2064/2")  !pressure tube outer radius!
$ctir=PAR("12.8956/2")  !calandria tube inner radius!
$ctor=PAR("13.175/2")  !calandria tube outer radius!
$cos30=PAR("(3**0.5)/2")  !cos30,sin60!
$sin60=PAR("(3**0.5)/2")  !cos30,sin60!
$cos60=PAR("0.5")  !cos60,sin30!
$sin30=PAR("0.5")  !cos60,sin30!
$cos20=PAR("0.9396926207859083")  !cos20!
$cos40=PAR("0.7660444431189780")  !cos40!
$cos80=PAR("0.1736481776669303")  !cos80!
$sin20=PAR("0.3420201433256687") !sin20!
$sin40=PAR("0.6427876096865393")  !sin40!
$sin80=PAR("0.9848077530122080")  !sin80!
```

```
$cos10=PAR("0.9848077530122080")
$cos15=PAR("0.9659258262890682")
$cos45=PAR("1/(2**0.5)")
$cos50=PAR("0.6427876096865393")
$cos5=PAR("0.9961946980917455")
$sin10=PAR("0.1736481776669303")
$sin15=PAR("0.2588190451025207")
$sin45=PAR("1/(2**0.5)")
$sin50=PAR("0.7660444431189780")
$sin5=PAR("0.0871557427476581")
$cos25=PAR("0.9063077870366499")
$cos35=PAR("0.8191520442889917")
$cos55=PAR("0.5735764363510460")
$sin25=PAR("0.4226182617406994")
$sin35=PAR("0.5735764363510460")
$sin55=PAR("0.8191520442889917")
$gap2=PAR("($mrr-$inrr)/2")
$gap3=PAR("($orr-$mrr)/2")
$gap4=PAR("$ptir-$co-$orr")
$ccl=PAR("$inrr/(2*$cos30)")  !length for central cell nodes!
$mcl=PAR("($mrr-$inrr)/(2*$cos30)") !length for 1st middle ring cell
nodes!
$scl=PAR("($orr-$mrr)/(2*$cos30)") !length for 2nd middle ring cell
nodes!
$c=PAR("0.235/2")
$coo=PAR("$co+$c")
$tcl=PAR("$coo/$cos30") !length for outer ring cell nodes!
$r3=PAR("($co**2+$inrr**2)**0.5")
$cosir1=PAR("0.9939992452945486")
$sinir1=PAR("0.1093869295385313")
$cosir2=PAR("0.5917314824696208")
$sinir2=PAR("0.8061351329983731")
$r4=PAR("($co**2+$mrr**2)**0.5")
$cosmr1=PAR("0.9992724372993703")
$sinmr1=PAR("0.0381391669255617")
$cosmr2=PAR("0.8844648994656282")
$sinmr2=PAR("0.4666067312129734")
$cosmr3=PAR("0.8463257325400665")
$sinmr3=PAR("0.5326657060863969")
$cosmr4=PAR("0.5326657060863969")
$sinmr4=PAR("0.8463257325400665")
$r5=PAR("($co**2+$orr**2)**0.5")
$cosor1=PAR("0.9996961891034971")
$sinor1=PAR("0.0246481133546726")
$cosor2=PAR("0.9478372831906228")
$sinor2=PAR("0.3187545836436224")
$cosor3=PAR("0.9309769806660778")
$sinor3=PAR("0.3650778841149834")
$cosor4=PAR("0.7816552123364859")
$sinor4=PAR("0.6237107735378658")
$cosor5=PAR("0.7499682086034201")
$sinor5=PAR("0.6614738740752932")
$cosor6=PAR("0.5211939868722536")
```

```
$sinor6=PAR("0.8534382391527843")
$cos2=PAR("0.9990482215818577")
$sin2=PAR("0.0436193873653359")
$cos7=PAR("0.9914448613738104")
$sin7=PAR("0.1305261922205159")
$cos12=PAR("0.9762960071199333")
$sin12=PAR("0.2164396139381028")
$cos17=PAR("0.9537169507482269")
$sin17=PAR("0.3007057995042731")
$cos22=PAR("0.9238795325112867")
$sin22=PAR("0.3826834323650897")
$cos27=PAR("0.8870108331782217")
$sin27=PAR("0.4617486132350339")
$cos32=PAR("0.8433914458128857")
$sin32=PAR("0.5372996083468238")
$cos37=PAR("0.7933533402912351")
$sin37=PAR("0.6087614290087206")
$cos42=PAR("0.7372773368101240")
$sin42=PAR("0.6755902076156602")
$cos47=PAR("0.6755902076156602")
$sin47=PAR("0.7372773368101240")
$cos52=PAR("0.6087614290087206")
$sin52=PAR("0.7933533402912351")
$cos57=PAR("0.5372996083468238")
$sin57=PAR("0.8433914458128857")

! structure of the central pin!
pin=CCS(0.3,0.4,0.5,0.6077,0.612,$co//fuel,fuel,fuel,fuel,gap1,clad)
$peri1=PAR(!structure one peripheri!
("$ccl*$cos60","$ccl*$cos30"),($ccl,0),
("$ccl*$cos60","-$ccl*$cos30"),("-$ccl*$cos60","-$ccl*$cos30"),
("-$ccl",0),("-$ccl*$cos60","$ccl*$cos30"))
$coolreg1=PAR(!cool reg surrounding the central pin cell!
5,6,12,11,cool;6,1,7,12,cool;1,2,8,7,cool;2,3,9,8,cool;
3,4,10,9,cool  !4,5,11,10 is default coolant!)

$pincell1=PAR($peri1("$co*$cos60","$co*$cos30"),($co,0),
("$co*$cos60","-$co*$cos30"),("-$co*$cos60","-$co*$cos30"),
("-$co",0),("-$co*$cos60","$co*$cos30")/
6,cool/pin(0,0)/$coolreg1)
$peri2=PAR(!structure two peripheri!
   ("-$ccl",0),
   ("-($orr+$co)",0),
   ("-($orr+$co+($gap4/2))*$cos10","($orr+$co+($gap4/2))*$sin10"),
   ("-($orr+$co)*$cos20","($orr+$co)*$sin20"),
   ("-($orr+$co+($gap4/2))*$cos30","($orr+$co+($gap4/2))*$sin30"),
   ("-($orr+$co)*$cos40","($orr+$co)*$sin40"),
   ("-($orr+$co+($gap4/2))*$cos50","($orr+$co+($gap4/2))*$sin50"),
   ("-($orr+$co)*$cos60","($orr+$co)*$sin60"),
   ("-$ccl*$cos60","$ccl*$cos30"),
   ("-($inrr+$co*6/7)",0),
   ("-($inrr+$co+($gap2/4))*$cos30","($inrr+$co+($gap2/4))*$sin30"),
   ("-($inrr+$co*6/7)*$cos60","($inrr+$co*6/7)*$sin60"),
```

```
   ("-($mrr+$co*7/6)",0),
 ("-($orr-$co)*$cos20","($orr-$co)*$sin20"),
 ("-($orr-$co-($gap3/2))*$cos30","($orr-$co-($gap3/2))*$sin30"),
 ("-($orr-$co)*$cos40","($orr-$co)*$sin40"),
 ("-($mrr+$co*7/6)*$cos60","($mrr+$co*7/6)*$sin60"))

$coolreg2=PAR(1,10,18,21,22,cool;11,19,18,10,cool;
12,20,19,11,cool;22,21,20,12,9,cool;
13,25,28,10,cool;14,26,25,13,cool;15,27,26,14,cool;
10,28,27,15,11,cool;
15,34,33,12,11,cool;16,31,34,15,cool;17,32,31,16,cool;
12,33,32,17,cool;
13,2,3,4,14,cool;14,4,5,6,16,15,cool!;16,6,7,8,17,cool!)

$pincell2=PAR($peri2("-$r3*$cosir1","$r3*$sinir1"),
        ("-($inrr+$co)*$cos30","($inrr+$co)*$sin30"),
        ("-$r3*$cosir2","$r3*$sinir2"),
        ("-($inrr-$co)*$cos30","($inrr-$co)*$sin30"),
        ("-($inrr/2)*$cos30","($inrr/2)*$sin30"),
        ("-($inrr+$co)*5.7/9",0),
("-($inrr+$co)*$cos60*5.7/9","($inrr+$co)*$sin60*5.7/9"),
("-$r4*$cosmr1","$r4*$sinmr1"),
("-($mrr+$co)*$cos15","($mrr+$co)*$sin15"),
("-$r4*$cosmr2","$r4*$sinmr2"),
 ("-($mrr-$co)*$cos15","($mrr-$co)*$sin15"),
 ("-($mrr)",0)
 ("-$mrr*$cos30","$mrr*$sin30"),!30!
("-($mrr+$co)*$cos45","($mrr+$co)*$sin45"),!31!
      ("-$r4*$cosmr4","$r4*$sinmr4"),
("-($mrr-$co)*$cos45","($mrr-$co)*$sin45"),!33!
      ("-$r4*$cosmr3","$r4*$sinmr3"),
     ("-$mrr*$cos60","$mrr*$sin60"),!35!
     ("-($orr)",0),!36!
    ("-$orr*$cos20","$orr*$sin20")
   ! ("-$r5*$cosor1","$r5*$sinor1"),!
  ! ("-($orr+$co)*$cos10","($orr+$co)*$sin10"),!
   ! ("-$r5*$cosor2","$r5*$sinor2"),!
  !  ("-($orr-$co)*$cos10","($orr-$co)*$sin10"),!!40!
 !  ("-$r5*$cosor3","$r5*$sinor3"),
   ("-($orr+$co)*$cos30","($orr+$co)*$sin30"),
   ("-$r5*$cosor4","$r5*$sinor4"),
   ("-($orr-$co)*$cos30","($orr-$co)*$sin30"),
    ("-$orr*$cos40","$orr*$sin40"),
    ("-$r5*$cosor5","$r5*$sinor5"),
   ("-($orr+$co)*$cos50","($orr+$co)*$sin50"),
   ("-$r5*$cosor6","$r5*$sinor6"),
   ("-($orr-$co)*$cos50","($orr-$co)*$sin50"),
    ("-$orr*$cos60","$orr*$sin60")!/
     9,cool/pin("-$inrr*$cos30","$inrr*$sin30"),

   pin("-$mrr*$cos15","$mrr*$sin15"),
   pin("-$mrr*$cos45","$mrr*$sin45"),
   pin("-$orr*$cos10","$orr*$sin10"),
```

```
   pin("-$orr*$cos30","$orr*$sin30"),
   pin("-$orr*$cos50","$orr*$sin50")/$coolreg2)
$peri3=PAR(("-($orr+$co)",0),
          ("-$p/2",0),
          ("-($p/(2**0.5))*$cos45","($p/(2**0.5))*$sin45"),
         ("(-$p/2)*$cos60/$sin60","($p/2)"),
("-($orr+$co!+($gap4/2)!)*$cos60","($orr+$co!+($gap4/2)!)*$sin60"),
("-($orr+$co+($gap4/2))*$cos50","($orr+$co+($gap4/2))*$sin50"),
("-($orr+$co!+($gap4/2)!)*$cos40","($orr+$co!+($gap4/2)!)*$sin40"),
("-($orr+$co+($gap4/2))*$cos30","($orr+$co+($gap4/2))*$sin30"),
("-($orr+$co!+($gap4/2)!)*$cos20","($orr+$co!+($gap4/2)!)*$sin20"),
("-($orr+$co+($gap4/2))*$cos10","($orr+$co+($gap4/2))*$sin10"))

$coolreg3=PAR(12,4,180,179,178,177,176,175,2,mod;
2,175,176,177,178,179,180,174,173,172,171,170,169,168,mod;
168,169,170,171,172,173,174,167,166,165,164,163,162,161,mod;
161,162,163,164,165,166,167,160,159,158,157,156,155,154,mod;
154,155,156,157,158,159,160,153,152,151,150,149,148,147,mod;
147,148,149,150,151,152,153,146,145,144,143,142,141,140,mod;
140,141,142,143,144,145,146,139,138,137,136,135,134,133,mod;
133,134,135,136,137,138,139,132,131,130,129,128,127,126,mod;
126,127,128,129,130,131,132,125,124,123,122,121,120,119,mod;
119,120,121,122,123,124,125,118,117,116,115,114,113,112,mod;
112,113,114,115,116,117,118,13,111,110,109,108,107,106,105,
104,103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,mod;
4,12,3,mod;
66,67,68,69,70,71,72,73,74,75,76,77,78,79,80,81,82,83,84,85,86,
87,88,15,17,65,64,63,62,61,60,59,58,57,56,55,54,53,52,51,50,49,
48,47,46,45,44,43,16,14,gap2;
11,89,90,91,92,93,94,95,96,97,98,99,100,101,102,103,104,105,106,
107,108,109,110,111,13,15,88,87,86,85,84,83,82,81,80,79,78,77,76,
75,74,73,72,71,70,69,68,67,66,14,caltub;
43,44,45,46,47,48,49,
50,51,52,53,54,55,56,57,58,59,60,61,62,63,64,65,17,18,42,41,40,
39,38,37,36,35,34,33,32,31,30,29,28,27,26,25,24,23,22,21,20,
19,16,pretub
!20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,5,6,7,8,9,10,1,19,cool!)

$pincell3=PAR($peri3("-$ctor",0),
          ("(-$p/2)","($p/4)"),
         ("-$ctor*$cos60","$ctor*$sin60"),
         ("-$ctir",0),
         ("-$ctir*$cos60","$ctir*$sin60"),
        ("-$ptor",0),
        ("-$ptor*$cos60","$ptor*$sin60"),
        ("-$ptir*$cos60","$ptir*$sin60"),
        ("-$ptir",0),
       ("-$ptir*$cos2","$ptir*$sin2"),
       ("-$ptir*$cos5","$ptir*$sin5"),
       ("-$ptir*$cos7","$ptir*$sin7"),
       ("-$ptir*$cos10","$ptir*$sin10"),
       ("-$ptir*$cos12","$ptir*$sin12"),
```

```
("-$ptir*$cos15","$ptir*$sin15"),
("-$ptir*$cos17","$ptir*$sin17"),
("-$ptir*$cos20","$ptir*$sin20"),
("-$ptir*$cos22","$ptir*$sin22"),
("-$ptir*$cos25","$ptir*$sin25"),
("-$ptir*$cos27","$ptir*$sin27"),
("-$ptir*$cos30","$ptir*$sin30"),
("-$ptir*$cos32","$ptir*$sin32"),
("-$ptir*$cos35","$ptir*$sin35"),
("-$ptir*$cos37","$ptir*$sin37"),
("-$ptir*$cos40","$ptir*$sin40"),
("-$ptir*$cos42","$ptir*$sin42"),
("-$ptir*$cos45","$ptir*$sin45"),
("-$ptir*$cos47","$ptir*$sin47"),
("-$ptir*$cos50","$ptir*$sin50"),
("-$ptir*$cos52","$ptir*$sin52"),
("-$ptir*$cos55","$ptir*$sin55"),
("-$ptir*$cos57","$ptir*$sin57"),
("-$ptor*$cos2","$ptor*$sin2"),
("-$ptor*$cos5","$ptor*$sin5"),
("-$ptor*$cos7","$ptor*$sin7"),
("-$ptor*$cos10","$ptor*$sin10"),
("-$ptor*$cos12","$ptor*$sin12"),
("-$ptor*$cos15","$ptor*$sin15"),
("-$ptor*$cos17","$ptor*$sin17"),
("-$ptor*$cos20","$ptor*$sin20"),
("-$ptor*$cos22","$ptor*$sin22"),
("-$ptor*$cos25","$ptor*$sin25"),
("-$ptor*$cos27","$ptor*$sin27"),
("-$ptor*$cos30","$ptor*$sin30"),
("-$ptor*$cos32","$ptor*$sin32"),
("-$ptor*$cos35","$ptor*$sin35"),
("-$ptor*$cos37","$ptor*$sin37"),
("-$ptor*$cos40","$ptor*$sin40"),
("-$ptor*$cos42","$ptor*$sin42"),
("-$ptor*$cos45","$ptor*$sin45"),
("-$ptor*$cos47","$ptor*$sin47"),
("-$ptor*$cos50","$ptor*$sin50"),
("-$ptor*$cos52","$ptor*$sin52"),
("-$ptor*$cos55","$ptor*$sin55"),
("-$ptor*$cos57","$ptor*$sin57"),
("-$ctir*$cos2","$ctir*$sin2"),
("-$ctir*$cos5","$ctir*$sin5"),
("-$ctir*$cos7","$ctir*$sin7"),
("-$ctir*$cos10","$ctir*$sin10"),
("-$ctir*$cos12","$ctir*$sin12"),
("-$ctir*$cos15","$ctir*$sin15"),
("-$ctir*$cos17","$ctir*$sin17"),
("-$ctir*$cos20","$ctir*$sin20"),
("-$ctir*$cos22","$ctir*$sin22"),
("-$ctir*$cos25","$ctir*$sin25"),
("-$ctir*$cos27","$ctir*$sin27"),
("-$ctir*$cos30","$ctir*$sin30"),
```

```
        ("-$ctir*$cos32","$ctir*$sin32"),
        ("-$ctir*$cos35","$ctir*$sin35"),
        ("-$ctir*$cos37","$ctir*$sin37"),
        ("-$ctir*$cos40","$ctir*$sin40"),
        ("-$ctir*$cos42","$ctir*$sin42"),
        ("-$ctir*$cos45","$ctir*$sin45"),
        ("-$ctir*$cos47","$ctir*$sin47"),
        ("-$ctir*$cos50","$ctir*$sin50"),
        ("-$ctir*$cos52","$ctir*$sin52"),
        ("-$ctir*$cos55","$ctir*$sin55"),
        ("-$ctir*$cos57","$ctir*$sin57"),
        ("-$ctor*$cos2","$ctor*$sin2"),
        ("-$ctor*$cos5","$ctor*$sin5"),
        ("-$ctor*$cos7","$ctor*$sin7"),
        ("-$ctor*$cos10","$ctor*$sin10"),
        ("-$ctor*$cos12","$ctor*$sin12"),
        ("-$ctor*$cos15","$ctor*$sin15"),
        ("-$ctor*$cos17","$ctor*$sin17"),
        ("-$ctor*$cos20","$ctor*$sin20"),
        ("-$ctor*$cos22","$ctor*$sin22"),
        ("-$ctor*$cos25","$ctor*$sin25"),
        ("-$ctor*$cos27","$ctor*$sin27"),
        ("-$ctor*$cos30","$ctor*$sin30"),
        ("-$ctor*$cos32","$ctor*$sin32"),
        ("-$ctor*$cos35","$ctor*$sin35"),
        ("-$ctor*$cos37","$ctor*$sin37"),
        ("-$ctor*$cos40","$ctor*$sin40"),
        ("-$ctor*$cos42","$ctor*$sin42"),
        ("-$ctor*$cos45","$ctor*$sin45"),
        ("-$ctor*$cos47","$ctor*$sin47"),
        ("-$ctor*$cos50","$ctor*$sin50"),
        ("-$ctor*$cos52","$ctor*$sin52"),
        ("-$ctor*$cos55","$ctor*$sin55"),
        ("-$ctor*$cos57","$ctor*$sin57"),
("-((($p/2)-$ctor)/10)+$ctor)",0), !112!
("-((($p/2)-$ctor)/10)+$ctor)*$cos10","((($p/2)-
$ctor)/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos20","((($p/2)-
$ctor)/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos30","((($p/2)-
$ctor)/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos40","((($p/2)-
$ctor)/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos50","((($p/2)-
$ctor)/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos60","((($p/2)-
$ctor)/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*2/10)+$ctor)",0), !119!
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos10","((($p/2)-
$ctor)*2/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos20","((($p/2)-
$ctor)*2/10)+$ctor)*$sin20"),
```

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("-((($p/2)-$ctor)*2/10)+$ctor)*$cos30","((($p/2)-
$ctor)*2/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos40","((($p/2)-
$ctor)*2/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos50","((($p/2)-
$ctor)*2/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos60","((($p/2)-
$ctor)*2/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*3/10)+$ctor)",0), !126!
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos10","((($p/2)-
$ctor)*3/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos20","((($p/2)-
$ctor)*3/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos30","((($p/2)-
$ctor)*3/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos40","((($p/2)-
$ctor)*3/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos50","((($p/2)-
$ctor)*3/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos60","((($p/2)-
$ctor)*3/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*4/10)+$ctor)",0), !133!
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos10","((($p/2)-
$ctor)*4/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos20","((($p/2)-
$ctor)*4/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos30","((($p/2)-
$ctor)*4/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos40","((($p/2)-
$ctor)*4/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos50","((($p/2)-
$ctor)*4/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos60","((($p/2)-
$ctor)*4/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*5/10)+$ctor)",0), !140!
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos10","((($p/2)-
$ctor)*5/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos20","((($p/2)-
$ctor)*5/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos30","((($p/2)-
$ctor)*5/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos40","((($p/2)-
$ctor)*5/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos50","((($p/2)-
$ctor)*5/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos60","((($p/2)-
$ctor)*5/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*6/10)+$ctor)",0), !147!
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos10","((($p/2)-
$ctor)*6/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos20","((($p/2)-
$ctor)*6/10)+$ctor)*$sin20"),
```

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("-((($p/2)-$ctor)*6/10)+$ctor)*$cos30","((($p/2)-
$ctor)*6/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos40","((($p/2)-
$ctor)*6/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos50","((($p/2)-
$ctor)*6/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos60","((($p/2)-
$ctor)*6/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*7/10)+$ctor)",0), !154!
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos10","((($p/2)-
$ctor)*7/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos20","((($p/2)-
$ctor)*7/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos30","((($p/2)-
$ctor)*7/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos40","((($p/2)-
$ctor)*7/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos50","((($p/2)-
$ctor)*7/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos60","((($p/2)-
$ctor)*7/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*8/10)+$ctor)",0), !161!
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos10","((($p/2)-
$ctor)*8/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos20","((($p/2)-
$ctor)*8/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos30","((($p/2)-
$ctor)*8/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos40","((($p/2)-
$ctor)*8/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos50","((($p/2)-
$ctor)*8/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos60","((($p/2)-
$ctor)*8/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*9/10)+$ctor)",0), !168!
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos10","((($p/2)-
$ctor)*9/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos20","((($p/2)-
$ctor)*9/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos30","((($p/2)-
$ctor)*9/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos40","((($p/2)-
$ctor)*9/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos50","((($p/2)-
$ctor)*9/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos60","((($p/2)-
$ctor)*9/10)+$ctor)*$sin60"),
("-($p/2)*$cos10","($p/2)*$sin10"),!175!
("-($p/2)*$cos20","($p/2)*$sin20"),
("-($p/2)*$cos30","($p/2)*$sin30"),
("-($p/2)*$cos40","($p/2)*$sin40"),
("-($p/2)*$cos50","($p/2)*$sin50"),
("-($p/2)*$cos60","($p/2)*$sin60")
```

```
         /10, cool//$coolreg3)
$peri4=PAR(("-($orr+$co)*$cos60","-($orr+$co)*$sin60"),!5!
("(-$p/2)*$cos60/$sin60","-($p/2)"),!4!
("-($p/(2**0.5))*$cos45","-($p/(2**0.5))*$sin45"),!3!
("-$p/2",0),!2!
("-($orr+$co)",0),!1!
("-($orr+$co+($gap4/2))*$cos10","-($orr+$co+($gap4/2))*$sin10"),!10!
("-($orr+$co)*$cos20","-($orr+$co)*$sin20"),!9!
("-($orr+$co+($gap4/2))*$cos30","-($orr+$co+($gap4/2))*$sin30"),!8!
("-($orr+$co)*$cos40","-($orr+$co)*$sin40"),!7!
("-($orr+$co+($gap4/2))*$cos50","-($orr+$co+($gap4/2))*$sin50"))!6!

$coolreg4=PAR(3,12,2,mod;
4,175,176,177,178,179,180,2,12,mod;
168,169,170,171,172,173,174,180,179,178,177,176,175,4,mod;
161,162,163,164,165,166,167,174,173,172,171,170,169,168,mod;
154,155,156,157,158,159,160,167,166,165,164,163,162,161,mod;
147,148,149,150,151,152,153,160,159,158,157,156,155,154,mod;
140,141,142,143,144,145,146,153,152,151,150,149,148,147,mod;
133,134,135,136,137,138,139,146,145,144,143,142,141,140,mod;
126,127,128,129,130,131,132,139,138,137,136,135,134,133,mod;
119,120,121,122,123,124,125,132,131,130,129,128,127,126,mod;
112,113,114,115,116,117,118,125,124,123,122,121,120,119,mod;
11,89,90,91,92,93,94,95,96,97,98,99,100,101,102,103,104,105,106,
107,108,109,110,111,13,118,117,116,115,114,113,112,mod;
!12,2,180,179,178,177,176,175,4,mod;
4,175,176,177,178,179,180,174,173,172,171,170,169,168,mod;
168,169,170,171,172,173,174,167,166,165,164,163,162,161,mod;
161,162,163,164,165,166,167,160,159,158,157,156,155,154,mod;
154,155,156,157,158,159,160,153,152,151,150,149,148,147,mod;
147,148,149,150,151,152,153,146,145,144,143,142,141,140,mod;
140,141,142,143,144,145,146,139,138,137,136,135,134,133,mod;
133,134,135,136,137,138,139,132,131,130,129,128,127,126,mod;
126,127,128,129,130,131,132,125,124,123,122,121,120,119,mod;
119,120,121,122,123,124,125,118,117,116,115,114,113,112,mod;
112,113,114,115,116,117,118,13,111,110,109,108,107,106,105,
104,103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,mod;
3,12,2,mod;!
16,43,44,45,46,47,48,49,
50,51,52,53,54,55,56,57,58,59,60,61,62,63,64,65,17,15,88,
87,86,85,84,83,82,81,80,79,78,77,76,
75,74,73,72,71,70,69,68,67,66,14,gap2;
14,66,67,68,69,70,71,72,73,74,75,76,77,78,79,80,81,82,83,84,85,86,
87,88,15,13,111,110,109,108,107,106,105,104,
103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,caltub;
19,20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,17,65,64,63,62,61,60,59,58,57,56,55,
54,53,52,51,50,49,48,47,46,45,44,43,16,pretub)
!;20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,5,6,7,8,9,10,1,19,cool!
$pincell4=PAR($peri4("-$ctor",0),
         ("(-$p/2)","-($p/4)"),
         ("-$ctor*$cos60","-$ctor*$sin60"),
```

```
  ("-$ctir",0),
  ("-$ctir*$cos60","-$ctir*$sin60"),
 ("-$ptor",0),
 ("-$ptor*$cos60","-$ptor*$sin60"),
 ("-$ptir*$cos60","-$ptir*$sin60"),
 ("-$ptir",0), !19!
("-$ptir*$cos2","-$ptir*$sin2"),
("-$ptir*$cos5","-$ptir*$sin5"),
("-$ptir*$cos7","-$ptir*$sin7"),
("-$ptir*$cos10","-$ptir*$sin10"),
("-$ptir*$cos12","-$ptir*$sin12"),
("-$ptir*$cos15","-$ptir*$sin15"),
("-$ptir*$cos17","-$ptir*$sin17"),
("-$ptir*$cos20","-$ptir*$sin20"),
("-$ptir*$cos22","-$ptir*$sin22"),
("-$ptir*$cos25","-$ptir*$sin25"),
("-$ptir*$cos27","-$ptir*$sin27"),
("-$ptir*$cos30","-$ptir*$sin30"),
("-$ptir*$cos32","-$ptir*$sin32"),
("-$ptir*$cos35","-$ptir*$sin35"),
("-$ptir*$cos37","-$ptir*$sin37"),
("-$ptir*$cos40","-$ptir*$sin40"),
("-$ptir*$cos42","-$ptir*$sin42"),
("-$ptir*$cos45","-$ptir*$sin45"),
("-$ptir*$cos47","-$ptir*$sin47"),
("-$ptir*$cos50","-$ptir*$sin50"),
("-$ptir*$cos52","-$ptir*$sin52"),
("-$ptir*$cos55","-$ptir*$sin55"),
("-$ptir*$cos57","-$ptir*$sin57"),
("-$ptor*$cos2","-$ptor*$sin2"),
("-$ptor*$cos5","-$ptor*$sin5"),
("-$ptor*$cos7","-$ptor*$sin7"),
("-$ptor*$cos10","-$ptor*$sin10"),
("-$ptor*$cos12","-$ptor*$sin12"),
("-$ptor*$cos15","-$ptor*$sin15"),
("-$ptor*$cos17","-$ptor*$sin17"),
("-$ptor*$cos20","-$ptor*$sin20"),
("-$ptor*$cos22","-$ptor*$sin22"),
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("-$ptor*$cos27","-$ptor*$sin27"),
("-$ptor*$cos30","-$ptor*$sin30"),
("-$ptor*$cos32","-$ptor*$sin32"),
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("-$ptor*$cos55","-$ptor*$sin55"),
("-$ptor*$cos57","-$ptor*$sin57"),
("-$ctir*$cos2","-$ctir*$sin2"),
```

```
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        ("-$ctir*$cos7","-$ctir*$sin7"),
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        ("-$ctir*$cos12","-$ctir*$sin12"),
        ("-$ctir*$cos15","-$ctir*$sin15"),
        ("-$ctir*$cos17","-$ctir*$sin17"),
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        ("-$ctir*$cos42","-$ctir*$sin42"),
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        ("-$ctir*$cos52","-$ctir*$sin52"),
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        ("-$ctor*$cos12","-$ctor*$sin12"),
        ("-$ctor*$cos15","-$ctor*$sin15"),
        ("-$ctor*$cos17","-$ctor*$sin17"),
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        ("-$ctor*$cos25","-$ctor*$sin25"),
        ("-$ctor*$cos27","-$ctor*$sin27"),
        ("-$ctor*$cos30","-$ctor*$sin30"),
        ("-$ctor*$cos32","-$ctor*$sin32"),
        ("-$ctor*$cos35","-$ctor*$sin35"),
        ("-$ctor*$cos37","-$ctor*$sin37"),
        ("-$ctor*$cos40","-$ctor*$sin40"),
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        ("-$ctor*$cos50","-$ctor*$sin50"),
        ("-$ctor*$cos52","-$ctor*$sin52"),
        ("-$ctor*$cos55","-$ctor*$sin55"),
        ("-$ctor*$cos57","-$ctor*$sin57"),
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("-((($p/2)-$ctor)/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos40",
```

```
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("-((($p/2)-$ctor)/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*2/10)+$ctor)",0), !119!
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos10",
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("-((($p/2)-$ctor)*2/10)+$ctor)*$cos20",
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("-((($p/2)-$ctor)*2/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin50"),
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"-((($p/2)-$ctor)*2/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*3/10)+$ctor)",0), !126!
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*4/10)+$ctor)",0), !133!
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*5/10)+$ctor)",0), !140!
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin40"),
```

```
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*6/10)+$ctor)",0), !147!
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos40",
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("-((($p/2)-$ctor)*6/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*7/10)+$ctor)",0), !154!
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin20"),
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"-((($p/2)-$ctor)*7/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*8/10)+$ctor)",0), !161!
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*9/10)+$ctor)",0), !168!
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*9/10)+$ctor)*$cos50",
```

```
"-((((($p/2)-$ctor)*9/10)+$ctor)*$sin50"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos60",
"-((((($p/2)-$ctor)*9/10)+$ctor)*$sin60"),
("-($p/2)*$cos10","-($p/2)*$sin10"),!175!
("-($p/2)*$cos20","-($p/2)*$sin20"),
("-($p/2)*$cos30","-($p/2)*$sin30"),
("-($p/2)*$cos40","-($p/2)*$sin40"),
("-($p/2)*$cos50","-($p/2)*$sin50"),
("-($p/2)*$cos60","-($p/2)*$sin60")
         /10,cool//$coolreg4)

$peri5=PAR(("($orr+$co)*$cos60","($orr+$co)*$sin60"),!5!
("($p/2)*$cos60/$sin60","($p/2)"),!4!
("($p/(2**0.5))*$cos45","($p/(2**0.5))*$sin45"),!3!
("$p/2",0),!2!
("($orr+$co)",0),!1!
("($orr+$co+($gap4/2))*$cos10","($orr+$co+($gap4/2))*$sin10"),!10!
("($orr+$co)*$cos20","($orr+$co)*$sin20"),!9!
("($orr+$co+($gap4/2))*$cos30","($orr+$co+($gap4/2))*$sin30"),!8!
("($orr+$co)*$cos40","($orr+$co)*$sin40"),!7!
("($orr+$co+($gap4/2))*$cos50","($orr+$co+($gap4/2))*$sin50"))!6!
!("($orr+$co)",0),
            ("$p/2",0),
          ("($p/(2**0.5))*$cos45","($p/(2**0.5))*$sin45"),
          ("($p/2)*$cos60/$sin60","($p/2)"),
("($orr+$co)*$cos60","($orr+$co)*$sin60"),
("($orr+$co+($gap4/2))*$cos50","($orr+$co+($gap4/2))*$sin50"),
("($orr+$co)*$cos40","($orr+$co)*$sin40"),
("($orr+$co+($gap4/2))*$cos30","($orr+$co+($gap4/2))*$sin30"),
("($orr+$co)*$cos20","($orr+$co)*$sin20"),
("($orr+$co+($gap4/2))*$cos10","($orr+$co+($gap4/2))*$sin10"))!

$coolreg5=PAR(3,12,2,mod;
2,12,4,175,176,177,178,179,180,mod;
180,179,178,177,176,175,4,168,169,170,171,172,173,174,mod;
174,173,172,171,170,169,168,161,162,163,164,165,166,167,mod;
167,166,165,164,163,162,161,154,155,156,157,158,159,160,mod;
160,159,158,157,156,155,154,147,148,149,150,151,152,153,mod;
153,152,151,150,149,148,147,140,141,142,143,144,145,146,mod;
146,145,144,143,142,141,140,133,134,135,136,137,138,139,mod;
139,138,137,136,135,134,133,126,127,128,129,130,131,132,mod;
132,131,130,129,128,127,126,119,120,121,122,123,124,125,mod;
125,124,123,122,121,120,119,112,113,114,115,116,117,118,mod;
118,117,116,115,114,113,112,11,89,90,91,92,93,94,95,96,97,
98,99,100,101,102,103,104,105,106,
107,108,109,110,111,13,mod;
16,43,44,45,46,47,48,49,
50,51,52,53,54,55,56,57,58,59,60,61,62,63,64,65,17,15,88,
87,86,85,84,83,82,81,80,79,78,77,76,
75,74,73,72,71,70,69,68,67,66,14,gap2;
14,66,67,68,69,70,71,72,73,74,75,76,77,78,79,80,81,82,83,84,85,86,
87,88,15,13,111,110,109,108,107,106,105,104,
103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,caltub;
```

```
19,20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,17,65,64,63,62,61,60,59,58,57,56,55,
54,53,52,51,50,49,48,47,46,45,44,43,16,pretub
!;20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,5,6,7,8,9,10,1,19,cool!)

$pincell5=PAR($peri5("$ctor",0),
          ("($p/2)","($p/4)"),
         ("$ctor*$cos60","$ctor*$sin60"),
         ("$ctir",0),
         ("$ctir*$cos60","$ctir*$sin60"),
        ("$ptor",0),
        ("$ptor*$cos60","$ptor*$sin60"),
        ("$ptir*$cos60","$ptir*$sin60"),
        ("$ptir",0),
       ("$ptir*$cos2","$ptir*$sin2"),
       ("$ptir*$cos5","$ptir*$sin5"),
       ("$ptir*$cos7","$ptir*$sin7"),
       ("$ptir*$cos10","$ptir*$sin10"),
       ("$ptir*$cos12","$ptir*$sin12"),
       ("$ptir*$cos15","$ptir*$sin15"),
       ("$ptir*$cos17","$ptir*$sin17"),
       ("$ptir*$cos20","$ptir*$sin20"),
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       ("$ptir*$cos27","$ptir*$sin27"),
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       ("$ptir*$cos37","$ptir*$sin37"),
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       ("$ptir*$cos42","$ptir*$sin42"),
       ("$ptir*$cos45","$ptir*$sin45"),
       ("$ptir*$cos47","$ptir*$sin47"),
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       ("$ptir*$cos57","$ptir*$sin57"),
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       ("$ptor*$cos5","$ptor*$sin5"),
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       ("$ptor*$cos15","$ptor*$sin15"),
       ("$ptor*$cos17","$ptor*$sin17"),
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       ("$ptor*$cos37","$ptor*$sin37"),
```

```
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```

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("((($p/2)-$ctor)/10)+$ctor)*$cos40",
"((($p/2)-$ctor)/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)/10)+$ctor)*$cos50",
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("((($p/2)-$ctor)*2/10)+$ctor)*$cos10",
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("((($p/2)-$ctor)*3/10)+$ctor)*$cos60",
"((($p/2)-$ctor)*3/10)+$ctor)*$sin60"),
("((($p/2)-$ctor)*4/10)+$ctor)",0), !133!
("((($p/2)-$ctor)*4/10)+$ctor)*$cos10",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos20",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos30",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos40",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos50",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos60",
"((($p/2)-$ctor)*4/10)+$ctor)*$sin60"),
```

```
("((((($p/2)-$ctor)*5/10)+$ctor)",0), !140!
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos10",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin10"),
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos20",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin20"),
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos30",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin30"),
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos40",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin40"),
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos50",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin50"),
("((((($p/2)-$ctor)*5/10)+$ctor)*$cos60",
"((((($p/2)-$ctor)*5/10)+$ctor)*$sin60"),
("((((($p/2)-$ctor)*6/10)+$ctor)",0), !147!
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos10",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin10"),
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos20",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin20"),
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos30",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin30"),
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos40",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin40"),
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos50",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin50"),
("((((($p/2)-$ctor)*6/10)+$ctor)*$cos60",
"((((($p/2)-$ctor)*6/10)+$ctor)*$sin60"),
("((((($p/2)-$ctor)*7/10)+$ctor)",0), !154!
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos10",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin10"),
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos20",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin20"),
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos30",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin30"),
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos40",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin40"),
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos50",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin50"),
("((((($p/2)-$ctor)*7/10)+$ctor)*$cos60",
"((((($p/2)-$ctor)*7/10)+$ctor)*$sin60"),
("((((($p/2)-$ctor)*8/10)+$ctor)",0), !161!
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos10",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin10"),
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos20",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin20"),
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos30",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin30"),
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos40",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin40"),
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos50",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin50"),
("((((($p/2)-$ctor)*8/10)+$ctor)*$cos60",
"((((($p/2)-$ctor)*8/10)+$ctor)*$sin60"),
("((((($p/2)-$ctor)*9/10)+$ctor)",0), !168!
```

```
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos10",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin10"),
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos20",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin20"),
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos30",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin30"),
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos40",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin40"),
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos50",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin50"),
("((((($p/2)-$ctor)*9/10)+$ctor)*$cos60",
"((((($p/2)-$ctor)*9/10)+$ctor)*$sin60"),
("($p/2)*$cos10","($p/2)*$sin10"),!175!
("($p/2)*$cos20","($p/2)*$sin20"),
("($p/2)*$cos30","($p/2)*$sin30"),
("($p/2)*$cos40","($p/2)*$sin40"),
("($p/2)*$cos50","($p/2)*$sin50"),
("($p/2)*$cos60","($p/2)*$sin60")
         /10,cool //$coolreg5)

$peri6=PAR(("($orr+$co)",0),
            ("$p/2",0),
           ("($p/(2**0.5))*$cos45","-($p/(2**0.5))*$sin45"),
          ("($p/2)*$cos60/$sin60","-($p/2)"),
 ("($orr+$co)*$cos60","-($orr+$co)*$sin60"),
  ("($orr+$co+($gap4/2))*$cos50","-($orr+$co+($gap4/2))*$sin50"),
   ("($orr+$co)*$cos40","-($orr+$co)*$sin40"),
  ("($orr+$co+($gap4/2))*$cos30","-($orr+$co+($gap4/2))*$sin30"),
   ("($orr+$co)*$cos20","-($orr+$co)*$sin20"),
   ("($orr+$co+($gap4/2))*$cos10","-($orr+$co+($gap4/2))*$sin10"))

$coolreg6=PAR(12,4,180,179,178,177,176,175,2,mod;
2,175,176,177,178,179,180,174,173,172,171,170,169,168,mod;
168,169,170,171,172,173,174,167,166,165,164,163,162,161,mod;
161,162,163,164,165,166,167,160,159,158,157,156,155,154,mod;
154,155,156,157,158,159,160,153,152,151,150,149,148,147,mod;
147,148,149,150,151,152,153,146,145,144,143,142,141,140,mod;
140,141,142,143,144,145,146,139,138,137,136,135,134,133,mod;
133,134,135,136,137,138,139,132,131,130,129,128,127,126,mod;
126,127,128,129,130,131,132,125,124,123,122,121,120,119,mod;
119,120,121,122,123,124,125,118,117,116,115,114,113,112,mod;
112,113,114,115,116,117,118,13,111,110,109,108,107,106,105,
104,103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,mod;
4,12,3,mod;
!2,12,11,mod;13,111,110,109,108,107,106,105,104,
103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,11,12,4,mod;
4,12,3,mod;!
66,67,68,69,70,71,72,73,74,75,76,77,78,79,80,81,82,83,84,85,86,
87,88,15,17,65,64,63,62,61,60,59,58,57,56,55,54,53,52,51,50,49,
48,47,46,45,44,43,16,14,gap2;
11,89,90,91,92,93,94,95,96,97,98,99,100,101,102,103,104,105,106,
107,108,109,110,111,13,15,88,87,86,85,84,83,82,81,80,79,78,77,76,
75,74,73,72,71,70,69,68,67,66,14,caltub;
```

```
43,44,45,46,47,48,49,
50,51,52,53,54,55,56,57,58,59,60,61,62,63,64,65,17,18,42,41,40,
39,38,37,36,35,34,33,32,31,30,29,28,27,26,25,24,23,22,21,20,
19,16,pretub
!;20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,41,42,18,5,6,7,8,9,10,1,19,cool!)

$pincell6=PAR($peri6("$ctor",0),
          ("($p/2)","-($p/4)"),
         ("$ctor*$cos60","-$ctor*$sin60"),
         ("$ctir",0),
         ("$ctir*$cos60","-$ctir*$sin60"),
        ("$ptor",0),
        ("$ptor*$cos60","-$ptor*$sin60"),
        ("$ptir*$cos60","-$ptir*$sin60"),
        ("$ptir",0),
       ("$ptir*$cos2","-$ptir*$sin2"),
       ("$ptir*$cos5","-$ptir*$sin5"),
       ("$ptir*$cos7","-$ptir*$sin7"),
       ("$ptir*$cos10","-$ptir*$sin10"),
       ("$ptir*$cos12","-$ptir*$sin12"),
       ("$ptir*$cos15","-$ptir*$sin15"),
       ("$ptir*$cos17","-$ptir*$sin17"),
       ("$ptir*$cos20","-$ptir*$sin20"),
       ("$ptir*$cos22","-$ptir*$sin22"),
       ("$ptir*$cos25","-$ptir*$sin25"),
       ("$ptir*$cos27","-$ptir*$sin27"),
       ("$ptir*$cos30","-$ptir*$sin30"),
       ("$ptir*$cos32","-$ptir*$sin32"),
       ("$ptir*$cos35","-$ptir*$sin35"),
       ("$ptir*$cos37","-$ptir*$sin37"),
       ("$ptir*$cos40","-$ptir*$sin40"),
       ("$ptir*$cos42","-$ptir*$sin42"),
       ("$ptir*$cos45","-$ptir*$sin45"),
       ("$ptir*$cos47","-$ptir*$sin47"),
       ("$ptir*$cos50","-$ptir*$sin50"),
       ("$ptir*$cos52","-$ptir*$sin52"),
       ("$ptir*$cos55","-$ptir*$sin55"),
       ("$ptir*$cos57","-$ptir*$sin57"),
       ("$ptor*$cos2","-$ptor*$sin2"),
       ("$ptor*$cos5","-$ptor*$sin5"),
       ("$ptor*$cos7","-$ptor*$sin7"),
       ("$ptor*$cos10","-$ptor*$sin10"),
       ("$ptor*$cos12","-$ptor*$sin12"),
       ("$ptor*$cos15","-$ptor*$sin15"),
       ("$ptor*$cos17","-$ptor*$sin17"),
       ("$ptor*$cos20","-$ptor*$sin20"),
       ("$ptor*$cos22","-$ptor*$sin22"),
       ("$ptor*$cos25","-$ptor*$sin25"),
       ("$ptor*$cos27","-$ptor*$sin27"),
       ("$ptor*$cos30","-$ptor*$sin30"),
       ("$ptor*$cos32","-$ptor*$sin32"),
       ("$ptor*$cos35","-$ptor*$sin35"),
```

```
("$ptor*$cos37","-$ptor*$sin37"),
("$ptor*$cos40","-$ptor*$sin40"),
("$ptor*$cos42","-$ptor*$sin42"),
("$ptor*$cos45","-$ptor*$sin45"),
("$ptor*$cos47","-$ptor*$sin47"),
("$ptor*$cos50","-$ptor*$sin50"),
("$ptor*$cos52","-$ptor*$sin52"),
("$ptor*$cos55","-$ptor*$sin55"),
("$ptor*$cos57","-$ptor*$sin57"),
("$ctir*$cos2","-$ctir*$sin2"),
("$ctir*$cos5","-$ctir*$sin5"),
("$ctir*$cos7","-$ctir*$sin7"),
("$ctir*$cos10","-$ctir*$sin10"),
("$ctir*$cos12","-$ctir*$sin12"),
("$ctir*$cos15","-$ctir*$sin15"),
("$ctir*$cos17","-$ctir*$sin17"),
("$ctir*$cos20","-$ctir*$sin20"),
("$ctir*$cos22","-$ctir*$sin22"),
("$ctir*$cos25","-$ctir*$sin25"),
("$ctir*$cos27","-$ctir*$sin27"),
("$ctir*$cos30","-$ctir*$sin30"),
("$ctir*$cos32","-$ctir*$sin32"),
("$ctir*$cos35","-$ctir*$sin35"),
("$ctir*$cos37","-$ctir*$sin37"),
("$ctir*$cos40","-$ctir*$sin40"),
("$ctir*$cos42","-$ctir*$sin42"),
("$ctir*$cos45","-$ctir*$sin45"),
("$ctir*$cos47","-$ctir*$sin47"),
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("$ctor*$cos10","-$ctor*$sin10"),
("$ctor*$cos12","-$ctor*$sin12"),
("$ctor*$cos15","-$ctor*$sin15"),
("$ctor*$cos17","-$ctor*$sin17"),
("$ctor*$cos20","-$ctor*$sin20"),
("$ctor*$cos22","-$ctor*$sin22"),
("$ctor*$cos25","-$ctor*$sin25"),
("$ctor*$cos27","-$ctor*$sin27"),
("$ctor*$cos30","-$ctor*$sin30"),
("$ctor*$cos32","-$ctor*$sin32"),
("$ctor*$cos35","-$ctor*$sin35"),
("$ctor*$cos37","-$ctor*$sin37"),
("$ctor*$cos40","-$ctor*$sin40"),
("$ctor*$cos42","-$ctor*$sin42"),
("$ctor*$cos45","-$ctor*$sin45"),
("$ctor*$cos47","-$ctor*$sin47"),
("$ctor*$cos50","-$ctor*$sin50"),
("$ctor*$cos52","-$ctor*$sin52"),
```

```
       ("$ctor*$cos55","-$ctor*$sin55"),
       ("$ctor*$cos57","-$ctor*$sin57"),
("((($p/2)-$ctor)/10)+$ctor)",0), !112!
("((($p/2)-$ctor)/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)/10)+$ctor)*$sin60"),
("((($p/2)-$ctor)*2/10)+$ctor)",0), !119!
("((($p/2)-$ctor)*2/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)*2/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)*2/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)*2/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)*2/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)*2/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*2/10)+$ctor)*$sin60"),
("((($p/2)-$ctor)*3/10)+$ctor)",0), !126!
("((($p/2)-$ctor)*3/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)*3/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)*3/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)*3/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)*3/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)*3/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*3/10)+$ctor)*$sin60"),
("((($p/2)-$ctor)*4/10)+$ctor)",0), !133!
("((($p/2)-$ctor)*4/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)*4/10)+$ctor)*$cos60",
```

```
"-((($p/2)-$ctor)*4/10)+$ctor)*$sin60"),
(("(($p/2)-$ctor)*5/10)+$ctor)",0), !140!
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin10"),
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin20"),
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin30"),
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin40"),
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin50"),
(("(($p/2)-$ctor)*5/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*5/10)+$ctor)*$sin60"),
(("(($p/2)-$ctor)*6/10)+$ctor)",0), !147!
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin10"),
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin20"),
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin30"),
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin40"),
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin50"),
(("(($p/2)-$ctor)*6/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*6/10)+$ctor)*$sin60"),
(("(($p/2)-$ctor)*7/10)+$ctor)",0), !154!
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin10"),
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin20"),
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin30"),
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin40"),
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin50"),
(("(($p/2)-$ctor)*7/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*7/10)+$ctor)*$sin60"),
(("(($p/2)-$ctor)*8/10)+$ctor)",0), !161!
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin10"),
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin20"),
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin30"),
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin40"),
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin50"),
(("(($p/2)-$ctor)*8/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*8/10)+$ctor)*$sin60"),
```

```
("((($p/2)-$ctor)*9/10)+$ctor)",0), !168!
("((($p/2)-$ctor)*9/10)+$ctor)*$cos10",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin10"),
("((($p/2)-$ctor)*9/10)+$ctor)*$cos20",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin20"),
("((($p/2)-$ctor)*9/10)+$ctor)*$cos30",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin30"),
("((($p/2)-$ctor)*9/10)+$ctor)*$cos40",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin40"),
("((($p/2)-$ctor)*9/10)+$ctor)*$cos50",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin50"),
("((($p/2)-$ctor)*9/10)+$ctor)*$cos60",
"-((($p/2)-$ctor)*9/10)+$ctor)*$sin60"),
("($p/2)*$cos10","-($p/2)*$sin10"),!175!
("($p/2)*$cos20","-($p/2)*$sin20"),
("($p/2)*$cos30","-($p/2)*$sin30"),
("($p/2)*$cos40","-($p/2)*$sin40"),
("($p/2)*$cos50","-($p/2)*$sin50"),
("($p/2)*$cos60","-($p/2)*$sin60")
        /10,cool //$coolreg6)

$peri7=PAR(("-($orr+$co)",0),
("(-$p/2)/$sin60",0),
 !("(-$p/2)*$cos40","($p/2)*$sin40"),!
("(-$p/2)*$cos60/$sin60","($p/2)"),
("-($orr+$co)*$cos60","($orr+$co)*$sin60"),
("-($orr+$co+($gap4/2))*$cos50","($orr+$co+($gap4/2))*$sin50"),
("-($orr+$co)*$cos40","($orr+$co)*$sin40"),
("-($orr+$co+($gap4/2))*$cos30","($orr+$co+($gap4/2))*$sin30"),
("-($orr+$co)*$cos20","($orr+$co)*$sin20"),
("-($orr+$co+($gap4/2))*$cos10","($orr+$co+($gap4/2))*$sin10"))

$coolreg7=PAR(2,176,175,174,173,mod;
3,179,178,177,176,mod;
!2,3,179,178,177,176,175,174,173,mod;!
173,174,175,176,177,178,179,172,171,170,169,168,167,166,mod;
166,167,168,169,170,171,172,165,164,163,162,161,160,159,mod;
159,160,161,162,163,164,165,158,157,156,155,154,153,152,mod;
152,153,154,155,156,157,158,151,150,149,148,147,146,145,mod;
145,146,147,148,149,150,151,144,143,142,141,140,139,138,mod;
138,139,140,141,142,143,144,137,136,135,134,133,132,131,mod;
131,132,133,134,135,136,137,130,129,128,127,126,125,124,mod;
124,125,126,127,128,129,130,123,122,121,120,119,118,117,mod;
117,118,119,120,121,122,123,116,115,114,113,112,111,110,mod;
110,111,112,113,114,115,116,11,109,108,107,106,105,
104,103,102,101,100,99,98,97,96,95,94,93,92,91,90,89,
88,87,10,mod;
64,65,66,67,68,69,70,71,72,73,74,75,76,77,78,79,80,81,82,83,84,
85,86,13,15,63,62,61,60,59,58,57,56,55,54,53,52,51,50,49,
48,47,46,45,44,43,42,41,14,12,gap2;
!18,19,20,21,22,23,24,25,26,27,28,29,30,31,32,33,34,
35,36,37,38,39,40,16,4,5,6,7,8,9,1,17,cool; !
10,87,88,89,90,91,92,93,94,95,96,97,98,99,100,101,102,103,104,105,106,
```

```
107,108,109,11,13,86,85,84,83,82,81,80,79,78,77,76,
75,74,73,72,71,70,69,68,67,66,65,64,12,caltub;
41,42,43,44,45,
46,47,48,49,50,51,52,53,54,55,56,57,58,59,60,61,62,63,15,16,40,
39,38,37,36,35,34,33,32,31,30,29,28,27,26,25,24,23,22,21,20,
19,18,17,14,pretub)

$pincell7=PAR($peri7("-$ctor",0),
! ("(-$p/2)*$cos20","($p/2)*$sin20"),!
         ("-$ctor*$cos60","$ctor*$sin60"),
         ("-$ctir",0),
         ("-$ctir*$cos60","$ctir*$sin60"),
        ("-$ptor",0),
        ("-$ptor*$cos60","$ptor*$sin60"),
        ("-$ptir*$cos60","$ptir*$sin60"),
        ("-$ptir",0),
       ("-$ptir*$cos2","$ptir*$sin2"),
       ("-$ptir*$cos5","$ptir*$sin5"),
       ("-$ptir*$cos7","$ptir*$sin7"),
       ("-$ptir*$cos10","$ptir*$sin10"),
       ("-$ptir*$cos12","$ptir*$sin12"),
       ("-$ptir*$cos15","$ptir*$sin15"),
       ("-$ptir*$cos17","$ptir*$sin17"),
       ("-$ptir*$cos20","$ptir*$sin20"),
       ("-$ptir*$cos22","$ptir*$sin22"),
       ("-$ptir*$cos25","$ptir*$sin25"),
       ("-$ptir*$cos27","$ptir*$sin27"),
       ("-$ptir*$cos30","$ptir*$sin30"),
       ("-$ptir*$cos32","$ptir*$sin32"),
       ("-$ptir*$cos35","$ptir*$sin35"),
       ("-$ptir*$cos37","$ptir*$sin37"),
       ("-$ptir*$cos40","$ptir*$sin40"),
       ("-$ptir*$cos42","$ptir*$sin42"),
       ("-$ptir*$cos45","$ptir*$sin45"),
       ("-$ptir*$cos47","$ptir*$sin47"),
       ("-$ptir*$cos50","$ptir*$sin50"),
       ("-$ptir*$cos52","$ptir*$sin52"),
       ("-$ptir*$cos55","$ptir*$sin55"),
       ("-$ptir*$cos57","$ptir*$sin57"),
       ("-$ptor*$cos2","$ptor*$sin2"),
       ("-$ptor*$cos5","$ptor*$sin5"),
       ("-$ptor*$cos7","$ptor*$sin7"),
       ("-$ptor*$cos10","$ptor*$sin10"),
       ("-$ptor*$cos12","$ptor*$sin12"),
       ("-$ptor*$cos15","$ptor*$sin15"),
       ("-$ptor*$cos17","$ptor*$sin17"),
       ("-$ptor*$cos20","$ptor*$sin20"),
       ("-$ptor*$cos22","$ptor*$sin22"),
       ("-$ptor*$cos25","$ptor*$sin25"),
       ("-$ptor*$cos27","$ptor*$sin27"),
       ("-$ptor*$cos30","$ptor*$sin30"),
       ("-$ptor*$cos32","$ptor*$sin32"),
       ("-$ptor*$cos35","$ptor*$sin35"),
```

```
("-$ptor*$cos37","$ptor*$sin37"),
("-$ptor*$cos40","$ptor*$sin40"),
("-$ptor*$cos42","$ptor*$sin42"),
("-$ptor*$cos45","$ptor*$sin45"),
("-$ptor*$cos47","$ptor*$sin47"),
("-$ptor*$cos50","$ptor*$sin50"),
("-$ptor*$cos52","$ptor*$sin52"),
("-$ptor*$cos55","$ptor*$sin55"),
("-$ptor*$cos57","$ptor*$sin57"),
("-$ctir*$cos2","$ctir*$sin2"),
("-$ctir*$cos5","$ctir*$sin5"),
("-$ctir*$cos7","$ctir*$sin7"),
("-$ctir*$cos10","$ctir*$sin10"),
("-$ctir*$cos12","$ctir*$sin12"),
("-$ctir*$cos15","$ctir*$sin15"),
("-$ctir*$cos17","$ctir*$sin17"),
("-$ctir*$cos20","$ctir*$sin20"),
("-$ctir*$cos22","$ctir*$sin22"),
("-$ctir*$cos25","$ctir*$sin25"),
("-$ctir*$cos27","$ctir*$sin27"),
("-$ctir*$cos30","$ctir*$sin30"),
("-$ctir*$cos32","$ctir*$sin32"),
("-$ctir*$cos35","$ctir*$sin35"),
("-$ctir*$cos37","$ctir*$sin37"),
("-$ctir*$cos40","$ctir*$sin40"),
("-$ctir*$cos42","$ctir*$sin42"),
("-$ctir*$cos45","$ctir*$sin45"),
("-$ctir*$cos47","$ctir*$sin47"),
("-$ctir*$cos50","$ctir*$sin50"),
("-$ctir*$cos52","$ctir*$sin52"),
("-$ctir*$cos55","$ctir*$sin55"),
("-$ctir*$cos57","$ctir*$sin57"),
("-$ctor*$cos2","$ctor*$sin2"),
("-$ctor*$cos5","$ctor*$sin5"),
("-$ctor*$cos7","$ctor*$sin7"),
("-$ctor*$cos10","$ctor*$sin10"),
("-$ctor*$cos12","$ctor*$sin12"),
("-$ctor*$cos15","$ctor*$sin15"),
("-$ctor*$cos17","$ctor*$sin17"),
("-$ctor*$cos20","$ctor*$sin20"),
("-$ctor*$cos22","$ctor*$sin22"),
("-$ctor*$cos25","$ctor*$sin25"),
("-$ctor*$cos27","$ctor*$sin27"),
("-$ctor*$cos30","$ctor*$sin30"),
("-$ctor*$cos32","$ctor*$sin32"),
("-$ctor*$cos35","$ctor*$sin35"),
("-$ctor*$cos37","$ctor*$sin37"),
("-$ctor*$cos40","$ctor*$sin40"),
("-$ctor*$cos42","$ctor*$sin42"),
("-$ctor*$cos45","$ctor*$sin45"),
("-$ctor*$cos47","$ctor*$sin47"),
("-$ctor*$cos50","$ctor*$sin50"),
("-$ctor*$cos52","$ctor*$sin52"),
```

```
       ("-$ctor*$cos55","$ctor*$sin55"),
       ("-$ctor*$cos57","$ctor*$sin57"),
("-((($p/2)-$ctor)/10)+$ctor)",0), !110!
("-((($p/2)-$ctor)/10)+$ctor)*$cos10","((($p/2)-
$ctor)/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos20","((($p/2)-
$ctor)/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos30","((($p/2)-
$ctor)/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos40","((($p/2)-
$ctor)/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos50","((($p/2)-
$ctor)/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)/10)+$ctor)*$cos60","((($p/2)-
$ctor)/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*2/10)+$ctor)",0), !117!
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos10","((($p/2)-
$ctor)*2/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos20","((($p/2)-
$ctor)*2/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos30","((($p/2)-
$ctor)*2/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos40","((($p/2)-
$ctor)*2/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos50","((($p/2)-
$ctor)*2/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*2/10)+$ctor)*$cos60","((($p/2)-
$ctor)*2/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*3/10)+$ctor)",0), !124!
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos10","((($p/2)-
$ctor)*3/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos20","((($p/2)-
$ctor)*3/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos30","((($p/2)-
$ctor)*3/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos40","((($p/2)-
$ctor)*3/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos50","((($p/2)-
$ctor)*3/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*3/10)+$ctor)*$cos60","((($p/2)-
$ctor)*3/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*4/10)+$ctor)",0), !131!
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos10","((($p/2)-
$ctor)*4/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos20","((($p/2)-
$ctor)*4/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos30","((($p/2)-
$ctor)*4/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos40","((($p/2)-
$ctor)*4/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos50","((($p/2)-
$ctor)*4/10)+$ctor)*$sin50"),
```

```
("-((($p/2)-$ctor)*4/10)+$ctor)*$cos60","((($p/2)-
$ctor)*4/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*5/10)+$ctor)",0), !138!
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos10","((($p/2)-
$ctor)*5/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos20","((($p/2)-
$ctor)*5/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos30","((($p/2)-
$ctor)*5/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos40","((($p/2)-
$ctor)*5/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos50","((($p/2)-
$ctor)*5/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*5/10)+$ctor)*$cos60","((($p/2)-
$ctor)*5/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*6/10)+$ctor)",0), !145!
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos10","((($p/2)-
$ctor)*6/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos20","((($p/2)-
$ctor)*6/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos30","((($p/2)-
$ctor)*6/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos40","((($p/2)-
$ctor)*6/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos50","((($p/2)-
$ctor)*6/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*6/10)+$ctor)*$cos60","((($p/2)-
$ctor)*6/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*7/10)+$ctor)",0), !152!
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos10","((($p/2)-
$ctor)*7/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos20","((($p/2)-
$ctor)*7/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos30","((($p/2)-
$ctor)*7/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos40","((($p/2)-
$ctor)*7/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos50","((($p/2)-
$ctor)*7/10)+$ctor)*$sin50"),
("-((($p/2)-$ctor)*7/10)+$ctor)*$cos60","((($p/2)-
$ctor)*7/10)+$ctor)*$sin60"),
("-((($p/2)-$ctor)*8/10)+$ctor)",0), !159!
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos10","((($p/2)-
$ctor)*8/10)+$ctor)*$sin10"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos20","((($p/2)-
$ctor)*8/10)+$ctor)*$sin20"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos30","((($p/2)-
$ctor)*8/10)+$ctor)*$sin30"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos40","((($p/2)-
$ctor)*8/10)+$ctor)*$sin40"),
("-((($p/2)-$ctor)*8/10)+$ctor)*$cos50","((($p/2)-
$ctor)*8/10)+$ctor)*$sin50"),
```

```
("-((((($p/2)-$ctor)*8/10)+$ctor)*$cos60","((((($p/2)-
$ctor)*8/10)+$ctor)*$sin60"),
("-((((($p/2)-$ctor)*9/10)+$ctor)",0), !166!
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos10","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin10"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos20","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin20"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos30","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin30"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos40","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin40"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos50","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin50"),
("-((((($p/2)-$ctor)*9/10)+$ctor)*$cos60","((((($p/2)-
$ctor)*9/10)+$ctor)*$sin60"),
("-($p/2)",0),!173!
("-($p/2)*$cos10","($p/2)*$sin10"),!174!
("-($p/2)*$cos20","($p/2)*$sin20"),
("-($p/2)*$cos30","($p/2)*$sin30"),
("-($p/2)*$cos40","($p/2)*$sin40"),
("-($p/2)*$cos50","($p/2)*$sin50"),
("-($p/2)*$cos60","($p/2)*$sin60")
         /9,cool//$coolreg7)

pc1=STR($pincell1)
pc2=STR($pincell2)
pc3=STR($pincell3)
pc4=STR($pincell4)
pc5=STR($pincell5)
pc6=STR($pincell6)
pc7=STR($pincell7)
$k=PAR(4)
ring1=CNX(pc2,pc2,pc2,pc2,pc2,pc2/
(1,8,9)$k(2,2,1)/
(2,8,9)$k(3,2,1)/
(3,8,9)$k(4,2,1)/
(4,8,9)$k(5,2,1)/
(5,8,9)$k(6,2,1))
!(6,8,9)$k(1,2,1))!
ring2=CNX(pc3,pc7,pc5,pc6,pc7,pc4/
      (1,5,4)$k(2,1,2)/
      (2,4,3)$k(3,1,2)/
(3,5,4)$k(4,1,2)/
(4,5,4)$k(5,1,2)/
(5,4,3)$k(6,1,2))
!(6,1,2)$k(1,1,2))!

system=CNX(pc1,ring1,ring2/
(1,5,6)$k(2-1,1,9)/
(2-1,2,3)$k(3-1,1,10))

system=BDRY((3-1,3,3)$k(white))
ov1=OVLM('D2O-288'/*-*-**/
```

```
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)
!ov11=OVLM('D2O-288a'/*-*-0-cool)
ov12=OVLM('D2O-288b'/*-*-0-cool)
ov13=OVLM('D2O-288c'/*-*-0-cool)
ov14=OVLM('D2O-288d'/*-*-0-cool)!
!ov11=OVLM('D2O-288a'/*-*-**/
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)!
!ov12=OVLM('D2O-288b'/*-*-**/
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)!
!ov13=OVLM('D2O-288c'/*-*-**/
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)!
!ov14=OVLM('D2O-288d'/*-*-**/
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)!
!ov15=OVLM('D2O-288e'/*-*-**/
'Zr-1'/*-*-*-clad/
'Zr-2'/*-*-0-caltub/
'Zr-3'/*-*-0-pretub/
'UO2-687'/*-*-*-fuel/
nitr/*-*-*-gap1/
cadi/*-*-0-gap2/
'D2O-68'/*-*-0-mod)!
w1=OVSM(ov1)
!w11=OVSM(w1/ov11)
```

```
w12=OVSM(w11/ov12)
w13=OVSM(w12/ov13)
w14=OVSM(w13/ov14)
w15=OVSM(ov15)!

temp1=OVLT(561.15/*-*-**/
609.15/*-*-*-clad/
960.15/*-*-*-fuel/
960/*-*-*-gap1/
341.15/*-*-0-mod/
450.15/*-*-0-gap2/
340.15/*-*-0-caltub/
560.15/*-*-0-pretub)
w2=OVST(temp1)
void=OVLD(1.0/*-*-**)
void1=OVLD(!1.0/*-*-**/!0.95/*-*-0-cool)
void2=OVLD(!1.0/*-*-**/!0.9/*-*-0-cool)
void3=OVLD(!1.0/*-*-**/!0.85/*-*-0-cool)
void4=OVLD(!1.0/*-*-**/!0.5/*-*-0-cool)
void5=OVLD(!1.0/*-*-**/!0.00005/*-*-0-cool)
void6=OVLD(!1.0/*-*-**/!0.00005/*-*-0-cool/
0.95/*-*-0-mod)
void7=OVLD(!1.0/*-*-**/!0.00005/*-*-0-(cool,mod))
w3=OVSD(void)
w31=OVSD(w3/void1)
w32=OVSD(w3/void2)
w33=OVSD(w3/void3)
w34=OVSD(w3/void4)
w35=OVSD(w3/void5)
w36=OVSD(w3/void6)
w37=OVSD(w3/void7)
$pw=PAR(32)
ist=STAT(w1,w3,w2,$pw)
ist1=STAT(w1,w31,w2,$pw)
ist2=STAT(w1,w32,w2,$pw)
ist3=STAT(w1,w33,w2,$pw)
ist4=STAT(w1,w34,w2,$pw)
ist5=STAT(w1,w35,w2,$pw)
ist6=STAT(w1,w36,w2,$pw)
ist7=STAT(w1,w37,w2,$pw)

burnup=PATH(/(ist),50,100,200,300,400,500,
750,1000,1250,1500,2000,2500,3000,3500,4000,
4500,5000,5500,6000,6500,7000,7500,8000,9000,
10000,11000,12000,13000,14000,15000,16000)

branches=TREE(burnup/P,(ist1),P,(ist2),P,(ist3),
P,(ist4),P,(ist5),P,(ist6),P,(ist7)/
50,1500,3000,6000,6500,9000)

gr1=GROUP(N/0)
gr2=GROUP(N/0.625,0)
grall=GROUP(N/)
```

```
allregs=AREA(*-*-**)
assembly=AREA(<*-*-**>)
fuelmap=AREA(*-*-*-<fuel>)
fuelmap1=AREA(<*-*-*-fuel>)
center=AREA(<1-1-fuel>)
rg1=AREA(<2-*-1-fuel>)
rg2=AREA(<2-*-(2,3)-fuel>)
rg3=AREA(<2-*-(4-6)-fuel>)

iscom1=MICRO(gr1,allregs!assembly!/
92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603/
!ab,fi,nf,kf,ed!)

iscom2=MICRO(grall,fuelmap/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

iscom3=MACRO(gr2,fuelmap/ab,fi,nf,kf,ch)

nmde1=MICRO(gr2,fuelmap/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
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43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

nmde2=MICRO(gr2,fuelmap1/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
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38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

nmde3=MICRO(gr2,center/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

nmde4=MICRO(gr2,rg1/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

nmde5=MICRO(gr2,rg2/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

nmde6=MICRO(gr2,rg3/92235,92238,94238,
94239,94240,94241,94242,93237,95241,95242,95243,
64155,62649,60643,45103,55633,54631,64657,62652,
43599,63653,60645,62651,62650,42595,59641,57639,
62647,36583,60644,44601,63651,40593,53629,56637,
42597,46605,47609,63654,55635,60642,40591,44602,
39589,60646,44600,60648,64656,46608,46607,48113,
38590,63655,61647,55634,54634,45605,61648,61748,
63656,61649,59643,54633,58641,62653,60647,44603
/ab,fi,nf,kf)

iscom4=MACRO(gr2,allregs/ab,fi,nf,kf,ch)
Case8=RUN(BSQ:0.00002539)
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APPENDIX G

Capacity factor calculations for the CIRUS reactor

Burnup required to achieve weapons-grade plutonium quality = 1066.67 MWd/tU

This is equivalent to (1066.67 / 365.25) MW-yrs/tU = 2.92 MW-yrs/tU

Total quantity of fuel used by CIRUS = 10.5 tons of UO_2 = 10.5*(238/270) tU = 9.25 tU

So the total heat produced from 9.25 tU is 9.25 tU * 2.92 MW-yrs/tU = 27.03 MW-yrs

For capacity factor over a year with 280 days of operation at 40 MWth the calculation is given by:

(27.03 MW-yrs / 40 MWth) * (280 days/ 365.25 days) = 0.5179 ~**50%**

Capacity factor calculations for the DHRUVA reactor

Burnup required to achieve weapons-grade plutonium quality = 1055 MWd/tU

This is equivalent to (1055 / 365.25) MW-yrs/tU = 2.89 MW-yrs/tU

Total quantity of fuel used by DHRUVA = 6.35 tons of UO_2 in one core change = 6.35*(238/270) tU = 5.597 tU, so for 5 core changes the fuel quantity = 27.98 tU

So the total heat produced from 27.98 tU is 27.98 tU * 2.89 MW-yrs/tU = 80.88 MW-yrs

For capacity factor over a year with ((5 core changes / year) * (67 days of operation / core change)) at 100 MWth the calculation is given by:

(80.88 MW-yrs / 100 MWth) * (5 * 67 days/ 365.25 days) = 0.7418 ~**75%**

APPENDIX H

The standard designs speed of a nuclear submarine would be around 30–35 knots (30 knots = 15 m/sec). Estimates of the ^{235}U (100% Enriched) for U.S. nuclear submarines is close to 0.6–0.7 grams per shaft horse power per year. While the requirements of Russian submarines are likely to be about 0.315-0.35 grams/shaft horse power/year. The difference is because of the fast reactors with fuel enrichments being used by the Russian submarines and PWR cores by the U.S. submarines. The propulsion power for Charlie Class Submarine is 20,000 shp. Due to the smaller distances that the ATV (Advanced Technology Vessel, name coined for the nuclear submarine of India) is likely to traverse, it could be assumed that ATV would require about 0.3 grams/shaft horse power/year (1 shp = 746 watts). Considering propulsion power of 20,000 shp and refueling requirements in every 15 years, the ATV requirement for ^{235}U
= 0.3 grams/shp/year * 20,000 shp * 15 years = 90 kilograms ^{235}U. That is equal to 100 kilograms of 90% w/o ^{235}U.

The Uranium enrichment activity of Rattehali plant became operational in 1990 with a higher end speculated installed capacity of 2000 SWU/year and centrifuge's capacity of 3 SWU/machine/year. Considering the facilities capabilities over a period of 10 years, it can be said that the separated effort is 10 years * 2000 SWU/year = 20,000 SWU. Now the feed required is ~ 213Kg of natural uranium to produce 1 kilogram of 90% ^{235}U. Therefore the total expected 90% ^{235}U accumulation over this period is 20,000/192 = 104.4 Kgs. This is nearly equivalent to the requirement of the one core of fuel for ATV, computed earlier.

VITA

Taraknath Woddi Venkat Krishna was born to Pattabhiram Woddi and Basanti Patra in Berhampur (city), Orissa (state), India. He graduated from the College of Engineering and Technology of Orissa University of Agriculture and Technology and received his Bachelor of Technology in Instrumentation & Electronics in July 1996. Later, he went on to graduate with a Masters in nuclear engineering from Texas A&M University in May of 2005 after working as a reactor operator for 6 years in a nuclear power plant. In 2007 he is graduating with Ph.D. in nuclear engineering. He is happily married to a sweet girl Divya Epari. His departmental address is Department of Nuclear Engineering c/o Dr. William Charlton, 129 Zachry Engineering Center, Texas A&M University, College Station, Texas - 77843.