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# The evolution of the Indian nuclear power programme

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ABSTRACT

The strategy for growth of nuclear power in India was planned nearly sixty years ago, noting the rather small uranium and large thorium reserves in the country. This has prompted India to adopt the well-known three stage programme. The first stage is primarily based on the Pressurised Heavy Water Reactors (PHWRs). The evolution of technology of PHWRs is discussed in the first part of this paper. India has constructed eighteen PHWRs, has achieved impressive availability factors and some of these reactors have achieved annual capacity factors of nearly hundred percent in the recent past. Having installed a nuclear power generation capacity of 6780 MWe (4460 MWe from PHWRs, 2000 MWe from PWRs and 320 MWe from BWRs), India is now poised to launch a major expansion programme. This will be based on the increased availability of uranium from import and from the augmented domestic supply. In the immediate future, the nuclear power capacity will grow by installing a series of indigenous PHWRs in addition to light water reactors built under international civil nuclear cooperation agreements. The growth of nuclear capacity in this period is aimed at increasing the share of nuclear power in meeting the base-load demand of non-carbon electricity required for the rapid economic growth in the country.

India embarked on its second stage programme with the successful operation of a research reactor named Fast Breeder Test Reactor (FBTR). Based on the experience of FBTR and following the development of all the required enabling technologies in India, the Prototype Fast Breeder Reactor (PFBR) of 500 MWe (gross) capacity has been designed, constructed and is now at an advanced stage of commissioning. A large increase in the nuclear power generation capacity is envisaged through deployment of fast breeder reactors. These reactors will not only help in building up nuclear power capacity but also, in due course, enable conversion of thorium into fissile $U^{233}$, which will fuel the reactors in the third stage of India's nuclear programme. The adoption of the closed fuel cycle for both thermal and fast reactors has dual objectives: multiplying the fissile inventory by fertile to fissile conversion and reducing the burden of long lived radioactive waste-both being essential for attaining near sustainability of nuclear power.

India has designed an Advanced Heavy Water Reactor (AHWR). This reactor can use one of the following fuels:

(i) Low enriched uranium dioxide-thorium dioxide based mixed oxide (MOX) fuel primarily in once through mode.
(ii) Plutonium dioxide – thorium dioxide MOX fuel in the closed fuel cycle mode. Eventually when enough of $U^{233}$ is generated the reactor can operate with the $U^{233}$– thorium MOX fuel.

A large number of passive safety features are incorporated in this design.

As a part of the long term objective of the development of Accelerator Driven Sub-critical Systems (ADSS), work on the development of high power linear accelerator has been initiated.


## 1. Introduction

India started its nuclear programme at a time when the industrial infrastructure in the country was at its infancy. There was no

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research base from where a coordinated and mission oriented technology development activity could be initiated. At that time research activities were essentially confined to a limited number of university departments where facilities for large scale experimentation were not available. Homi Bhabha who initiated the nuclear programme in India recognised the importance of basic research and training of professionals in the early phase of programme. Right from the beginning the Indian nuclear programme was driven by an ambition of maximising the use of indigenous technologies and raw materials. The results of geological survey indicated that while the uranium reserve in the country was quite limited, there existed a huge reserve of thorium (Bhabha and Prasad, 1958; Ramanna, 1987) in the monazite bearing areas on the long Indian coastline. In the 1950s, worldwide, there was an expectation that nuclear energy could meet a major part of world's energy requirement and could sustain the same for a long time (Lewis, 1972; The History of Nuclear Energy (2016)).

A modest uranium reserve and a large thorium resource in India led to the adoption of the three stage programme (Bhabha and Prasad, 1958; Kakodkar, 1998). The first stage is primarily based on the Pressurised Heavy Water Reactors (PHWRs), which use natural uranium dioxide as the fuel and heavy water as both moderator and coolant. Due to the excellent neutron economy, the energy extracted per tonne of mined uranium in PHWRs was known to be better than other thermal reactors. The selection of PHWRs was appropriate also from the considerations that isotope enrichment technology and the manufacturing infrastructure for large size pressure vessels were not available in the country at that point of time. In order to be self-sufficient, development of technologies related to the manufacturing of PHWR-fuel and the heavy water production, were taken up quite early in the programme.

The second stage is primarily aimed at providing the needed additional fissile materials for the required sustainable growth of the installed nuclear power capacity by converting fertile nuclides into fissile ones. Plutonium fuelled fast reactors, due to their associated high breeding ratio, were the best choice for the second stage. Keeping this in view a closed fuel cycle was chosen and work on reprocessing of the PHWR spent fuel was initiated quite early. The advantage of the closed fuel cycle towards achieving a significant reduction of the radiotoxic burden of nuclear waste was also recognised (Balu et al., 1998). With the installation of the Fast Breeder Test Reactor (FBTR) in 1985 India entered the second stage of its programme (Paranjape and Sundaram, 1985). With the experience gained in operating FBTR and in handling molten sodium, the Prototype Fast Breeder Reactor (PFBR) of a capacity 500 MWe (gross) was designed. The design and development of PFBR was done by IGCAR, with the help of major experimental facilities for design validation, the support of Indian academic and other research institutions for additional specialised computational and experimental research based inputs, and the support of Indian industry for the development of manufacturing technologies for special equipment. The reactor has been constructed and is now nearly ready for commissioning.

Large scale utilisation of thorium for nuclear power was envisaged in the third stage. Once the installed nuclear capacity reaches a desired level and an adequate inventory of fissile nuclides is accumulated India will be ready to launch the third stage reactors with the $U^{233}$-Th fuel cycle. Fission characteristics of $U^{233}$ are such that breeding is possible in a wide spectrum of neutron energy from thermal to fast, though not as efficiently as $Pu^{239}$ in the fast neutron spectrum (>1 MeV). Power generation capacity established at that point of time can be sustained over several centuries with the operation of $U^{233}$- thorium fuel cycle which also has the advantage of generating much less long-lived radioactive waste.

With the increased availability of uranium from augmented domestic and foreign sources (under the international civil nuclear cooperation), the scope of the first stage of the Indian nuclear programme has been extended to include Light Water Reactors (LWRs) of foreign as well as Indian designs, along with the Indian PHWRs. The latest generation of the Indian PHWRs is designed for 700 MWe (gross) capacity.

As the opening paper of this issue, the present paper gives a brief account of the technology evolution, which has happened over the last six decades. The papers which follow elaborate the course of development in different technology domains. The present paper is divided in different sections. Section - 2 deals with the setting up of R&D facilities including research reactors and creation of industrial scale infrastructure for production of nuclear fuel, heavy water and electronic control systems. Indian contributions to the development of PHWR technology, which are the mainstay of Indian nuclear power programme today, are discussed in section- 3. The development of fast reactors, which forms the second stage, is discussed in section - 4. A study on the evolving fuel cycle with PHWR – Light Water Reactors (LWR) – Fast reactors in tandem is discussed in section – 5.0. Long-term perspectives of nuclear energy and accelerator driven systems are discussed in section-6 and -7 respectively.

## 2. Setting up of R&D and industrial scale facilities

India's nuclear programme, initiated just after the country became an independent sovereign state, was based on the following fundamental principles: (a) It should be indigenous both in terms of technology and materials and (b) It should grow in a manner that a significant part of the energy demand of the country can be met from nuclear energy in the long run without adversely affecting the environment. A great visionary, Homi Bhabha created the blueprint of this programme which was initiated with the formation of Indian Atomic Energy Commission in 1948. Due emphasis was given to basic research, development of scientific manpower and a stepwise capacity building in nuclear technology and nuclear materials (Ramanna, 1987; Sundaram et al., 1998; Anderson, 2010).

In order to achieve self-sufficiency in nuclear technology a comprehensive research facility was created in Atomic Energy Establishment, Trombay (AEET), which was later renamed as Bhabha Atomic Research Centre (BARC) after Bhabha's demise in 1966. Research teams in all disciplines relevant for the development of the nuclear technology in its entirety were nucleated and nurtured-which included reactor physics and engineering, electronics and control, materials science and processing, health physics and safety engineering, chemical engineering and reprocessing technology apart from core science disciplines.

The importance of the development of human resources was recognised right from the inception of the Indian atomic energy establishment. The BARC Training School was established in 1957 to provide postgraduate education in nuclear science and engineering to a very select group of fresh graduates in basic sciences and relevant branches of engineering. This institution has turned out over 8000 professionals in Physics, Chemistry, and Biology and in different branches of Engineering. A great majority of scientists and engineers who have contributed to the development of nuclear technology in India are graduates of BARC Training School. The fact that lectures and laboratory training are provided by specialists in respective fields has been responsible for the high quality of the education. With the establishment of Homi Bhabha National Institute in 2006 the opportunities for continuing education and acquiring higher academic degrees for the researchers in institutions of the Department of Atomic Energy (DAE) have greatly enhanced.

To initiate research in reactor physics and reactor engineering, the building of a research reactor, named APSARA (Prasad and Rao, 1959), was taken up as one of the first activities. This swimming pool type reactor which attained criticality in 1956 has the distinction of being the first nuclear reactor of Asia. Highly enriched uranium-aluminium alloy (supplied by United Kingdom Atomic Energy Authority) was used as the fuel and light water as moderator and coolant. This research reactor of 1 MWt power rating has been extensively used for production of useful isotopes, basic research in nuclear sciences and some engineering experiments. It was decommissioned in 2010 and a new 2 MWt swimming pool type research reactor is now being constructed. Since India's first stage nuclear programme was based primarily on natural uranium fuelled and heavy water moderated reactor, a 40 MWt research reactor CIRUS was constructed with the Canadian collaboration (Sage et al., 1958). This reactor achieved criticality in 1960. It used natural uranium metal as fuel in the form of cylindrical rods, with aluminium clad and heavy water as moderator and light water as coolant. This reactor was used in production of useful isotopes, testing of nuclear fuels and structural materials, basic research in nuclear sciences and research in neutron scattering, crystallography and biology. This reactor helped in generating considerable experience in reactor operation and in solving many problems of power reactors till it was shut down permanently in 2010. In 1961, a 100 Wt reactor named ZERLINA was commissioned using natural uranium as fuel with aluminium clad and heavy water as moderator and coolant and shutoff rods made up of cadmium (Deniz, 1963). The purpose of this reactor was to study reactor physics of different lattice configurations of fuel in a heavy water moderated reactor. This experimental reactor facility was decommissioned in 1983 after it generated large quantum of data on various lattice configurations. A 100 MWt reactor named DHRUVA designed and constructed indigenously achieved criticality in 1985 (Moorthy et al., 1991; Agarwal et al., 2006; Shukla, 2007). This reactor uses natural uranium metal as fuel cladded with an aluminium alloy and heavy water as moderator and coolant. Apart from generating useful isotopes this reactor provides facilities for research in both basic sciences and engineering. A fast critical facility, PURNIMA which achieved criticality in 1972 was designed and constructed using plutonium dioxide as fuel to get experience in plutonium handling, to study the behaviour of plutonium fuel and for validating the reactor physics design and nuclear data for fast reactors (Iyengar et al., 1979). For a focussed attention to the development of the fast reactor technology Reactor Research Centre, later named as Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam was established. Major facilities in this centre include FBTR, molten sodium loop, post irradiation examination and materials testing, and reprocessing of fast reactor fuel. A 30 kW research reactor using $U^{233}$ – aluminium alloy as fuel, named KAMINI is installed at Kalpakkam primarily for neutron radiography work. The high temperature fission chambers, to be used for neutron flux measurement in PFBR, are tested in this reactor. To demonstrate some of the enabling technologies necessary for the third stage, the design of Advanced Heavy Water Reactor (AHWR) is made in BARC. In the equilibrium core configuration AHWR (Pu- $U^{233}$ based fuel version) will produce nearly two third of its power from $U^{233}$. This design also incorporates several unique passive safety features and it meets to a large extent the criteria for advanced nuclear energy systems identified by the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) forum of International Atomic Energy Agency (IAEA). Several critical experiments are underway before the start of construction of this reactor. A Critical Facility (CF) for AHWR and 540 MWe PHWR was constructed (Raina, 2016) specifically for validating the reactor physics design and relevant nuclear data. Table 1 gives a brief summary of all Indian research reactors.

From the inception of the nuclear programme in India the search for domestic resources of nuclear materials was initiated by setting up Atomic Minerals Division (AMD). Developmental activities on processing of materials leading to manufacturing of nuclear fuels and structural materials, on heavy water technology, on reprocessing and waste management technologies and on electronic control systems were all initiated and expanded in parallel tracks at BARC. As the size of the programme grew, industrial scale facilities were created, all under the umbrella of Department of Atomic Energy (DAE). Uranium Corporation of India (UCIL), Indian Rare Earths (IRE) and Nuclear Fuel Complex (NFC) were set up to provide the entire infrastructure from mining to the production of finished nuclear fuels and structural components. After mastering the technology of production of heavy water, industrial scale plants were commissioned to meet the growing requirement of the heavy water. Today the installed capacity of heavy water production meets the entire requirement of PHWRs, currently operating and under construction. Some quantity of heavy water is also being exported mainly for research and non-nuclear applications. Electronics Corporation of India (ECIL) has a robust technology base to supply all the sophisticated instrumentation for the nuclear power sector. Several Indian private sector and public sector industrial organisations rose to the occasion to take up the challenge for manufacturing nuclear grade components and equipment such as calandria vessels, end fittings, steam generators, turbines etc for nuclear power plants.

India entered the nuclear power era in 1969 with the installation of two Boiling light Water Reactors (BWRs) at Tarapur imported from General Electric (GE) Company of United State of America on turnkey basis (Chakravarti and Srinivasan, 1964). Tarapur Atomic Power Stations (TAPS) units- 1&2 have been operating for over forty-five years. The licensing design basis of these reactors was as per original design of GE. Design modifications were carried out by India from time to time based on safety reviews and based on Indian as well as world experience. The secondary steam generators (SSGs) of these reactors were having the problem of tube leaks right from the beginning. In order to limit recurring personnel radiation exposures and operational problems with large number of tubes that were required to be plugged, a decision was taken to isolate the SSGs from the steam and feed water side, and the units were de-rated from 210 MWe to 160 MWe in 1984. These reactors are fuelled with low enriched uranium dioxide. Some experimental uranium and plutonium Mixed Oxide (MOX) fuel clusters were irradiated in these reactors when there were some difficulties in the procurement of the required enriched uranium fuel.

A comprehensive review of the units was taken up in 2002–2006 wherein the design basis review of TAPS safety system was taken up as a relicensing activity. After a comprehensive regulatory review followed by safety up-gradations the two BWRs were given extension of license for five years till March 2011. A large programme for extensive inspection of surveillance coupons, reactor pressure vessel and core shroud welds was taken up by Nuclear Power Corporation of India Limited (NPCIL) with the technical support of BARC, and based on the outcome the operating licence of the reactors has been further incrementally extended. Following the Fukushima accident in 2011, all Indian nuclear power plants were reviewed for their capability to withstand external events of severe nature. As part of this exercise and subsequent follow-up actions, several additional modifications and augmentation have been carried out in the TAPS BWRs too.

## 3. Indigenisation of the PHWR technology

The work on PHWR technology started in India with the

**Table 1**
Indian research reactors.

| Name | Year | Fuel | Moderator/coolant | Power | Purpose |
|---|---|---|---|---|---|
| APSARA | 1956 | HEU[a] - Al | $H_2O/H_2O$ | 1 MWt | Isotope production |
| CIRUS | 1960 | Natural U | $D_2O/H_2O$ | 40 MWt | Isotope production and Basic science Research |
| ZERLINA | 1961 | Natural U | $D_2O/D_2O$ | 100 W | Reactor physics research |
| PURNIMA | 1972 | $PuO_2$ | Nil/Air | 1 W | To gain experience in plutonium handling |
| DHRUVA | 1985 | Natural U | $D_2O/D_2O$ | 100 MWt | Isotope production and Basic science Research |
| FBTR | 1985 | (Pu + U)C | Nil/Liq. Sodium | 40 MWt | To gain experience in sodium technology and qualify critical componants for fast reactors |
| KAMINI | 1996 | $U^{233}$- Al | $H_2O/H_2O$ | 30 kW | Neutron radiography |
| Critical facility (AHWR) | 2008 | Natural U/Thorium | $D_2O/D_2O$ | 100 W | Validation of reactor physics design |

[a] Highly Enriched Uranium.

construction of Rajasthan Atomic Power station (RAPS) unit-1 with the power output of 200 MWe (gross). The RAPS-1 was constructed with Canadian collaboration and achieved criticality in 1972. Its design was similar to the Canadian Douglas Point reactor. The design of PHWRs, subsequently constructed in India (Prasad, 1996) went through a series of evolutions pertaining to control and shut down systems, fuel handling system, containment etc. From the installation of Narora Atomic Power Stations (NAPS) unit-1 in 1989, the 220 MWe (gross) Indian PHWR design was standardised.

The moderator level control mechanism which is used for raising/lowering reactor power in RAPS-1 and 2 and MAPS-1 and 2 was replaced by shim rods (2 in number) in 220 MWe PHWRs constructed subsequently (Bajaj and Gore, 2006). The shutdown system was also changed. The moderator dump system was replaced by two independent fast acting and diverse shutdown systems – primary (PSS) and secondary (SSS). While PSS comprises 14 vertical shutoff rods, SSS is constituted by 12 liquid poison tubes in which lithium pentaborate solution is introduced whenever required. Automatic Liquid Poison Addition System (ALPAS) which adds boron solution into moderator through moderator recirculation line was introduced in NAPS onwards. This is further augmented by a system known as Gravity Addition of Boron (GRAB). Kaiga onwards, ALPAS and GRAB were replaced by the Liquid Poison Injection System (LPIS) (Soni et al., 1997) which injects boron solution directly in to moderator.

Introduction of larger reactors – 540 MWe (gross) and 700 MWe (gross) – required major changes in the design of reactor control and shutdown systems. The large size of the reactor core being neutronically loosely coupled necessitated dividing the core into several zones which are monitored and controlled independently. The control system of 540/700 MWe Indian PHWRs (Bajaj and Gore, 2006; Bhardwaj, 2006) consists of 17 adjuster rods, 4 mechanical control rods and zone controller units made of six tubes in which light water level changes in different compartments. The configuration of six tubes in which light water level is controlled in different zones is shown in Fig. 1.

There are two independent fast acting and diverse Shut Down Systems (SDS) in 540/700 MWe reactors. The SDS - 1 consists of twenty-eight vertical gravity driven mechanical shut off rods. The SDS - 2 has six horizontal tubes in the direction perpendicular to both shut off rods and fuel channels. These tubes have perforations through which gadolinium nitrate solution is added to the moderator. The progressive changes introduced in PHWRs are summarised in Appendix-1 where the reactors are listed chronologically. Bulk power monitoring in 220 MWe PHWRs is done using signals from reactor regulating system. The channel outlet temperature is monitored using thermocouples in each coolant channel. Since 540 MWe and 700 MWe PHWRs are neutronically loosely coupled reactors compared to 220 MWe, vanadium self power neutron detectors (SPNDs) are kept at suitable locations in the core to construct detailed neutron flux distribution. For reactor regulation and protection prompt acting cobalt/inconel SPNDs are used. In the 700 MWe reactors Regional Overpower Protection System (ROPS) is incorporated.

The power of the Indian PHWR was increased from 220 MWe to 540 MWe by increasing the size of the core and the total inventory of fissile material in the core. This is accomplished by an increased coolant channel diameter for holding larger diameter fuel bundles consisting of 37 fuel pins instead of 19, by putting 12 bundles in place of 10 in a coolant channel and by increasing the number of coolant channels to 392 from 306, the latter numbers corresponding to 220 MWe reactors. The increment in the power capacity from 540 MWe to 700 MWe is accomplished by allowing partial boiling of the coolant near exit end of the channel. Fig. 2 (a) and (b) show the 19 fuel pin bundles used in 220 MWe PHWRs and the 37 fuel pin bundles used in 540/700 MWe PHWRs respectively.

The fuel handling system underwent a major design modification (Muktibodh, 2011) in order to qualify for operating in a stipulated seismic event. In the RAPS and MAPS reactors the design of fuelling machine employs the concept of carriage supporting the guide columns moving horizontally on rails fixed on the floor. This was changed to the concept of fixed guide column with bridge moving vertically up and down in NAPS and subsequent reactors. In the new design the service area of the fuelling machine is a small fully enclosed area where the activity and heavy water vapour is much lower. The coolant channels and fuelling machine are shown in Fig. 3(a) and (b) respectively.

The containment is the last barrier to prevent the spread of radioactive material to external environment in case of a severe accident. The containment structure (Chatterjee et al., 1994; Kakodkar, 1989; Kakodkar and Grover, 2004) has undergone significant changes in the Indian PHWRs starting from RAPS reactors to 700 MWe PHWRs. These changes are (i) step-wise transition from the single to the double containment, (ii) introduction of containment spray system replacing dousing water tank and later

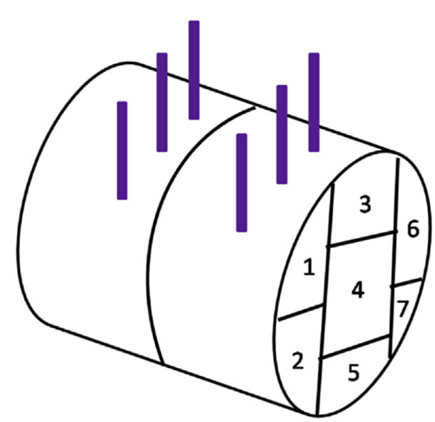
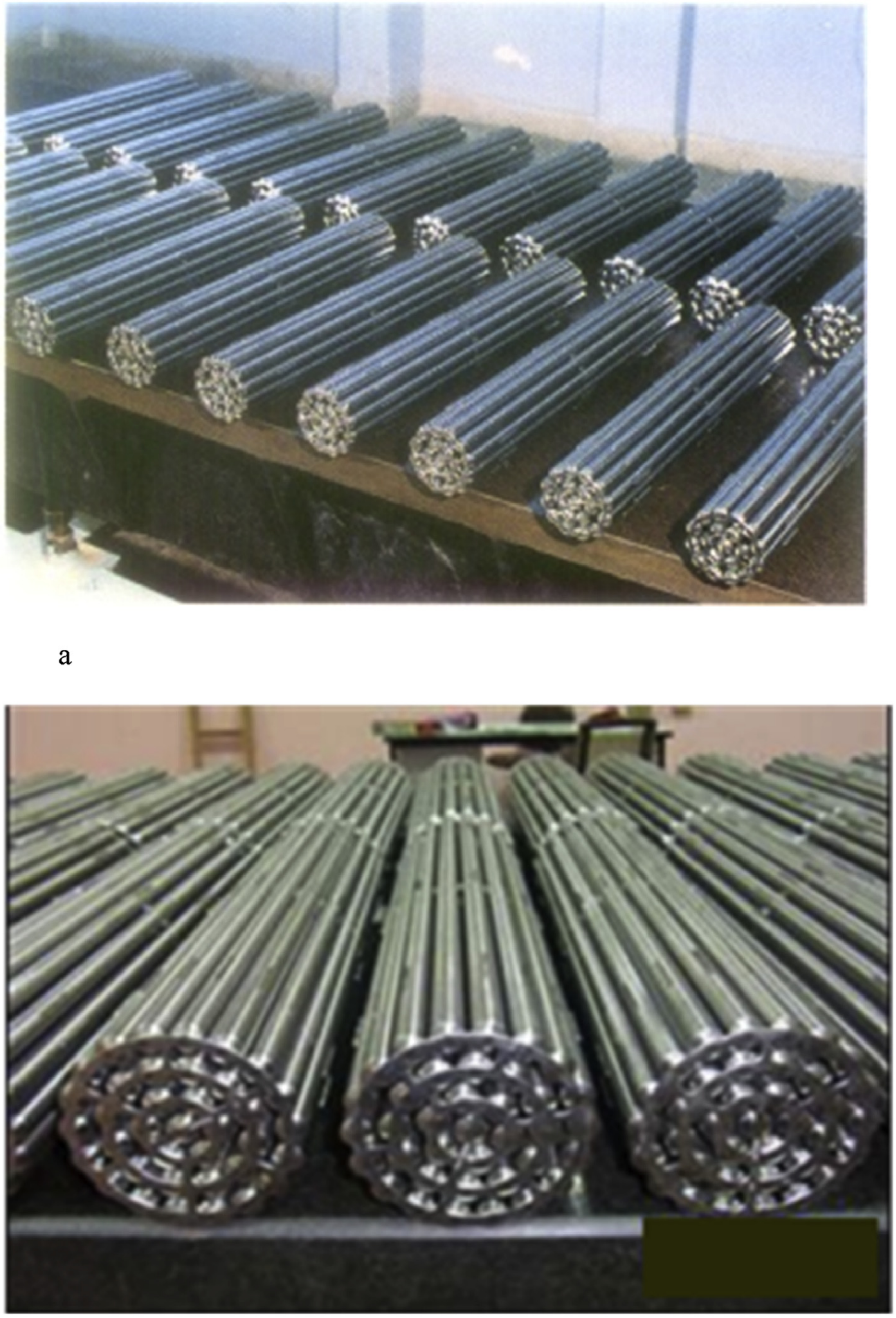



**Fig. 1.** Zone control unit of 540 MWe PHWR dividing reactor into 14 zones.



a

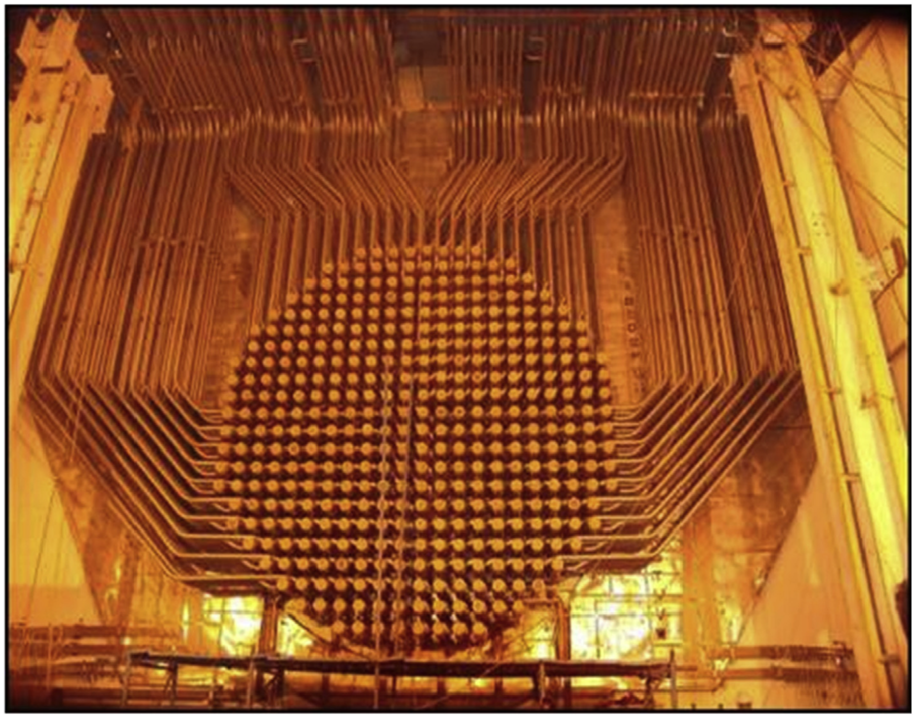
b

**Fig. 2.** (a, b) 19 & 37 fuel pin bundles (Courtesy Nuclear Fuel Complex, Hyderabad, India).

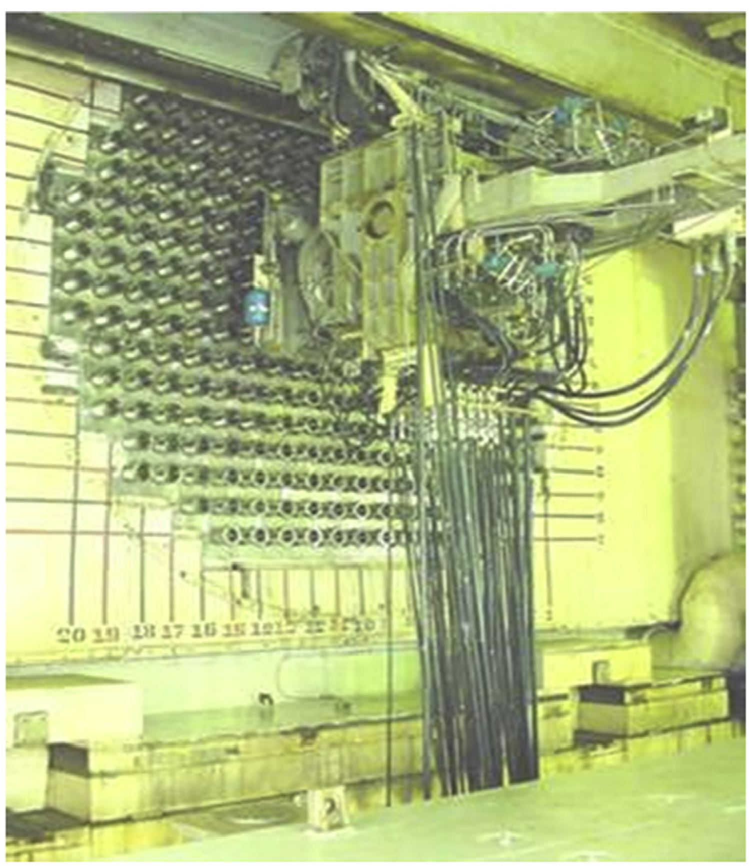
vapour suppression pool (iii) provision of four openings in the dome for installation and replacement of steam generator and (iv) provision of steel liners on the wall of inner containment.

In 540 MWe, two loops concept is used so that in case of a Loss Of Coolant Accident (LOCA) only half of the core is voided while in 700 MWe PHWRs interleaving concept is used to reduce positive reactivity inserted during LOCA. The Passive Decay Heat Removal System (PDHRS) is introduced in 700 MWe PHWRs to remove the decay heat during station blackout condition.

The design changes implemented in the construction of PHWRs in India have been possible due to a series of developments in the computational capabilities in reactor physics, thermal hydraulics, both probabilistic and deterministic safety analysis and stress analysis of reactor components. The indigenously developed computer codes, which were validated against the international benchmarks and some experimental data collected from power reactors, were extensively utilised in the PHWR technology development. For example, the development of three dimensional space-time kinetics code (Jain and Gupta, 1986; Fernando, 2012) which takes care of the neutronic decoupling of large – size reactors has been deployed for safety related neutronic calculations. The computer codes developed for integral neutron transport and



a




b

**Fig. 3.** (a, b) Coolant channels of 540 MWe PHWR and Fuelling machine of PHWR (Courtesy Nuclear Power Corporation of India Limited (NPCIL, India)).

multi-group multi-dimensional diffusion (Rastogi, 1989; Krishnani and Srinivasan, 1981; Degweker, 1985; Srinivasan, 1996) helped in designing optimum positions of various reactivity devices for control and shutdown systems and fuel management.

The materials development programme has played a key role in the successful deployment of PHWRs in India. The whole gamut of materials including the uranium dioxide fuel, zircaloy-2 cladding material, Zr- 2.5% Nb pressure tube material (Singh et al., 2000), Zr- 2.5% Nb- 1% Cu garter spring material, 403 martensitic stainless steel for end-fitting and low alloy steels for the primary heat transport system were indigenously developed. Some of the challenges encountered are control of grain size, porosity and non-stoichiometry of uranium dioxide pellets, fabrication of Zr- 2.5% Nb pressure tubes, fabrication of seamless calandria tubes and development of tools and methodologies for life management of high temperature − high pressure components.

Specifications of uranium dioxide fuel pellets are quite stringent. The grain size, porosity and non-stoichiometry have strong influence on thermal conductivity and fission gas retention capacity. Extensive experimental programme was required for determining the optimum sintering conditions to achieve the right combinations of these properties. The techniques of coating of inner surface of cladding tubes with a very fine graphite layer helped reducing fuel−clad interactions drastically. The first few PHWRs were initially fitted with zircaloy-2 pressure tubes which exhibited a somewhat short operating life (~8 full power years) on account of enhanced corrosion and hydriding. The introduction of Zr- 2.5% Nb pressure tubes has ensured that the irradiation creep and hydrogen intake rates of this material are sufficiently low to extend the operating life of this component quite significantly. However the control of microstructure (Srivastava et al., 1995; Banerjee and Mukhopadhaya, 2007), crystallographic texture and concentration of trace impurities is extremely crucial for achieving an extended life of this pressure boundary material.

Seamless calandria tubes, a component made up of zircaloy-2 were manufactured for the first time in India. In-reactor performance of these tubes has been found to be much superior to the welded tubes which were used earlier. When the concept of tight fitting garter spring was introduced, the question of using Zr- 2.5% Nb- 1% Cu for this component was raised in view of the possibility of their failure due to hydrogen pick up. Based on systematic experimental and stress analysis work it was concluded that Zr- 2.5% Nb- 1% Cu will meet the requirements of tight fitting garter springs (De et al., 1993), as it has now been proved from service experience.

Many renovation activities were undertaken in Indian PHWRs. Some examples are as follows: Zircaloy-2 coolant tubes were replaced by Zr- 2.5% Nb tubes by carrying out En-masse Coolant Channel Replacement (EMCCR) using indigenous technology in six reactors namely RAPS-2, MAPS-1 & 2, NAPS 1 & 2, and KAPS-1 (Varghese et al., 2011). The carbon steel feeders form an important part of primary Heat Transport System (PHT). The Flow Assisted Corrosion (FAC) leads to thinning of these components at some specific bends. En-masse Feeder Replacement (EMFR) is also carried out (Krishna Kumar et al., 2009) in some of the Indian PHWRs where the feeders were replaced by a higher Cr-alloy steel more resistant to FAC.

Many repair work were carried out successfully by developing appropriate tools. The installation of moderator spargers in MAPS is an important example. Inlet and outlet manifolds are provided in RAPS-1 & 2 and MAPS-1 & 2. The inlet manifolds provide the entry path of cold moderator heavy water into the calandria vessel and the hot moderator comes out through outlet manifolds to the moderator heat exchangers for cooling. A failure was observed in the inlet manifolds in MAPS reactors. Interim measures were resorted to for fixing this problem but these resulted in a reduction in the reactor power level from 220 MWe to 170 MWe. Finally a very innovative solution was found after detailed experimental and analytical studies and full scale mock-ups. The problem was sorted out by installing spargers in three coolant channels positions. Cold heavy water is supplied to the calandria vessel through the openings in these spargers. After carrying out this complicated operation successfully the reactor was brought back to the original 220 MWe power (Kakodkar et al., 1998; Venkat Raj, 1999; Jain, 2010).

More detailed account of the technology development and the design of control and shutdown systems are included in a recent review paper (Banerjee and Gupta, 2017).

### 3.1. Fabrication of fuel bundles and production of heavy water for PHWRs

Nuclear fuel complex (NFC), established in 1971 in Hyderabad, India, caters to the production of fuel and structural materials (Ganguli, 2002) required for all the nuclear power plants in India. Apart from making natural uranium dioxide fuel bundles for PHWRs, NFC produces enriched uranium dioxide fuel for the BWRs in Tarapur. Uranium Corporation of India Limited (UCIL) is responsible to mining, milling and beneficiation of uranium ore and supply the processed yellow cake to NFC. Till recently the uranium mines were all located in and around Jaduguda in Jharkhand state. The recent addition of the Tummalapalle mine and mill has enhanced the supply of indigenous uranium and is expected to make a very significant impact on indigenous uranium supply in future. The reasonably assured reserve of uranium in the country stands at the level of about 230,000 tonnes (Department of Atomic Energy (DAE) India, 2016) of uranium oxide ($U_3O_8$) and further efforts of finding additional reserves are being pursued. Pressure tubes, calandria tubes (Saibaba et al., 2002) and all other structural components used in PHWRs are also fabricated at NFC. There is one detailed article in this issue of PNE on fuel fabrication in India (Setty et al., 2017).

Research on the production of heavy water started in BARC in 60's using $H_2S - H_2O$ exchange process in a pilot plant. The first plant to produce heavy water was commissioned in 1962 in Nangal, which was later dismantled. Heavy Water Board (HWB), responsible for the production of heavy water in India currently operates six heavy water plants at Kota, Manuguru, Baroda, Hazira, Thal and Tuticorin. While the first two plants use the $H_2S - H_2O$ exchange process the remaining ones deploy ammonia - hydrogen exchange process. HWB has successfully reduced the specific energy consumption for heavy water production (Kamath, 2001) as illustrated in Fig. 4. During the year 2015-16 the overall specific energy consumption for heavy water was 28.6 GJ/kg (Department of Atomic Energy (DAE) India, 2016). Today India is world's largest producer of heavy water and is in a position to export heavy water. The present capacity is adequate for meeting even the future requirements arising out of installation of new PHWRs.

### 3.2. Performance of Indian PHWRS

The capacity and availability factors of Indian PHWRs from 2000 − 2015 are more than 80% except few years when there was shortage of natural uranium supply. Uninterrupted operation of RAPS-5 for 765 days and continuous run exceeding one year for many reactors are testimonies for smooth operation of PHWRs in India. Apart from good operating practices, the credit for this impressive operating experience can be given to the life management programme of reactor components particularly coolant channels. Periodic In-Service Inspection (ISI) of coolant channels is performed by indigenously developed tools. BARC Channel

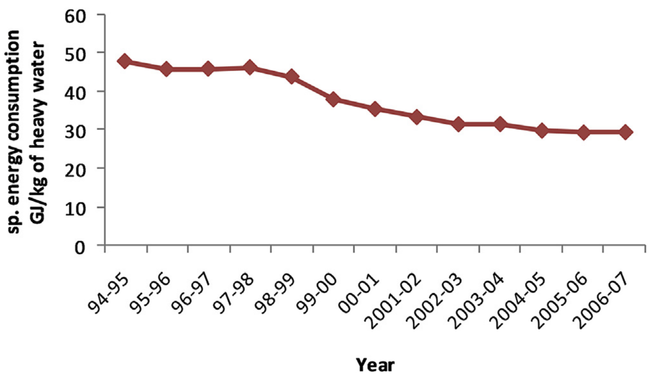



**Fig. 4.** Specific energy of heavy water in different calendar year (Courtesy Department of Atomic Energy (DAE), India).

Inspection System (BARCIS) (Puri and Singh, 2009) has the following non-destructive testing capabilities:

- Ultrasonic detection of flaws in circumferential and longitudinal directions
- Eddy current detection of garter spring location and tilt
- Eddy current estimation of the annular gap between a pressure tube and the corresponding calandria tube
- Eddy current detection of flaws in circumferential and longitudinal directions in pressure tube surfaces
- Inclimation based sag measurement of pressure tubes

BARCIS is used to monitor diametrical creep, garter spring location and presence of defects if any etc.

During operation pressure tubes undergo the in–reactor creep process which may lead to a contact between a pressure tube and the corresponding calandria tube and thereby creation of cold spots in pressure tubes. Hydrogen content in pressure tubes increases with operating life (about 2 ppm per year). There is a tendency for hydrogen to migrate towards localized cold spots. Eventually the hydrogen level in such localized areas exceeds the limit of solubility resulting in hydride precipitation and at a later stage in the formation of hydride blisters. These blisters being brittle in nature can act as sites for crack nucleation. A precise knowledge of the instance of a contact between the pressure tube and calandria tube is, therefore, extremely valuable for predicting the residual life and judging the fitness of service of pressure tubes. A non-intrusive vibration analysis system which can identify the contacting channels has been developed and is routinely, deployed (Moorthy et al., 1995).

The fitness for service for pressure tubes is regularly assessed using both ISI data and theoretical analyses of degradation processes. In order to estimate the hydrogen concentration in pressure tubes a scraping tool is developed which is used for scraping sliver samples from pressure tubes (Rupani and Sinha, 1997).

The economic performance of PHWRs can be assessed on the basis of capital investment per MWe installation or unit energy cost. The capital cost per MWe installation for 700 MWe PHWRs is considerably lower than that for 1000 MWe PWRs if they are constructed at the same point of time and in the same location (India Energy, 2016). There are several elements that enter the costing and therefore, a direct comparison is difficult. However one can see that cost wise PHWRs can even beat the economy of scale.

### 3.3. Irradiation of different fuels in PHWRs

The design of small fuel bundles stacked in horizontal channels permit on power fuelling with the help of fuelling machines. It is because of this design that replacement of one type of fuel bundle by another can be done fairly easily. This feature makes PHWRs suitable for adopting different types of fuel such as U- Pu MOX, U - Th MOX, thorium dioxide, depleted uranium dioxide, and slightly enriched uranium dioxide (Soni et al., 2005; Prasad et al., 2010). Several bundles of some of these fuels have been irradiated in Indian PHWRs for experimental purposes which have generated valuable data on irradiation behavior of these bundles (Bhardwaj and Das, 1986; Bhardwaj et al., 2003).

#### 3.3.1. Use of Slightly Enriched Uranium (SEU)

PHWRs have low burn up due to use of natural uranium. This leads to management of large quantities of initial and irradiated fuel. Use of SEU can enhance burn-up quite significantly (Boczar et al., 2002b; Gupta et al., 2008). Some slightly enriched fuel bundles were irradiated in one of Indian PHWR. The 19 fuel pin bundles with collapsible cladding withstood the burn up of 25,000 MW d/t without any fuel failure. Of course this needed some changes in the bundle design to accommodate additional fission gas release and increased fuel-clad interactions (Prasad et al., 2010).

#### 3.3.2. Use of depleted uranium

While India faced the shortage of natural uranium in the first decade of twenty first century, the concept of using depleted uranium was implemented in several PHWRs which operated at reduced power level. Normally, PHWRs using natural uranium have flattened power distribution. In order to minimize the natural uranium requirement some depleted uranium (0.3 wt% $U^{235}$) fuel bundles were used in its place. Under this condition, the reactor power was reduced, and a peaking in power distribution was established. The reduced neutron leakage and reduced neutron loss through neutron absorber rods helped in gaining some reactivity and improving the burn up. This however was an interim step which was discontinued once the fuel supply was restored.

### 3.4. Role of PHWRs in using thorium

High abundance in nature, high melting point and thermal conductivity, low fission gas release, higher neutron yield per fission over a wide range of neutron energy (in the thermal and epithermal range) of $U^{233}$, reduced production of long-lived TRans-Uranic (TRU) elements resulting from fuel burn-up and reduced fuel deterioration in the event of clad failure are some of the attractive features of thorium dioxide fuel. However, the utilisation of thorium has some associated problems that need to be addressed before large-scale deployment of thorium is considered (Sivasubramanian et al., 2000). Unlike natural uranium, thorium does not have any fissile nuclei and hence needs external supply of fissile material. Thorium will invariably require larger fissile inventory for criticality in comparison to uranium due to its higher neutron capture cross section compared to $U^{238}$. The thorium-based fuel cycle is associated with the generation of $U^{232}$ (half life ~ 68.9 years), daughter products of which have short half-lives. Two of these, $Bi^{212}$ (half-life ~ 60.5 min) and $Tl^{208}$ (half-life ~ 3.05 min), emit strong gamma rays. Therefore, fabrication and handling of $U^{233}$ (with trace of $U^{232}$) fuel would require $\gamma$ -shielding and remote processing (Boczar et al., 2002a). In principle, extracting energy from thorium is possible in present generation PHWRs (Banerjee et al., 2016). In the Indian context, thorium introduction in PHWR has so far been limited for the purpose of initial flux flattening and for generating data on irradiation behaviour of thoria (Balakrishnan, 1994; Balakrishnan et al., 2002).

### 3.5. Advanced heavy water reactor (AHWR)

AHWR is designed as a 300 MWe (Balakrishnan, and Kakodkar, 1992, Kakodkar, 1998; Sinha and Kakodkar, 2006) vertical pressure tube type, boiling light water cooled and heavy water moderated reactor which can use different types of fuels such as $U^{233}$-Th MOX, Pu–Th MOX, and Low Enriched Uranium (LEU) - Th MOX (Umasankari Kannan and Krishnani, 2013). The physics design of AHWR evolved with the objective of maximising thorium utilisation in the equilibrium core and of achieving a negative void coefficient of reactivity. Along with this the thermal hydraulics design has achieved primary heat removal by natural circulation. Many passive safety features such as decay heat removal by isolation condensers, passive shutdown system and emergency core cooling by water injection from accumulators, a large inventory of water in Gravity Driven Water Pool (GDWP) and containment coolers make this design inherently safe. Analysis of transients following station black-out (Mukesh Kumar et al., 2013) and failure of wired shut-down systems shows that peak clad temperature hardly rises even in the extreme condition of complete station blackout and failure of primary and secondary shutdown systems (Vijayan, 2015; Vijayan et al., 2017; *in this Special Issue*).

### 3.6. Capacity build-up of indian nuclear power in the first stage

The construction, operation and maintenance of nuclear power reactors in India is the responsibility of the state owned Nuclear Power Corporation of India Limited (NPCIL). Today twenty-two nuclear power reactors are operating with total nuclear installed capacity of 6780 MWe. This includes RAPS-1 which is currently on extended shutdown. Five more reactors are under advanced stage of construction which will add 3300 MWe to the nuclear installed capacity shortly. These include four 700 MWe PHWRs (Kakrapar Atomic Power project units 3&4 and Rajasthan Atomic Power project units 7&8) and the 500 MWe PFBR at Kalpakkam being constructed by another state owned corporation, Bharatiya Nabhakiya Vidyut Nigam (BHAVINI).

In addition sixteen 700 MWe PHWRs (Four units at Gorakhpur in Haryana two units at Chutka and four units at Bhimpur in Madhya Pradesh, two units in Kaiga in Karnataka and four units at Mahi Banswara in Rajasthan) are either in planning or early construction stages (Nuclear Power Corporation of India Limited (NPCIL), 2016).

In view of the growing demand of electricity and the concern over the climate change, there is a tremendous growth potential of nuclear energy in India. The per capita electricity consumption in India is around 1075 kWh which is about one third of the world average of around 3000 kWh (Central Electricity Authority (CEA), 2015a). With Indian economy growing at an impressive rate, the expected demand of electricity in next twenty years will be about three to four times that of the present. Undoubtedly a major part of this will be met from burning fossil fuels. However, there is a general consensus that burning of fossil fuel should be reduced as much as practicable by enhancing the contributions of primary energy sources such as solar, wind and nuclear. The combined installed capacity of solar and wind has witnessed a very impressive growth in last few years, the total capacity attaining a value of around 35 GW in 2015 (Central Electricity Authority (CAE), 2015b). The potential for the growth of solar energy in the country is very high and there is an ambitious plan of setting up of 100 GW solar capacity by 2022. In spite of such huge growth in solar and wind energy there will be a large gap between the projected total electricity demand and the supply from all renewable energy sources including solar and wind. Fig. 5 shows the projected growth scenario of electricity demand (Integrated Energy Policy, 2006) in next fifteen years. An upper bound estimate of the growth of electricity generation from renewable sources is made by assuming the achievement of the target of 175 GW in 2022 from solar and wind and a steady extrapolated growth by 170 GW every five year. Fig. 5 shows the gap between the electricity demand and the projected supply from all renewable sources. Nuclear energy can make a sizable contribution in reducing this gap.

The fact that the capacity factor for nuclear power (nearly 80%) is much higher compared to those of solar (~20%) and wind (~25%), the contribution in terms of energy produced per GWe of installation is nearly four times higher for nuclear power. This is reflected in the data of installed capacity and electricity generation with respect to nuclear and other renewable energies (Fig. 6) (Nuclear Power Corporation of India Limited (NPCIL), 2016; Central Electricity Authority (CEA), 2015b). Solar and wind energy being, by nature, a dispersed form of energy and their availability being intermittent, they are not suited for meeting the base load requirement. On the other hand the concentrated and reliable nuclear energy can provide uninterrupted power to meet the base load requirement. The high capacity factor, competitive tariff, and an impeccable safety record of Indian nuclear power stations generate enough confidence in projecting a target of about 10% of the total electricity generation by 2032 (as against 11% of the present global nuclear contribution). As the total annual electricity demand in next fifteen years will exceed 3850 TWh, a target of 400 TWh per year of nuclear electricity after fifteen years appears quite reasonable. With average capacity factor of 80% this would correspond to an installed nuclear capacity of nearly 57 GWe. The projected nuclear power capacity of 63 GWe (Integrated Energy Policy, 2006) is, therefore, well justifiable.

With India mastering the PHWR technology and its associated fuel cycle, the rapid growth in installed generation capacity using PHWRs was technically possible, provided adequate natural resources were available for the programme. With international civil nuclear cooperation, the required uranium is available now from international sources (Grover, 2017; *in this Special Issue*). Accordingly, an augmented programme for growth of PHWR based nuclear power plants is planned. In addition, with the induction of large capacity Light Water Reactors (LWRs) with international cooperation, a faster growth in the installed capacity, in the near term, has also become possible. Thus for an interim period till fast reactors start contributing a major part of nuclear power, there is a plan to install large PWRs (1000–1650 MWe) in clusters in a few energy parks at Kudankulam in Tamil Nadu, Jaitapur in Maharashtra, Kovvada in Andhra Pradesh, Mithi Virdi at Gujrat and Haripur in West Bengal. Each of these parks will have a potential capacity of about 10,000 MWe. At Kudankulam two 1000 MWe Russian PWRs

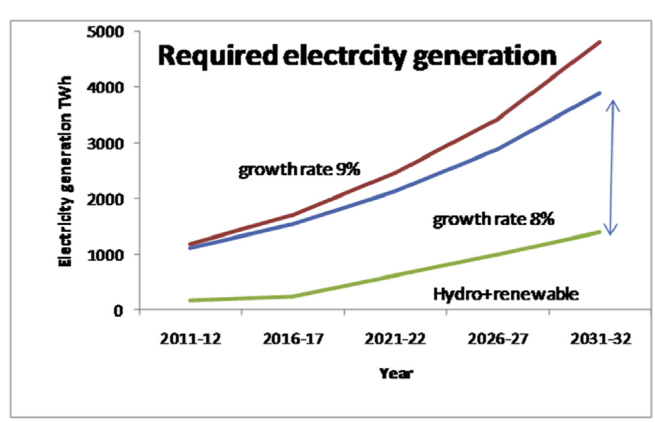



**Fig. 5.** Projected growth demand in next fifteen years.

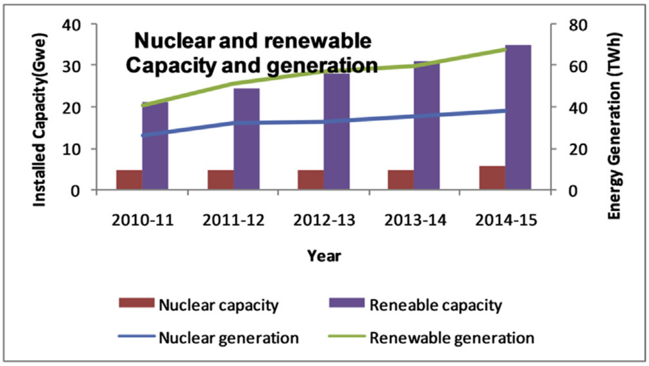



**Fig. 6.** Installed capacity and electricity generation with respect to nuclear and other renewable energies capacity.

are operational and two more are in early stage of construction. For other sites, NPCIL in India is having techno-commercial negotiations with American, French, and Russian companies. In parallel, design and development work for Indian PWR has been initiated. The installed capacity of the first stage of the Indian nuclear programme will, therefore, be much larger than what was originally anticipated based on indigenous uranium resources.

The rapid growth in the installed nuclear power capacity has a special relevance for India as this will also enable a faster growth in the inventory of fissile material which in turn will help in an early entry into the stage of thorium exploitation.

### *3.7. Reprocessing of PHWR spent fuel and radioactive waste management*

The indigenous reprocessing technology has enabled India to embark upon the fast reactor programme. Today plutonium obtained by reprocessing spent fuel from PHWRs is being used for making fuel for FBTR and has been used for making the first charge of fuel for PFBR (Dey and Bansal, 2006). In the once through fuel cycle operation the fuel utilisation remains below 1%. Fast Breeder Reactors (FBRs) with high burn-up and higher tolerance of neutron poisons can increase the fuel utilisation to a great extent. The multiple reprocessing and recycling option can indeed enhance the fuel utilisation by nearly a factor of 60 (World nuclear association, 2016). The introduction of thorium adds further to the supply of fertile nuclides and allows many fold increase in the supply of fissile nuclides by the fertile–fissile conversion.

The adoption of closed fuel cycle has the advantage, of achieving a significant reduction in the radioactive waste burden (Balu and Ramanujam, 1999). The decay of radio-toxicity of the spent fuel with time shows that in once through fuel cycle it takes nearly 200,000 years for the radio-toxicity level to come down to that of natural uranium ore. Once plutonium and minor actinides are separated (Wattal, 2017; *in this Special Issue*) and burnt in fast reactors the radio-toxicity burden drops down sharply (Amit Rajora et al., 2015). India is one of the very few countries that have developed spent fuel reprocessing in an industrial scale. Recently separation of Minor Actinides (MA) has also been implemented in a pilot plant scale (Wattal, 2017; *in this Special Issue*).

Highly radioactive spent fuel bundles are remotely handled by fuelling machine and are placed in spent fuel bay to allow for radioactive decay. After sufficient cooling, the bundles are shifted to reprocessing plants for the separation of uranium and plutonium (Natrajan, 2017, *in this Special Issue*). Rest of the long lived radioactive materials are vitrified and stored in an interim storage (Raj et al., 2006; Wattal, 2013). The total volume of high level waste accumulated so far is rather small due to a small installed nuclear power capacity and the operation of the closed fuel cycle. The safe storage of high level radio-active waste is, therefore, not of a serious concern at present.

## 4. India's second stage nuclear power programme

### *4.1. Fast Breeder Test Reactor (FBTR)*

India entered the fast breeder reactor programme with the construction of FBTR (Paranjape and Sundaram, 1985; Ramalingam et al., 1999), of power output of 40 MWt, based on the French design of RAPSODIE reactor. The first criticality of this liquid sodium cooled FBTR was achieved in 1985. A decade later modules of steam generator (SG) and turbo generator (TG) were added so that FBTR can generate 13.2 MWe. Originally it was planned to have 85% enriched uranium (70 wt %) with plutonium (30 wt %) mixed oxide (MOX) as the fuel. Due to the unavailability of enriched uranium, mixed carbide (MC) of plutonium (70 wt%) and uranium (30 wt%) was chosen as the fuel (Ganguli et al., 1986). The biggest challenge in using mixed carbide fuel in stainless steel cladding was to ensure that no carbon transfer from fuel to cladding occurs during operation. Also the formation of low melting eutectics should be avoided at the fuel-clad interface. A detailed thermodynamic assessment of the stability of the carbide fuel in stainless steel cladding was carried out and the stoichiometry of the mixed carbide fuel is precisely chosen so that neither carbon transfer from fuel to clad nor eutectic formation could occur (Mathews, 1995). The thermodynamic assessment came out to be true as a very high burn up (>150 GW d/t heavy metal) of fuel has been achieved consistently. The reactor started operation (Srinivasan et al., 2006) with a reduced number of fuel subassemblies which was gradually increased. Two types of fuel assemblies Mark-1 and Mark-2 are currently being used. While Mark-1 corresponds to the composition mentioned earlier Mark-2 fuel contains plutonium (55 wt %) and uranium (45 wt %) mixed carbide. FBTR uses six control rods made of boron carbide ($B_4C$) with 90% enrichment of $B^{10}$. It's a loop type reactor. While the Mark-1 fuel has seen maximum burn-up of 165 GW d/t of heavy metal, the Mark-2 fuel was discharged from core after it reached 100 GW d/t burn-up. Fuel performance was excellent as evident from the post irradiation examinations of spent fuels. All sodium pumps, which are very important components, have cumulatively operated for more than 800,000 h. FBTR has been extensively used for testing MOX fuel pins to be used in PFBR and thoria pins for production of $U^{233.}$

CORAL is the facility for reprocessing spent fuel from fast reactors (Natrajan and Raj, 2015), which has been a challenge due to the difficulties in dissolving carbide fuels. Plutonium recovered from the spent fuel is recycled as fresh fuel in FBTR enabling India to close the fast reactor fuel cycle. A new integrated facility is under construction which will reprocess the spent oxide fuel from fast breeder power reactors and fabricate fresh fuel to be recycled in fast reactors. The operation of FBTR over thirty years has generated considerable experience in the fast reactor technology specially that of handling liquid sodium in large quantity.

FBTR helped in validation of many theoretical analyses such as thermal hydraulic, neutronic, stress analysis and qualification of many critical components. The microstructure of the FBTR fuel after Post Irradiation Examination (PIE) (Arun kumar, 2013) to a level of 100 GW d/t burn-up is shown in Fig. 7.

### *4.2. Prototype Fast Breeder Reactor (PFBR)*

Based on the experience of FBTR, the design of a 1250 MWt (500 MWe (gross)) PFBR was initiated (Baldev Raj, 2006; Chetal et al.,

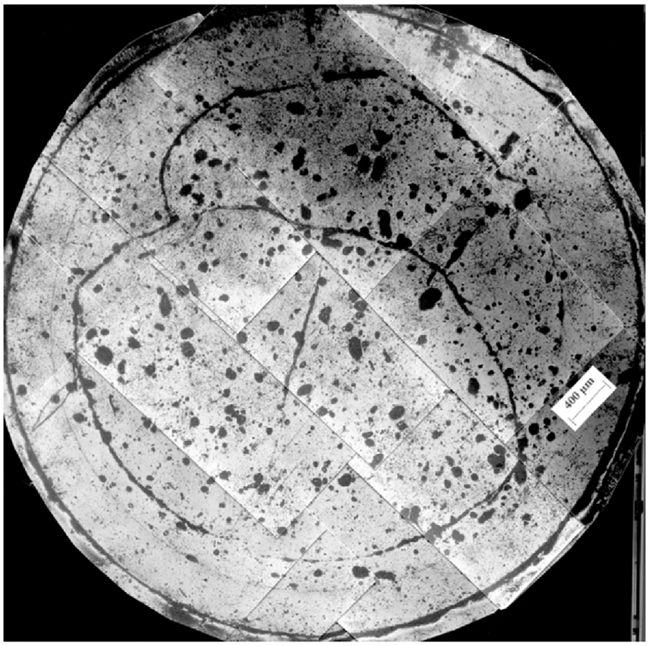



**Fig. 7.** Post Irradiation Examination (PIE) of FBTR fuel after 100 GW d/t (Curtsey IGCAR, India).

2006). The core of PFBR consists of two zones. The inner zone has plutonium (~21 wt %) and depleted uranium - MOX fuel while the outer zone has plutonium (~28 wt %) and depleted uranium -MOX fuel. It has two independent shut down systems, namely, Control and Safety Rods (CSR) and Diverse Safety Rods (DSR). These rods consist of boron carbide ($B_4C$) with $B^{10}$ enrichment of 65%. Axial and radial blankets are made of depleted uranium dioxide. The schematic of PFBR is shown in Fig. 8 (Chellapandi, 2013).

Liquid sodium is used as coolant. All reactivity coefficients except sodium void coefficient are negative. The reactor is pool type in which primary sodium circuit provides large thermal inertia which prevents rapid thermal excursion in case of Design Basis Events (DBE). A Safety Grade Decay Heat Removal System (SGDHRS), employing sodium-sodium and sodium-air heat exchangers, is incorporated which will remove the decay heat by means of natural circulation. Monitoring of the sodium flow and temperature is very essential in each of the fuel sub-assembly due to positive void coefficient. Thermocouples are used at the outlet of each fuel sub-assembly to monitor the sodium temperature. The coolant flow from the primary sodium pumps is also monitored. Several features are provided to prevent and detect sodium leaks which can cause fire. The PFBR core layout is shown in Fig. 9. The reactor is now at an advanced stage of commissioning (Puthiyavinayagam et al., 2017; *in this Special Issue*).

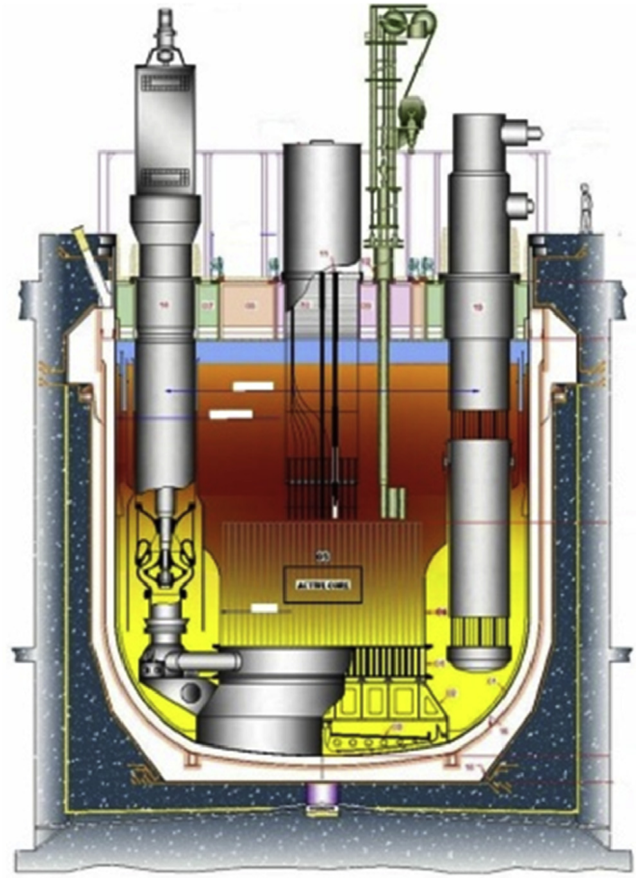


**Fig. 8.** The sketch of PFBR (Courtesy IGCAR, India).

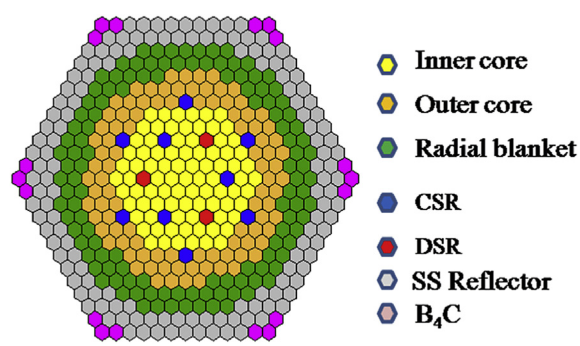



**Fig. 9.** PFBR core layout (Courtesy IGCAR, India).

## 5. LWR - PHWR – fast reactors in tandem

Indian nuclear power programme, as is expected to evolve in the next couple of decades, will consist of three types of reactors, namely, PHWRs, LWRs and FBRs. Originally the first stage of the programme envisaged deployment of indigenous PHWRs and uranium available within the country. With the availability of uranium from international sources, under international civil nuclear cooperation agreements and IAEA safeguards, the present plan is to have a mix of PHWRs and LWRs fuelled by domestic uranium as well as imported uranium.

It is very attractive to deploy a fuel cycle in which discharged fuel from LWR is reprocessed to extract both plutonium and ~1% enriched uranium. The latter can be introduced in PHWR for further utilising the fissile content ($U^{235}$). Plutonium generated from such a fuel cycle (both from PHWRs and LWRs) can fuel fast reactors which in turn can convert fertile ($U^{238}$ and $Th^{232}$) to fissile. India has successfully developed the technology of separation of minor actinides and lanthanides from the spent fuel during reprocessing. This will have a significant impact in reducing the radioactive life of the nuclear waste. Fast reactors will also incinerate the long lived minor actinides and can reduce the burden of radio-toxic waste to a very significant extent. In view of these developments it appears quite likely that the choice of closed fuel cycle that India has made right from the inception of the programme will be able to address the nuclear waste management issue adequately. The fuel cycle

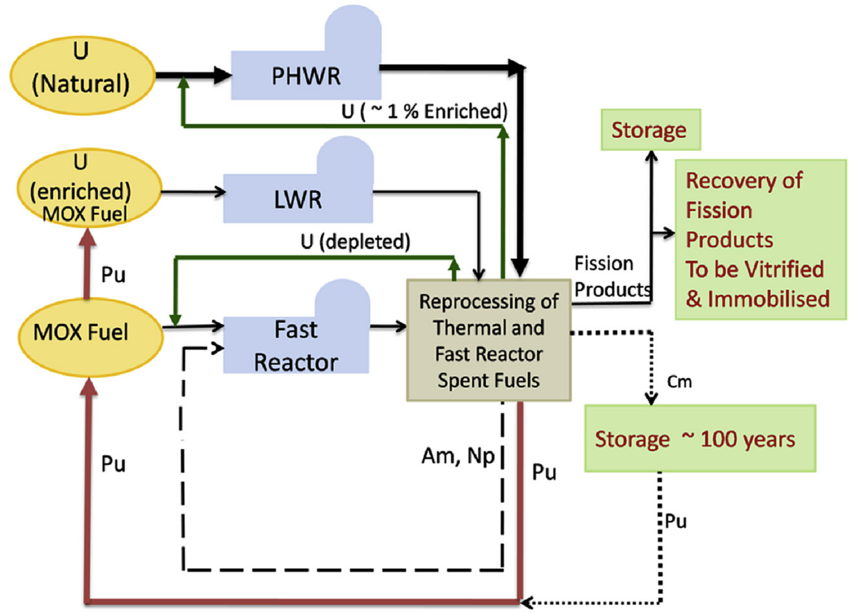



**Fig. 10.** Evolving fuel cycle.

which is gradually evolving is shown in Fig. 10.

## 6. Long term perspectives

Accumulation of sufficient quantity of $U^{233}$, transmuted from $Th^{232}$ in the blanket of fast reactors of the second stage will enable India to begin the third stage to tap energy from nearly inexhaustible source of thorium and provide energy security for the country for several centuries. Putting thorium in the blanket of Pu/$U^{238}$ fuelled fast reactors can produce $U^{233}$ without much affecting the breeding ratio, while introducing thorium both in the core (Pu/Th) and blanket of these reactors significantly reduces the breeding ratio (Ramanna and Lee, 1986). As the use of mixed oxide fuel in FBRs will not permit a shorter doubling time, metallic fuel is being developed which has a promise to reduce the doubling time to about 10–15 years.

The conceptual Molten Salt Breeder Reactor (MSBR) (Briant and Weinberg, 1958; Robertson, 1971) where the breeding ratio achieved was of the order of 1.05 in thermal energy range is an attractive option for extracting energy from thorium. Though the breeding gain of $Pu^{239}$ in fast reactors is higher than that of $U^{233}$ in thermal reactors, the doubling time for the latter can be made comparable to that of the former. This is because thermal reactors require smaller specific fissile inventory (fissile requirement per unit power for starting a new reactor) compared to that of fast reactors. Experimental facilities are set up at BARC to study thermal hydraulic behavior of molten salts used in MSBR and their compatibility with other structural materials (Department of Atomic Energy (DAE) India, 2016; Vijayan et al., 2017; *in this Special Issue*).

The technology of high temperature reactor is under development at BARC (Dulera et al., 2017; *in this Special Issue*). The objective of these reactors is to produce hydrogen through thermo - chemical process along with production of power.

## 7. Accelerator driven subcritical systems (ADSS)

In a sub-critical reactor, external supply of neutrons is needed to maintain a constant reactor power. Non-fissile neutrons generated through a spallation reaction can provide 20 to 40 neutrons per single event when a high-energy proton beam of 1–2 GeV (Kapoor, 2002) collides with a heavy atom nucleus such as lead or uranium.

These systems can be used to burn minor actinides of long half-lives and/or to generate fissile nuclei $U^{233}$ from thorium. Many studies are being carried out to evolve a suitable ADSS design for thorium utilisation. A facility for carrying out experiments on the physics of ADSS is set up at Purnima labs, BARC (Amar Sinha, 2013). The sub critical reactor, with effective multiplication factor ($k_{eff}$) equal to 0.89, uses natural uranium as fuel, polyethylene as moderator and Beryllium Oxide (BeO) as reflector. Supply of neutrons comes from fusion of D-T and D-D reactions.

An interesting concept of one way coupled fast and thermal subcritical reactor with spallation neutron source in centre has been developed by Degweker et al. (1999). A detailed paper on ADSS is in this Special Issue (Degweker et al., 2017; *in this Special Issue*).

As a first step towards the development of ADSS India has been pursuing work on high power linear accelerators.

## 8. Conclusions

The three stage nuclear programme, outlined by Homi Bhabha in the 1950s, still remains attractive for meeting the ever-increasing demand of electricity in India for sustaining its economic growth without adversely affecting the environment and climate. The indigenous development of the entire range of nuclear technologies, particularly, the mastery of the closed fuel cycle and a steady progress in the fast reactor technology has placed the country in a leading position. In order to exploit the large energy stored in abundant thorium in its soil, it is imperative for the country at this stage to expand the programme rapidly for an early entry to the third stage. The recent development of actinide and lanthanide separation in the closed fuel cycle operation offers an excellent prospect of reducing the burden of long lived radioactive waste which can eventually be incinerated in fast reactors and accelerator driven systems.

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## Appendix-A. Different features of Indian PHWRs.

| Site of reactor | Capacity gross (MWe), Year of first approach to criticality | Containment Design | New features added in control and shut down system |
|---|---|---|---|
| Rajasthan Atomic Power Station (RAPS) unit 1 | 100 | Cylindrical wall: Reinforced concrete | **Control System:** Moderator level control, 4 corner Adjuster rods, 2 central adjuster rods |
| | 1972 | Dome: Pre-stressed concrete | **Shut down system:** Moderator Dump |
| Rajasthan Atomic Power Station (RAPS) unit 2 | 200 1980 | Same as RAPS-1 | Same as RAPS-1 |
| Madras Atomic Power Station (MAPS) unit 1 & 2 | 2 × 220 | Inner cylindrical wall and dome: Pre-stressed concrete | Same as RAPS-1 |
| | 1983, 1985 | outer cylindrical wall: Reinforced rubble masonry | |
| Narora Atomic Power Station (NAPS) unit 1 & 2 | 2 × 220 | Primary containment: Pre-stressed concrete, Flat Roof | **Control System:** 4 Absorber rods, 2 Regulating rods 2 Shim rods ALPAS-CAM mode: 1 mk equivalent boron whenever shim rod is inside core |
| | 1989, 1991 | Secondary containment: Cylindrical wall and dome Reinforced concrete | **Shut down system:** 14 mechanical shut off rods 12 Liquid Poison Tubes ALPAS-BAM mode: 8 ppm equivalent boron into moderator |
| Kakrapar Atomic Power Station (KAPS) unit 1 & 2 | 2 × 220 1992, 1995 | Primary containment: Same as NAPS-1 Dome of secondary containment has two openings | Same as NAPS-1 |
| Kiaga Atomic Power Station (KGS) unit 1, 2, 3 & 4 | 4 × 220 | Both primary and secondary containment have 2 openings each | Same as discussed above, changes are as follows: |
| | 2000, 1999, 2007, 2010 | | Shim rods full length rods. ALPAS- BAM and GRAB replaced by LPIS system. |
| Rajasthan Atomic Power Station (RAPS) unit 3, 4,5 & 6 | 4 × 220 1999, 2000, 2009, 2010 | Same as in KGS | Same as in KGS |
| Tarapur Atomic Power Station (TAPS) unit-3 & 4 | 2 × 540 | Primary containment: Pre-stressed concrete | **Control System:** 17 Adjuster rods, 4 Control rods, Zone Controller Units: light water filled in different compartments of 6 tubes which divides reactor into 14 zones, |
| | 2006, 2005 | Secondary containment: Reinforced concrete | **Shut down system:** 28 mechanical shut off rods, 6 horizontal Tubes with perforations which inject Gadolinium nitrate solution into moderator |

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