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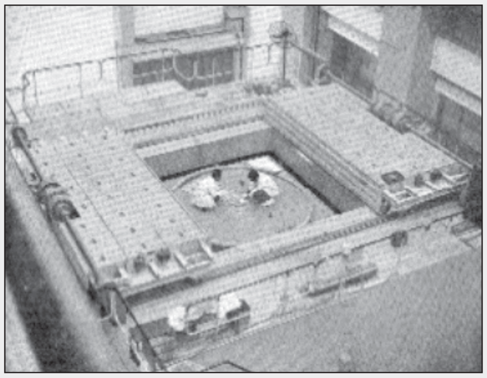
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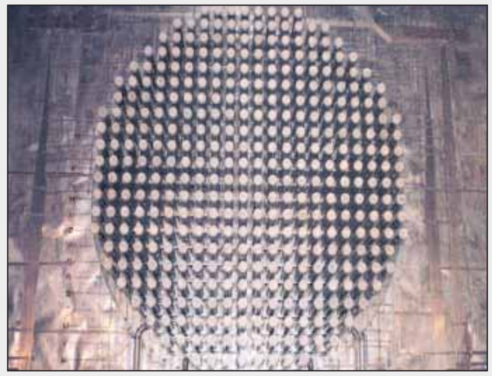
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Chapter 2

Powering the Nation – Research & Power Reactors


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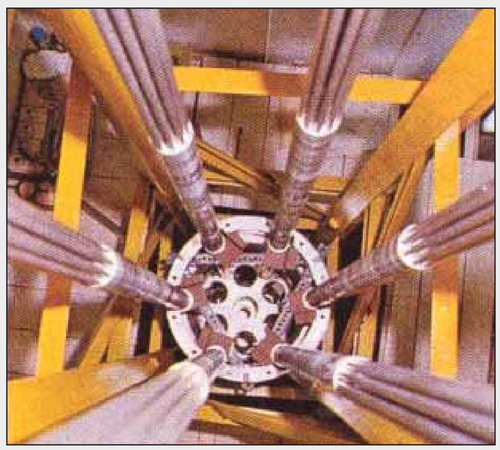
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Captions for Photo-Collage

1. *Panoramic view of the DHRUVA and CIRUS reactors located at BARC, Trombay.*

2. *Neutron beam utilization facility at DHRUVA. These facilities are extensively used for characterization of materials using techniques like neutron scattering, neutron activation analysis and for isotope production.*

3. *Top view of ZERLINA reactor core. The advantage of this reactor is extreme flexibility of reconfiguring core geometries, which is crucial for optimization of reactor design.*

4. *View of the end-fitting and end shield of a Pressurised Heavy Water Reactor. This design has a unique feature of refueling without having to shut down the reactor. This has enabled achieving high plant capacity factors.*

5. *Steam generators for the 540 MWe reactors being constructed at TAPS. These reactors are the next generation high power PHWR's and recently, one of the reactors was made operational. The capacity to manufacture very large components to exacting standards has been pioneered by DAE and has significantly benefited Indian industry.*

6. *Quality assurance - the highest priority of the Department. Dr. M.R. Srinivasan, former Chairman AEC, personally inspecting the turbine blades.*

7. *Top view of control plug of FBTR during out-of-pile tests.*

Research Reactors – The Test Beds of Nuclear Technology

Panoramic view of the DHRUVA and CIRUS reactors located in BARC

On December 2nd 1942, something incredible was happening at the squash courts of the university of Chicago. Enrico Fermi, the Italian born physicist was demonstrating, to the world, the feasibility of extracting controlled energy from self-sustaining neutron chain reaction. Shortly thereafter, as early as in 1944, even before India attained independence, Bhabha was already formulating the strategy for setting up a nuclear research programme in India. Right from the formative years of the programme, Bhabha wanted India to be self-reliant in this newly emerging area.

> *"...The emphasis throughout has been on developing know-how indigenously and on growing people, able to tackle the tasks, which lie ahead..."*
>
> ***- H. J. Bhabha***

In line with his vision, one of the earliest decisions he took was to build research reactors of different types. Each of these rapidly accelerated the learning process necessary for developing an intimate understanding of the complex issues involved in the control of nuclear chain reaction. The design of reactors involves high levels of optimization of geometry, fuel design, safety, materials selection, irradiation behavior of fuel and structural materials. These could be mastered only by building test reactors, which use different types of fuels, structural materials, coolants etc. Towards this end, several research reactors were systematically designed and built during different stages of the programme. The first of these reactors was a swimming pool type of reactor, aptly christened "APSARA" – the celestial water nymph, by Pandit Nehru himself. The basic design for this reactor was frozen in July 1955 and Indian scientists and engineers completed the construction in just over a year. With APSARA, India became the first Asian country outside the erstwhile Soviet Union, to have designed and built its own nuclear reactor .

The next crucial step involved the planning of larger reactors having much higher neutron flux and power than what was available at APSARA. This plan materialized in 1960 with the building of CIRUS, a high power (40 MWt) research reactor. This reactor, then known as the Canada India Reactor or CIR for short, was built in collaboration with Canada. These early gains catapulted India into an ambitious nuclear program. CIRUS and APSARA became centres of excellence in nuclear education. Experiments carried out at APSARA and CIRUS have provided the necessary confidence and expertise for design and safe operation of many nuclear reactors in the country.

In early 1961, a zero energy critical facility named ZERLINA (Zero Energy Reactor for Lattice Investigations and New Assemblies) was built, for studying various geometrical aspects (lattice parameters) of a reactor fuelled with natural uranium and moderated with heavy water.

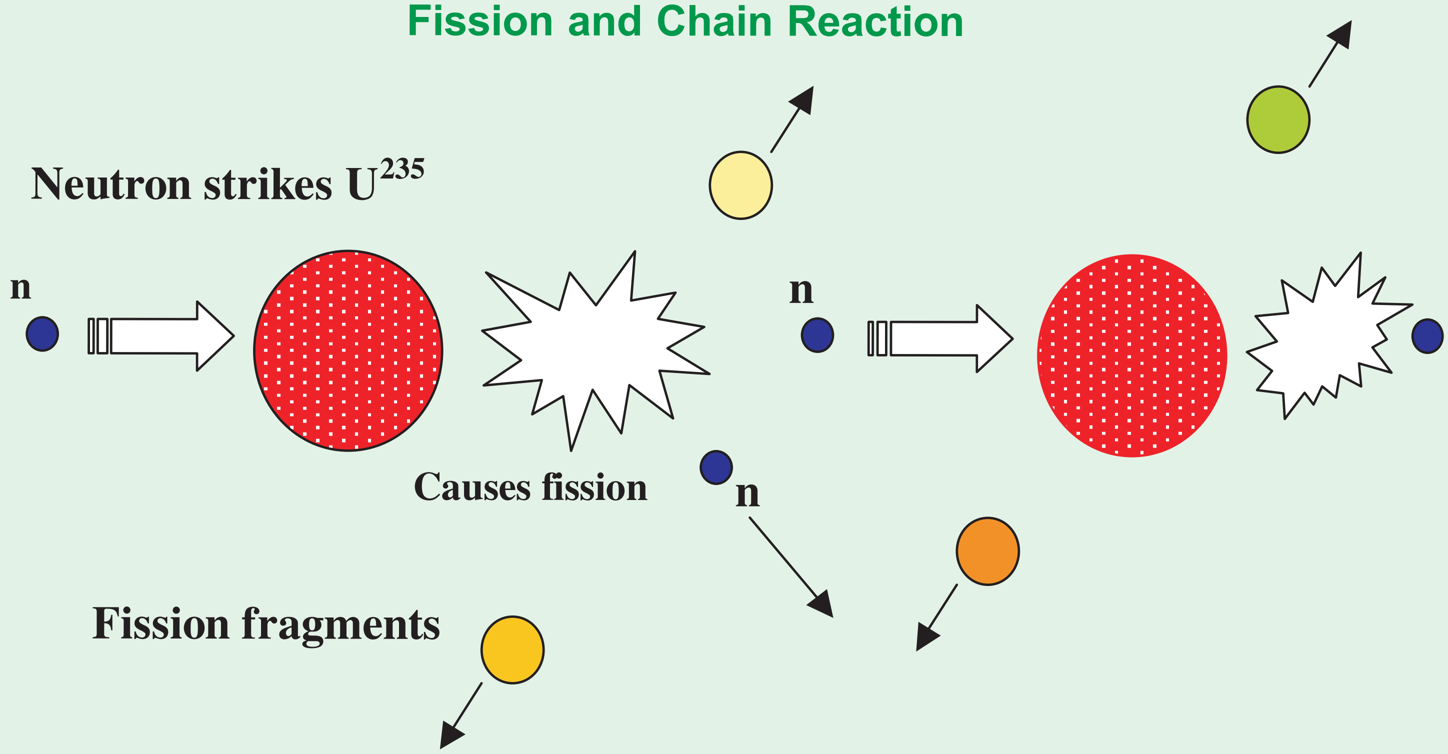
It may be recalled that the three-stage programme calls for building of plutonium based reactors in the second stage. Therefore, the next logical step was to build a critical facility, which used plutonium as fuel. Such a test reactor was built in 1972 and was named PURNIMA (Plutonium Reactor for Neutron Investigations in Multiplying Assemblies). This reactor was intended for studying the behaviour of plutonium fuel in a pulsed fast reactor (PFR). Following this, a critical facility called PURNIMA-2 was designed, with a solution containing 400 gms of uranyl nitrate serving as the fuel for this facility. It attained criticality in 1984.

In the early seventies, a need was felt for a research reactor having even larger neutron flux and irradiation volumes than CIRUS, for meeting the growing requirements for radioisotopes and research. This culminated in building of a totally indigenous 100 MWt research reactor, having the highest flux in Asia at that time. It attained criticality in August 1985 and was named DHRUVA.

Apart from thermal power reactors, it was also realized in the initial stages that adequate energy security could be provided only by taking recourse to fast reactors as second stage power programme. Towards, gaining first hand experience in the fast reactor technology, it was decided to construct a Fast Breeder Test Reactor (FBTR) at Reactor Research Centre, Kalpakkam (later renamed as Indira Gandhi Centre for Atomic Research). FBTR was commissioned in 1985, with indigenous plutonium-uranium mixed carbide fuel, providing valuable design and operational experience. Based on the successful operation of FBTR, broad based R&D and component development , adequate experience and confidence was gained to embark on commercial phase of fast reactor programme in terms of 500 MWe FBR at Kalpakkam in 2003.

As a part of studies with ^{233}U fuel, a 30 kW pool type research reactor, KAMINI (Kalpakkam MINI), was designed and built. Prior to this, a mock up of the core of this reactor, was built and made critical in April 1992. It was given the name PURNIMA-3. KAMINI was made operational in 1996. This reactor is being extensively used as a neutron source for research applications such as neutron radiography of irradiated nuclear fuel and pyro-devices for the Indian Space programme.

Fission and Chain Reaction

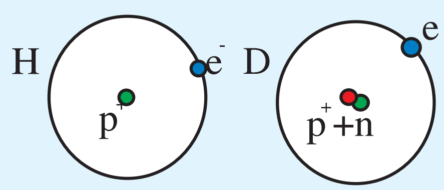


Fission, Chain Reaction and Criticality

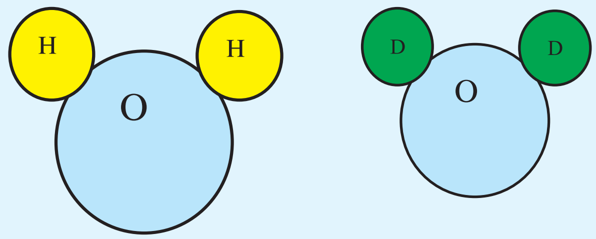
When neutrons strike a **uranium-235** nucleus (red ball in figure) in the **fuel**, sometimes the nucleus splits into two smaller nuclei or undergoes **fission**. These nuclei called **fission fragments** usually have different sizes. The kinetic energy of the fission fragments is transferred to other atoms in the fuel as **heat energy,** which is eventually used to produce steam to drive the turbines. Every fission event results in the release of 2 to 3 energetic neutrons. For every fission event, if at least one of the emitted neutrons, on an average, causes another fission, a **self-sustaining chain reaction** will result. Under such a condition, when the power remains constant in time, or the chain reaction is so controlled that exactly only one neutron from every fission event contributes to the subsequent fission process, the reactor is said to have become **critical**.

Moderator

The chances of occurrence of a fission event are greatly enhanced if the energetic fission neutrons are slowed down to very low energies (thermal energies). This is done when the neutrons lose their energy on being elastically or inelastically scattered by light atoms or molecules. These substances made of such light atoms/ molecules are called moderators - usually **light water** (H_2O) or **heavy water** (D_2O). Graphite is also occasionally used for this purpose.



Schematic Representation of hydrogen and deuterium



Schematic of light water and Heavy water molecules

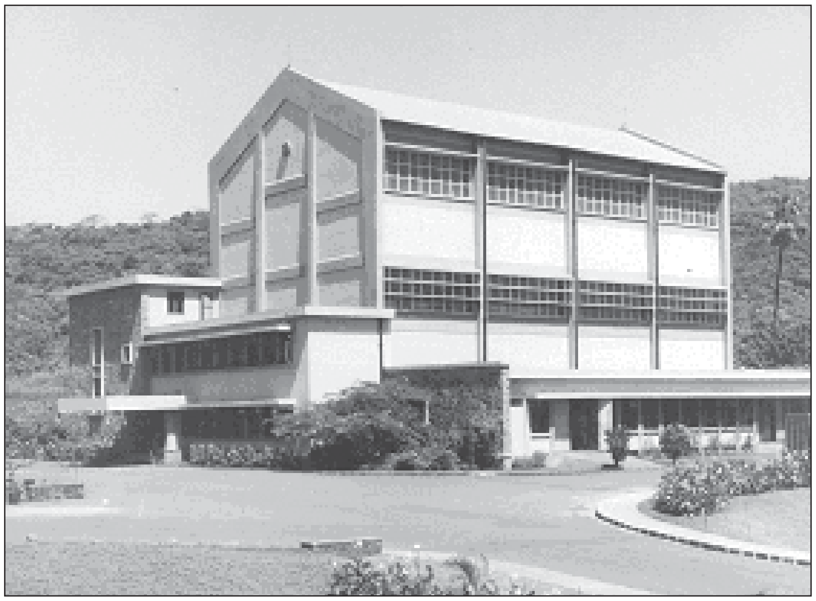
A good moderator is one that rapidly slows down neutrons by taking away their kinetic energy, but does not absorb them. Light water slows down neutrons better than heavy water, but it also absorbs more neutrons. When natural uranium (containing only 0.7% of fissile nuclei with the remaining 99.3% being ^{238}U atoms that can contribute to non-fission absorption) is used as fuel, and light water is used as moderator, not enough neutrons are available to sustain the fission chain reaction. Therefore heavy water is used in PHWR's, which uses natural uranium oxide as fuel. In enriched fuel having more ^{235}U or Pu, light water can, however, be used as moderator.

Breeding and fuel utilization

Judicious utilization of nuclear fuel is an important consideration for the long-term exploitation of nuclear energy. Therefore, an optimum use of both **fissile** and **fertile** isotopes is essential. Fertile isotopes like **^{238}U** and **^{232}Th** are converted into fissile **^{239}Pu** and **^{233}U** in a reactor. In a PHWR, some amount of ^{239}Pu is also formed. However, the quantity of fissile ^{235}U consumed (0.5%) is greater than the quantity of fissile Pu (~0.3%) that is produced. The ratio of fissile isotopes produced in a reactor to that which is consumed is called the **breeding ratio**. When this ratio is greater than one, more fissile material is produced than is consumed. Reactors for which the breeding ratio is more than one are called **breeder reactors**.

Fast Reactors

Most neutrons produced in a fission event are highly energetic with energies in the range of millions of electron volts. In moderated reactors, most of the neutrons causing fission are slow or thermal neutrons and hence these reactors are called **thermal reactors**. If the reactor does not contain materials with low mass number to moderate the neutrons significantly, the majority of fission events are caused by **fast neutrons**. A reactor in which this is the case, is called a **fast reactor**. The fuel in these reactors must however contain significantly more fissile material to sustain the neutron chain reaction, to compensate for the lower probability of fission by faster neutrons. High breeding ratios can be achieved in fast reactors. Since the neutrons in a fast reactor should not be slowed down by moderating materials, high mass number materials like **liquid sodium metal** have to be used as a **coolant** to remove heat energy from the reactor core.



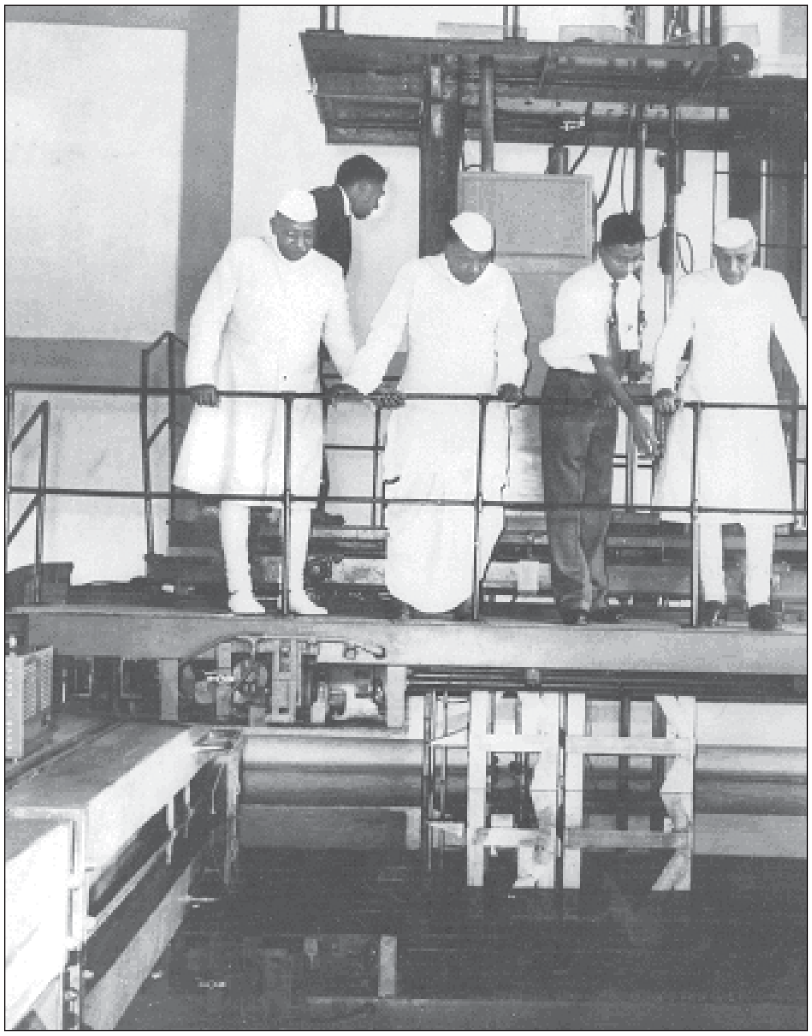
Abode of the APSARA – from dreams to reality

"APSARA" (the celestial water nymph) lovingly christened by Pandit Nehru himself, is a reactor of historical importance to the atomic energy community in India. It was the first ever reactor built in the whole of Asia, outside the erstwhile Soviet Union, at that time. Designing, building and operating this 1 MWt reactor provided very valuable experience and it instilled immense confidence in Indian scientists and engineers.

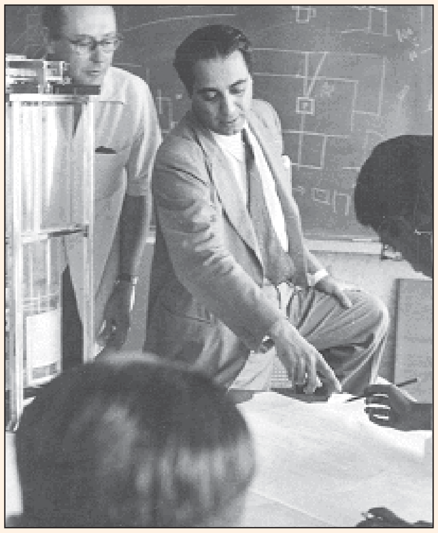
Construction of this reactor was completed in a year's time and the reactor attained its first criticality at 3.46 pm on August 4, 1956. This is a swimming pool type reactor with a simple yet versatile design, which allows for a host of important and useful experiments. It basically consists of the reactor core, which is an assembly of fissile fuel elements made of an alloy of uranium and aluminium, clad in an aluminium sheath. The core is suspended in a large pool of demineralized water, and can be positioned at different locations with an overhead crane. The reactor pool is 8.4 m long, 2.9 m wide and 8 m deep and is made of 2.6 m thick reinforced concrete walls, lined with thick plates of stainless steel. The pool of water in which the core containing the uranium fuel is suspended has multiple functions. It not only slows down neutrons emerging from nuclear fission in uranium but also removes the heat energy produced by the nuclear reaction. The process of slowing down the neutrons (moderation) enables the chain reaction to become self-sustaining. The removal of heat from the fuel by the water (coolant) is essential to prevent the fuel from melting down. Thus, the large tank of water built like a swimming pool acts both as a coolant, as well as a moderator. It also acts as a reflector of neutrons and provides shielding. The average neutron flux available in the core is around 10^{12} n cm^{-2} sec^{-1} at an operating power of 400 kW.

The whole reactor system, including the control system, for APSARA was indigenously designed, fabricated and commissioned.

APSARA has completed more than 49 years of operation. It is still in good condition and expected to remain so for quite some time to come. In his message on completion of 40 years of APSARA, Dr. R. Chidambaram, the then Chairman, AEC



Prime Minister Nehru admiring APSARA – the first Indian nuclear research reactor



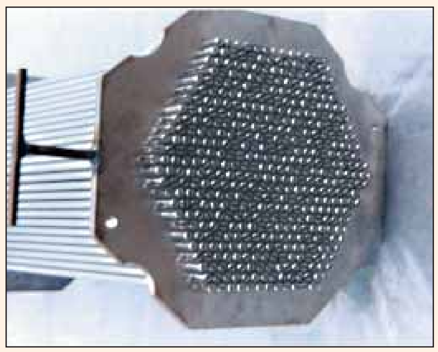
The making of APSARA - Bhabha at the drawing board

Excitement in the control room of APSARA at the first criticality (1956)

had said: *'40 years is a long period. Many good traditions have come to stay at Trombay. It is our duty to take the programme to greater heights on the basis of foundation and traditions established by our pioneers'*.

Utilization of APSARA: In its long innings, APSARA has been instrumental in carrying out advanced studies in the field of neutron physics, fission physics, radiochemistry, biology, irradiation techniques and R&D work on reactor technology. APSARA also laid the foundation for production of short-lived radioisotopes. Since the reactor allowed ease of access, various stable elements could be irradiated with neutrons to produce radioactive isotopes. It, therefore, served not only as the stepping-stone for various nuclear activities, but also marked the beginning of production and application of radioisotopes in the country. These isotopes opened new vistas in the field of medicine, industry and agriculture in India. Neutron activation analysis technique developed with APSARA found wide applications in chemistry, archeology and forensic sciences. One of the unique facilities available at APSARA is the shielding corner. The process of fission produces highly energetic particles and gamma radiation. In order to minimize the presence of these in the external environment, specialized shielding material has to be provided. The shielding corner of APSARA has been extremely useful in ratifying the design adequacy of many of the later reactors like the DHRUVA, heavy water power reactors and the more recent prototype fast reactor design. A number of experiments were carried out for studies on radiation streaming through penetrations and ducts of various shapes for the proposed 300 MWe Advanced Heavy Water Reactor (AHWR). Neutron Radiography (NR) facility was established in APSARA in the early seventies. This facility has been used successfully for confirming that all components in a device used in space research applications have been fully assembled in the required configuration before the device is lifted into a space vehicle.

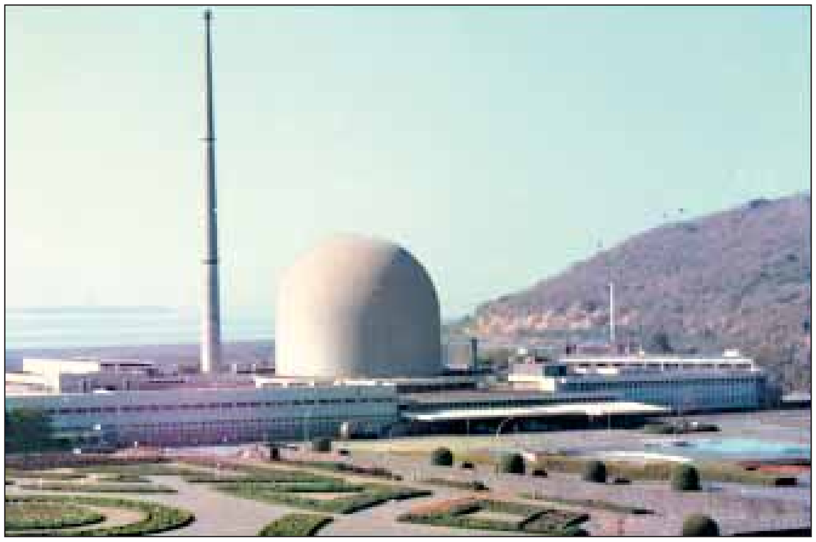
As APSARA is ageing, it is planned to carry out extensive refurbishment of the reactor to extend its life and to upgrade its safety features to be in line with the current safety standards. As part of the refurbishment, it is proposed to modify the basic design to incorporate certain salient design features of the proposed 20 MWt Multi Purpose Research Reactor (MPRR). The modified APSARA would not only act as a demonstration core for MPRR but also offer enhanced facilities for isotope production, material irradiation, neutron radiography and condensed matter research.



Model of lower axial plenum region of FBR core subassembly used for shielding experiments in APSARA to estimate the neutron damage to the grid plate.

Configuration of FBR radial shield model with stainless steel and sodium blocks. At extreme left are depleted uranium assemblies used to generate fast neutron spectrum in APSARA

CIRUS - The Workhorse



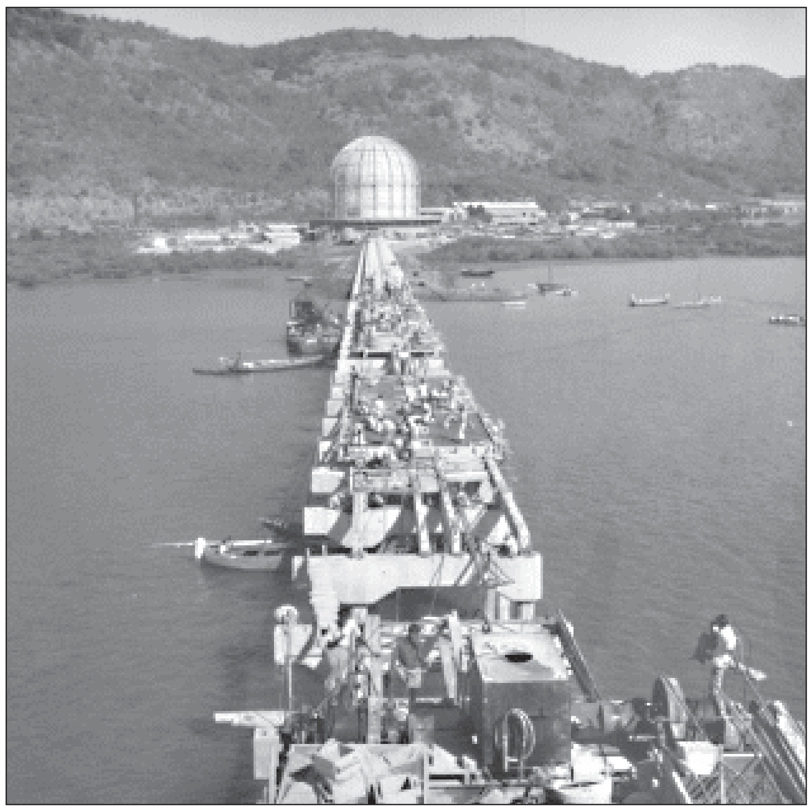
CIRUS - Symbol of modern India

CIRUS was built at Trombay in the late fifties with the assistance of Canada. The basic design of CIRUS was based on a Canadian NRX reactor, which was located in a relatively sparsely populated area. NRX had the advantage of Ottawa river nearby, which provided continuous supply of fresh water for once-through cooling. On the other hand, with limited availability of fresh water at Trombay, a closed loop system consisting of primary cooling using demineralised light water and secondary cooling based on sea water was adopted for reactor cooling. Significant modification in the form of a domed steel shell was also provided for CIRUS, to ensure public safety.

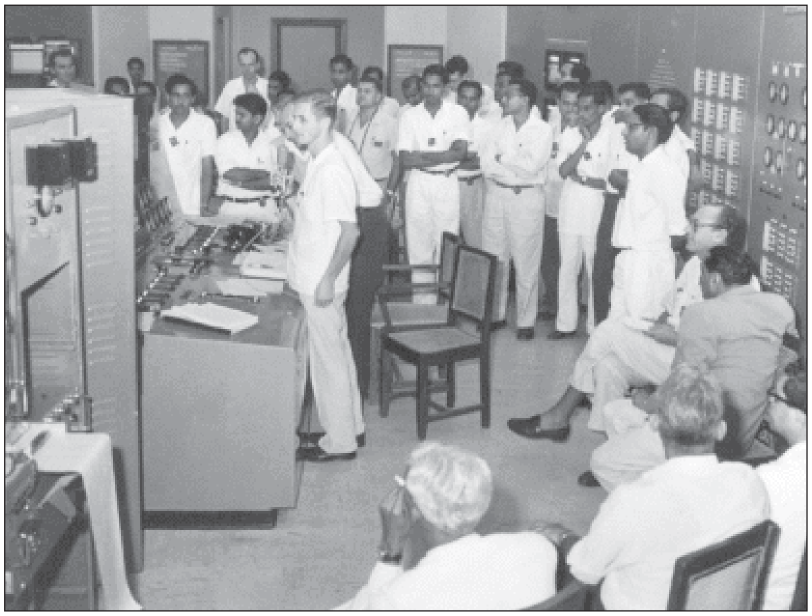
While planning CIRUS, Bhabha decided that unlike APSARA, CIRUS should have Indian made fuel elements. The first fuel element was successfully fabricated in June 1959 at Atomic Fuels Division (AFD), BARC . Two of the Indian made fuel rods were tested in NRX reactor. NRX studies had shown that these two rods fabricated and supplied for irradiation were the best rods they ever had in their reactor, thereby instilling the confidence for indigenous fuel fabrication. For the first charge of CIRUS, half the numbers of fuel rods were fabricated at AFD and the remaining were obtained from Canada. CIRUS attained the first criticality on July 10, 1960.

CIRUS is a vertical tank type reactor of 40 MWt capacity and at that time, its neutron flux was among the highest in any reactor in the world. The reactor is fuelled with metallic natural uranium, moderated with heavy water and cooled by demineralized light water. The reactor core is housed in a calandria, a cylindrical aluminium vessel with aluminium lattice tubes located between top and bottom tube sheets. Two annular rings of graphite reflector, cast iron thermal shield and a heavy concrete biological shield surround the reactor vessel. On top and at the bottom of reactor vessel, there are water cooled aluminium and steel-thermal-shields.

Fuel assemblies are located inside the lattice tubes and are cooled by forced recirculation of de-mineralized light water in a closed loop with the coolant flowing from top to bottom. The choice of light water as coolant also enabled incorporation of passive safety features such as a high head water storage tank called ball tank (based on its shape) to provide coolant under gravity in case of non-availability of re-circulation pump. On stoppage of coolant recirculation pumps, reactor is automatically shutdown and cooling is established by one-pass gravity assisted flow of water from ball tank. Coolant outlet from the core is led to a under ground concrete tank (dump tank) from where water is pumped back to the ball tank using pumps



Construction activities at CIRUS – sea water intake bridge for cooling purpose



Anxious midwives! Scientists and engineers in the control room during first criticality of CIRUS

Prime Minister Lal Bahadur Shastri – an enthusiatic proponent of science and technology visits CIRUS accompanied by Homi Bhabha

provided with emergency power supply. The heat from re-circulating primary coolant water, the heavy water moderator and thermal shield cooling water is transferred to seawater coolant in shell and tube heat exchangers with seawater flowing in a once-through mode. Helium is used as the cover gas over the heavy water moderator. The operating power of the reactor is controlled by controlling the level of the heavy water inside the reactor vessel. Suitable instrumentation to sense the reactor neutron flux and automatic control loop to maintain power at

> *"…The force of the mighty nuclear power which could be available from the new facilities at Trombay would change the face of the rural areas and this is a historic event…"*
>
> ***– Pandit Jawaharlal Nehru,***
> *Inaugurating the CIRUS reactor and other facilities at Trombay on January 16, 1961.*

the desired level are provided. For rapid shut down of the reactor six shut off rods containing boron carbide are provided. These rods are parked above the active region of the reactor core during operation of the reactor. On sensing a fault the shut off rods are quickly inserted into the reactor to bring the fission chain reaction to a halt. Simultaneously, the heavy water moderator in the reactor vessel is also dumped to a separate tank known as dump tank to ensure long-term safety.

Prime Minister Indira Gandhi, a source of inspiration and encouragement, is taken on a tour of CIRUS - accompanied by Vikram Sarabhai

During the initial years of operation, CIRUS posed certain challenges and all of them were successfully overcome due to availability of multi-disciplinary expertise in DAE even in those formative years. In the early phase of operation of the reactor, a significant challenge was encountered due to obstruction of coolant flow arising from deposition of solids on the fuel element surface. The problem was solved by cleaning all the affected fuel elements with suitable chemicals and maintaining the required water chemistry. Such a problem was never encountered in NRX due to its once-through cooling water flow. To overcome the problem of frequent failures of clad of the fuel, thickness of aluminium clad was increased suitably. Another problem hampering the smooth operation was the growth of algae in the ball tank containing a large quantity of water (4000 m^3). This was investigated in detail and solved subsequently. By 1963, all the teething troubles of CIRUS were overcome and the reactor was operating at its rated power of 40 MWt.

Utilization

CIRUS has provisions for several neutron beam tubes that have been used extensively to carry out experiments utilizing neutrons from the reactor. Radioisotopes such as ^{60}Co (used for industrial radiography, irradiation of food and medical products), ^{192}Ir (used for industrial radiography), ^{99}Mo (used in nuclear medicine), ^{32}P (used for agricultural research) etc., are produced in CIRUS. The target materials for production of the above isotopes are filled in aluminium capsules and irradiated in the reactor using special assemblies called tray rods. These tray rods are handled every week while the reactor is on power. CIRUS alone was catering to all the requirement of radioisotopes till DHRUVA became operational in 1986.

To cater to the requirements of service irradiations for periods ranging from a few hours to a few days, facilities, known as self-serve units, are provided. The required target samples filled in capsules are enclosed in an aluminium ball and rolled into the specific positions for irradiation. On completion of irradiation, the balls are rolled out and collected in shielded flasks. These operations are carried out while the reactor is on power. The irradiated capsules are handled in hot cells having remote handling facilities. The reactor also has a pneumatic carrier facility, using which short-term irradiations can be carried out. This facility allows the target material encased in a capsule to be sent into the reactor and to receive it back in a laboratory, which is adjacent to the reactor. This facility is extensively used for neutron activation analysis, for determining trace quantities of materials in a given sample. The reactor has also been used for silicon transmutation doping experiments, which are needed by the electronics industry. In order to utilize and develop thorium technology, irradiation of thorium was started in CIRUS very early. After irradiation, ^{232}Th gets converted to ^{233}U (a fissile element). The inventory of ^{233}U necessary for fabrication of the first fuel charge of KAMINI reactor was produced in CIRUS.

A facility for test irradiation of power reactor fuel and materials is available at CIRUS. Development of Mixed Oxide fuel for the boiling water reactor at Tarapur, was taken up using this facility. This facility is also utilized for validating various design assumptions and analyses by carrying out test irradiations and later examining the fuel. Structural materials for PHWRs, like the end shield, zircaloy pressure tubes etc., were irradiated for material evaluation in CIRUS.

Rejuvenation of CIRUS

The CIRUS reactor has operated for nearly four decades with an average availability factor of about 70% and by reactor standards, it is old. However, given the long innings at the crease, some of the components started showing signs of ageing during the early nineties, necessitating extra efforts for maintenance. The upkeep of this reactor and its life extension threw-up challenges, which have been effectively met by adopting innovative methods. At the end of the design life of 30 years, detailed studies were undertaken to evaluate the condition of various important systems and components of CIRUS to determine their remnant life. Investigations were carried out on radiation damage of reactor core components,

Wigner energy

Graphite is used in some reactors to slow down neutrons. When fast neutrons collide with the carbon atoms of graphite, they are slowed by transfer of energy to the carbon atoms. A part of the transferred energy causes displacement in carbon atoms of the graphite crystal lattice. This process stores-up energy called “Wigner energy”. Under certain conditions, this energy can be catastrophically released.

the erosion and corrosion of piping and valves in the primary coolant system, moderator and cover gas system. “Wigner energy” stored in the graphite reflector, emergency cooling water storage reservoir (ball tank) and other components were also studied to ensure safe operation of the reactor. These studies indicated that the active life of the reactor can be extended by about 10 to 15 years after replacing / repairing some of the important equipments and components. Estimates of the cost and effort required were found to be economical and beneficial to the nuclear programme. The experience gained in partial decommissioning of the reactor for the required refurbishing activities would be invaluable in planning future decommissioning activities for both research and power reactors. An extended outage of the reactor was taken for implementing the required actions. After unloading irradiated fuel from the core and draining the coolant circuit, further inspections were undertaken which led to identification of certain additional refurbishing requirements. However, from detailed studies it was concluded that there was no necessity for replacing the Reactor Vessel (RV) itself and reactor operation can be resumed with the same reactor vessel. In order to ensure good health of RV, chemistry of moderator and cover gas systems are maintained well within the stipulated limits. This has ensured that RV tube leaks are rare. In fact, over forty years of operation, only two positions have been plugged for leak using remote handling tools developed in-house. These repairs were carried out after inspection with remote viewing cameras, Eddy Current Testing (ECT) for wall thickness measurement and volumetric examination for flaw detection by specially developed differential coil ECT probe.

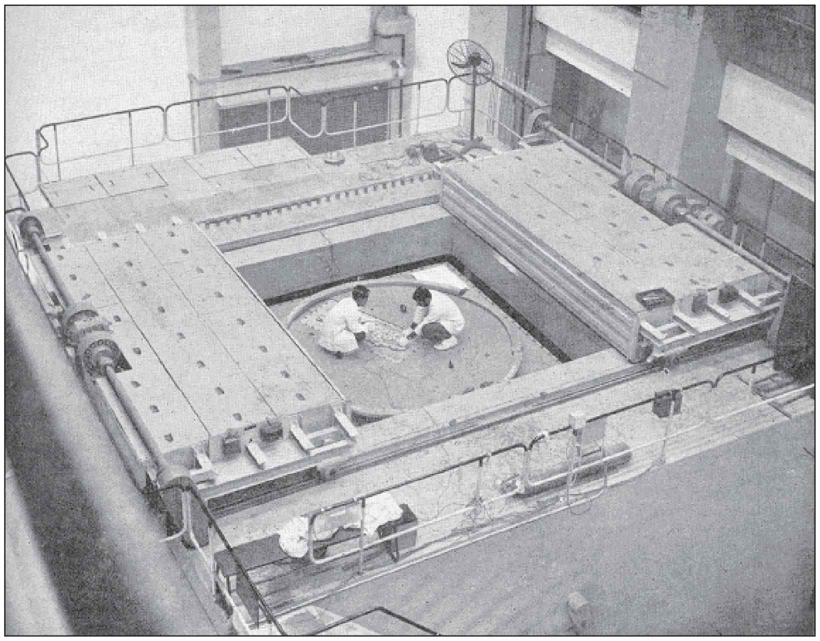
Another challenging maintenance task carried out remotely, involved flange joints having elastomer gaskets and joining the aluminium pipes extending from top of the reactor vessel with the stainless steel helium cover gas pipelines located above the upper steel thermal shield. Detailed checks established that the flange joints between aluminium and stainless steel pipes were leaking. It was suspected that this was caused by deterioration of elastomer gasket material due to irradiation and ageing. These flanged joints are located in a 200mm vertical gap between the top steel thermal shield and the top concrete biological shield at a distance of about 4m from the top of the reactor. Special split sealing clamps were remotely installed around all the 8 flange joints and tightened to compress the old elastomer gaskets. Another problem that required remote installation was related to a minor leak in the top aluminium thermal shield. A leak was identified at a weld joint in a coolant inlet pipe of the top aluminium thermal shield. As the leak location was near the core region and 5m below the top of the reactor, a hollow aluminium plug with expandable ‘C’ shaped rings at both ends and a straight portion in the middle to cover the leaky section, was developed and installed remotely. This was another task that was carried out after extensive in-house developmental work and mock-ups.

All major civil structures were inspected and the observations have been documented for future comparisons. Detailed inspection of sea water pump house and the jetty constructed on pre-cast RCC piles was carried out and they were found to be in good condition. A detailed seismic re-evaluation of these various structures was carried out. It was found that almost all the structures meet the current safety standards. Over the years a small leak had developed near the bottom of the central inspection shaft of ball tank near a concrete pour joint. During the refurbishing outage, the tank was emptied and decontaminated and repairs were carried out successfully. The central vertical inspection shaft of the ball tank was strengthened by steel jacketing it with supporting stiffeners to meet the present day seismic loading requirements.

Improvements were also carried out in emergency ventilation system with combined charcoal cum HEPA filters for trapping radioiodine in an unlikely event of an accident. Fire detection and mitigation system was also improved with the state-of-art technology. Safety analysis was carried out for the refurbished CIRUS reactor to ensure that the overall safety objectives are met. A low temperature vacuum evaporation based desalination unit developed in house, was integrated with the primary coolant system of CIRUS.

After systematic commissioning of all reactor systems and loading of fresh fuel the reactor was restarted and taken to high power operation. The rejuvenated workhorse is back in action to serve the nation for many more years to come. The experience so gained will be useful for refurbishment, life extension and upgradation of operating nuclear reactors.

ZERLINA - Delight of the Reactor Physicist



Top view of ZERLINA

ZERLINA, the **Z**ero **E**nergy **R**eactor for **L**attice **I**nvestigations and **N**ew **A**ssemblies, was the third reactor to be built in India and was a totally indigenous facility. This reactor was built for the purpose of studying various core geometries. The geometric arrangements of the fuel elements in the core, called the lattice has a significant bearing on the physics of reactors. At the time of embarking on a large power programme, there was a need for an experimental reactor in which different types of lattices could be assembled with ease. This would help evaluate their reactor physics characteristics and develop satisfactory computer codes for the design of both research and power reactors. These considerations led to the building of ZERLINA.

This reactor was made critical for the first time on January 14, 1961. A cylindrical aluminium tank of 2.29 m diameter and 4.35 m height with dished end at the bottom formed the reactor-vessel. A 73.5 cm thick and 3 m high radial graphite reflector surrounded the tank with 7 cm annular gap in between the tank and reflector. A 1.5 mm thick cadmium shutter operated in the gap. The fuel was in the form of natural uranium metal, 3.56 cm in diameter and 243.8 cm in length, clad with 1 mm thick aluminum. A maximum of 80 fuel rods could be suspended from the top of the tank in the heavy water, which acted as both moderator and coolant. The moderator was covered with dry nitrogen gas. The mass of the fuel in the first core of ZERLINA was 3.7 tons. The maximum operating power was restricted to 100 Watts, which corresponds to a thermal neutron flux of about 10^8 n cm^{-2} s^{-1} at the core centre. The temperature rise in the fuel rod, at this power, was very low. The lattice pitch could be easily changed by re-arrangement of girders placed on the tank top from which the fuel was suspended and so the core could be easily changed to obtain different configurations. It was possible to create a thermal column having a thermalised neutron flux, by removing a few peripheral fuel rods. A thermal column is useful as reference while measuring neutron spectrum parameters. A facility called batch addition/ removal system that added/ removed a small but accurate quantity of heavy water into/from the core was useful for measurement of level coefficient of reactivity near critical height.

Utilization

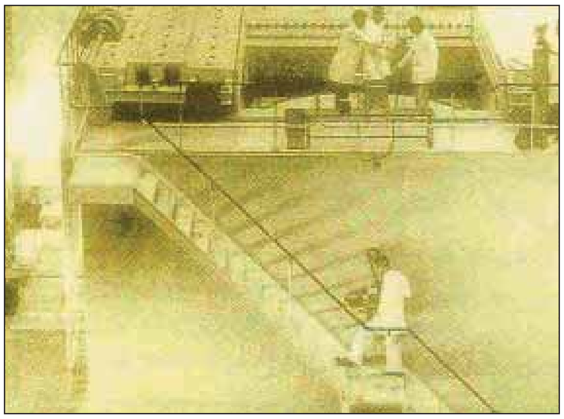
The zero energy critical facility was mainly used to find a geometry and material dependent factor called buckling of a sub-critical lattice. This was evaluated by a method called the substitution technique and by comparison with theoretical computations / validations or code adjustments. In this technique, the lattice under test forms one of the zones of a two-zone critical system, the second zone consisting of a lattice whose properties are known. The reference core is studied by measuring three dimensional neutron flux distributions for buckling and intra-cell parameters All the measurements for DHRUVA reactor lattice physics were carried out in ZERLINA during 1974-78.

This versatile critical facility was also used for many other experiments, where computation modeling of neutron absorbing / producing assemblies is difficult. Experiments were carried out on a few such assemblies proposed for the PHWR program. For example, plutonium boosters were studied extensively in ZERLINA during 1973-76. Reactivity worth and fine structure flux distribution inside the boosters and flux perturbations in the core were measured and compared with calculations.

ZERLINA played an important role in development of measurement techniques, inter-calibration of neutron activation foils made of alloy materials and standardization of counting set-ups and procedures - all aimed at improving the measurement accuracies. A beta scanner was developed for scanning beta activity from a long copper wire irradiated in the core to measure the axial/radial flux distributions. Reactor physics experiments carried out in ZERLINA included measurement of level coefficient of reactivity, temperature coefficient of reactivity, purity coefficient of reactivity, period *Vs* reactivity, rod worth, buckling and danger coefficient.

ZERLINA was also used for conducting experiments in reactor physics, which constitute a part of the curriculum in the Training School at BARC. Trainees conducted experiments like measuring the activity induced in the radiation detectors for obtaining the various reactor physics parameters. Approach to criticality, making reactor critical, raising or lowering of the power and measurement of reactivity worth by measuring reactor period, were some experiments that gave the trainees a feel for the subject of reactor physics.

ZERLINA was decommissioned in 1983, as it was felt that a lattice experiment facility was no longer needed. ZERLINA played a key role in evaluating various design parameters and some of the components of DHRUVA reactor, like reactor instrumentation and start-up system. The reactor core characteristics of MAPS reactor and various instruments, including safety related systems developed for India's nuclear programme by different agencies were also evaluated in ZERLINA.



ZERLINA - The stepping stone to power reactors

DHRUVA – A Milestone in Indigenous Development of Technology

Panoramic view of DHRUVA

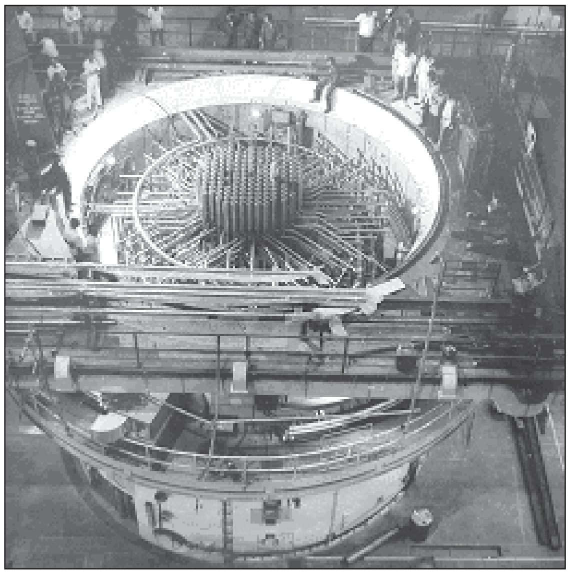
DHRUVA is a 100 MWt research reactor, with heavy water as coolant, moderator and reflector. It is fuelled with natural uranium in metallic form, which is clad with aluminium. The maximum thermal flux of the reactor is 1.8×10^{14} n cm^{-2} s^{-1}. The reactor has 146 channels with 127 fuel rods and nine shut off rods made of cadmium. Experimental and isotope production facilities are provided in the reactor. The primary cooling system has three loops, with each loop having a main coolant pump, a heat exchanger and a shut down cooling pump. In DHRUVA, the heavy water coolant and moderator are designed to get intermixed, as the reactor vessel rides on the coolant system as an expansion tank. This innovative feature also makes the system inherently safe, since any breach in the primary coolant boundary automatically brings down the moderator level in the reactor vessel, resulting in the shutdown of the reactor. Also the large moderator inventory would augment the coolant during initial phase of cooling after breach. The secondary coolant for the reactor is de-mineralized light water and the ultimate heat sink is seawater. For shut down cooling requirements of the reactor, three smaller capacity pumps are provided, each coupled to two different prime movers, based on diverse principles, one an electric motor and other a water turbine. It can be operated either by a motor with on-site power sources, or by a turbine operated by water from an overhead storage tank. The water, after running the turbine passes through the heavy water heat exchangers as secondary coolant, thus performing a dual function. An underground dump tank is provided to collect and pump back this water to the overhead tank.

The reactor is housed in a rectangular concrete containment building with a basement and a sub-basement for housing the various process equipments. Equipment connected with heavy water system, cover gas system and engineering loops are located inside the reactor building. An overhead concrete water storage tank, built over the roof of spent fuel storage bay, supplies water to power the water turbine drive of shutdown cooling pumps.

Design Innovations and Challenges in the Making of DHRUVA

The design, construction, commissioning and operation of DHRUVA was a completely indigenous effort. In addition to the engineers and scientists of DAE, several government institutions and public sector and private industrial organizations in the country participated in the effort. A few of the components for which the Indian industry could not meet the required specifications, at that point of time, had to be imported. With the experience of leaky coolant channels requiring plugging in NRX and in CIRUS reactors, it was decided that the coolant channels of the DHRUVA reactor would be made of zircaloy rolled to the stainless steel extension tube and that these would also be replaceable.

Positions of the fuel rods and the shut off rods were also kept interchangeable to have flexibility for any future changes in the fuel and the core configuration. To meet this requirement, the shut off drive mechanism, which includes the drive motor, clutch assembly, stopper assembly, dampener and guiding pulleys had to be accommodated in a 120 mm diameter guide tube. The height of the drive mechanism is 700 mm. This was one of the most challenging tasks for the designer. Also, being the primary fast acting shut down device, it had to perform most reliably. The design was made in such a manner that the clutch of the shut off rod will de-energize on a trip signal and the rod will drop into the reactor within three seconds under gravity and shut down the reactor.



DHRUVA - Erection of channel outlet pipes above the reactor vessel

Fuelling machine of DHRUVA, which loads and discharges nuclear fuel

Designing of the re-fuelling machine for the safe and reliable operation was another challenging work. The metallic uranium fuel clad with aluminium, has a length of over nine meters along with its aluminium shield. The machine carrying two such fuel assemblies is required to make a leak-tight joint with the coolant channel and continue cooling the fuel during the entire refuelling, till the assembly is discharged into a water filled pool. For this, the fuelling machine, provided with lead filled compartments for radiation shielding and weighing over 300 tons, is required to be aligned with the channel within an accuracy of ± 0.25 mm. Also, the fuelling machine is required to be capable of remote locking and un-locking of the fuel assembly with the fuel channel. A suitable ball guide mechanism was designed for this purpose. Though the fuelling machine is designed to carry out the difficult task of on-power refuelling, for safety reasons, refuelling is carried out only with reactor and main coolant pumps in shut down state. To fulfil the requirement of weekly delivery of radioisotopes for health care, the isotope tray rods are handled on-power, for which certain innovative design features have been developed so that safety during handling is ensured.

Compared to CIRUS, considerable improvements were incorporated in the shielding design of DHRUVA to limit radiation dose to operating personnel and public and also to have ease of decommissioning. Instead of cast iron side thermal shields used in CIRUS reactor, the design was simplified by submerging the reactor vessel into a pool of light water. This also reduced the induced argon activity in the ventilation air. The thermal shield on top of the reactor was designed keeping in mind the ease of transportation from the fabricator's premises to the reactor site. Thus, instead of using heavy metal slabs with cooling coils, a box type design with steel balls (filled at site) was adopted. Many of the reactor structure designs used in DHRUVA were forerunners and were adopted in the standardized Indian PHWR, which was being designed at that time.

To ensure long design life of the reactor, it was decided that the reactor vessel should be made of stainless steel. Manufacturing of the reactor vessel, which is over 7 meters in height, 3.72 meters in diameter and weighing about 30 tons, was carried out in-house to meet high standards of quality. The material used for the reactor vessel was subjected to a 100% volumetric and surface non-destructive testing. The weld joints were subjected to 100% radiography. Many evolving technologies such as plasma arc cutting of thick (50 mm) SS plates, precision welding with TIG and electron beam were developed and successfully employed for fabrication of the reactor vessel. A formidable problem of weld shrinkage and distortion was experienced during the fabrication process. To solve this, the sequencing of various welds was adjusted so that the shrinkage and distortion could be progressively balanced out. The joints between the lattice cup and the inlet plenum of the reactor vessel, which were not accessible for normal type of welding, were joined using electron beam

Prime Minister Rajiv Gandhi, accompanied by P. K. Iyengar, visiting DHRUVA

Russian President V. Putin with R. Chidambaram and A. Kakodkar. Seen in the backdrop is DHRUVA

welding. This advanced technology of welding with electron beam without an electrode was employed in the country for the first time on a large critical component, in a successful manner.

Fabrication of the 300 mm diameter beam hole re-entrant cans for use in neutron beam research posed another big challenge. As zircaloy-2 tubes of this diameter were not available, it was decided to roll and seam weld zircaloy-2 plate to form a tube. Plates of 9 to 12 mm thickness were hot rolled down to 4.0 mm thickness. Since hot rolling showed considerable variation in thickness, a cold rolling facility was set up at Midhani, Hyderabad, and plates with requisite uniform thickness were obtained. Further, electron beam welding of zircaloy-2 hemisphere to the zircaloy-2 tube was carried out at Defense Research and Development Laboratories, Hyderabad, after successful developmental trials to keep the weld distortion to a minimum. The welds were made in a glove box with argon atmosphere.

The beam hole re-entrant can was joined to the stainless steel nozzle of the reactor vessel using a rolled joint. At that time, the indigenously available experience of rolling cylindrical sections of stainless steel and zircaloy was limited to about 125 mm diameter and hence making the rolled joint with a diameter of 300mm was a challenge. The required technique was developed and extensive mock-up trials were carried out to attain perfection in quality. The trial joints were subjected to pullout test and the joints were ensured to have a factor of safety of five, as these joints were to last the life of the reactor.

A part of the neutron beam tube, which was a separate component, had to be embedded in the heavy concrete biological shield of the reactor where it was to be positioned accurately during pouring of the concrete. The two tubes were to be aligned to one another with in an accuracy of ±0.5 mm and an angle of ±10 sec of an arc and then ultimately joined together by welding. If the two pipes were welded together, before pouring the concrete, the shrinkage of concrete would give rise to bending stresses on the rolled joint. Unless such accuracy was attained, it would not be possible to use the beam tubes for their intended purpose. The amount of distortion that was likely to be encountered due to pouring of the concrete with a density of *3.5 g/cc* would be high. The task was accomplished successfully by providing suitable supports and restraints.

Curved neutron guides transport neutrons to an area outside the reactor hall, which accommodates a large number of instruments. This area is required to have a low radiation background. These guides carry neutrons over large distances by the principle of total internal reflection from a perfectly smooth surface with very little loss in intensity of neutron beam. Use of curved guides eliminates unwanted radiation from the reactor to the experiment area and two curved guides were developed indigenously for this purpose.

Commissioning and Operational Challenges

A major observation during the initial phase of operation of DHRUVA was the higher radioactivity in the heavy water primary coolant system due to fuel-clad failures. This had restricted the reactor operation and resulted in premature removal of some of the fuel assemblies. Apart from the economic penalty, this could result in increase in radiation fields on the equipment and piping thereby making it difficult to carry out the operation and maintenance activities. Hence this challenging problem had to be resolved before the reactor operation could be continued further. The problem was analyzed in detail by post irradiation examination of a number of fuel assemblies removed from the reactor. The root cause for the fuel failure was understood to be excessive fuel vibration. Investigations revealed that the diametral clearances between the fuel and the coolant channel bottom cup was the primary cause for excessive vibration. In addition, the frequency of vibration of the primary coolant heat exchangers located on the support girders was found to be closely matching with the natural frequency of the fuel assembly, thus giving rise to resonant vibration. The fuel assembly design was modified by introducing suitable soft supports in the form of aluminum leaf springs to minimize vibration. The modified assemblies were extensively tested, both out-of-pile and in-pile, to eliminate the vibration problem.

Due to excessive fuel vibration during the initial phase of operation, the aluminum cladding of the fuel was subjected to mechanical abrasion leading to wear-out at specific locations. This resulted in increase of aluminum and uranium content in the coolant water and the water became turbid. The turbidity was found to be in colloidal form and could not be removed by normal filtration techniques. Magnesium loaded ion exchange resin, developed in-house, was successfully employed in the removal of turbidity. A centrifuge separator was also successfully utilized for the same purpose. With the modified fuel assemblies and after control of the coolant water turbidity, the reactor was restarted during November 1986 and power was gradually raised to rated power in January 1988.

Visit of Prime Minister A.B. Vajpayee to DHRUVA, accompanied by Dr. Anil Kakodkar, Chairman AEC

Another significant experience during the initial phase of operation of DHRUVA was related to the Failed Fuel Detection (FFD) system. Outlet coolant water sample from each fuel channel is led to radiation detection units through suitably sized sample tubes. The increase in radiation field sensed by the detectors helps in detecting and locating the position of the failed fuel. In certain cases, the FFD system was unable to locate the failed fuel position. This was analyzed to be due to reduced flow in the sample line from this position to FFD system. Detailed analysis indicated that some of the sample lines were blocked by the release of dissolved helium cover gas present in the coolant. This was essentially happening when the shut down core cooling pumps were in operation, as the coolant system pressure in the sample tube tap-off region is at a lower value than the saturation pressure of the dissolved helium. Maintaining a lower pressure in this region at the top of the coolant channel was required to minimize the probability of leakage of the coolant heavy water in case of failure of the elastomer sealing rings. Suitable modifications and changes in the operating procedures were incorporated to ensure that the sample lines have proper flow and thereby the performance of the system was enhanced. Upgradations as necessary were also carried out on the FFD data acquisition and processing system.

Utilization

A large number of neutron beam tube facilities with diameters up to 300 mm are provided in DHRUVA. These beam tubes are extensively utilized for neutron beam research. Apart from the scientists of DAE, researchers from various academic institutions also utilize these facilities under the aegis of DAE inter–university consortium. DHRUVA is also provided with a pneumatic carrier facility for irradiation of samples that yield short-lived isotopes on irradiation. These samples require minimum transit time between the completion of irradiation and counting. Large irradiation volume has also been provided in the form of special assemblies to facilitate production of various radioisotopes for radio- pharmaceutical, industrial and research applications. These special assemblies are handled on-power every week to enable handling and delivery of the radiopharmaceuticals to various hospitals and radiation medicine centers. Commissioning of DHRUVA facilitated production of high specific activity radioisotopes in large quantities.

To facilitate the fuel and structural materials development program for Indian pressurized water reactors, a facility is installed in one of the 300 mm diameter radial beam holes which enables neutron irradiation of samples of fuel cladding, pressure tubes, end fittings, end shield etc. at controlled temperatures to assess the effect on their mechanical properties after irradiation. An in-pile loop facility with higher heat removal capacity (than available in CIRUS) is provided in DHRUVA. This facility can accommodate higher size fuel bundles of 500 MWe PHWR and Advanced Heavy Water Reactor for testing.

Radiopharmaceuticals are a special class of radiochemical formulations suitable for administration to humans orally or intravenously to carry out organ investigations in vivo or for therapeutic effect. About 30,000 consignments of radiopharmaceutical products with an activity of over 1000 Curies (mostly ^{99}Mo and ^{131}I) are handled per annum resulting in an estimated 2.5 to 3 lakh patient investigations and treatment.

Facility to extract a beam of neutrons from the DHRUVA for its utilization in experiments

View of PURNIMA reactor

As the experience gained in the design and operation of thermal research reactors was being fruitfully used to launch a power programme based on PHWRs, work was initiated to study reactor systems based on plutonium and uranium-233 and various aspects of utilization of thorium. During the period 1970 to 1992 design, construction and commissioning of the PURNIMA series of research reactors was undertaken. We recall that the second and third stages of our nuclear programme entails the construction of fast breeder reactors which use Pu and thermal reactors with ^{233}U based fuels respectively. The PURNIMA series of reactors were built for gaining experience in designing such reactors. Extensive experiments were carried out to understand physics of plutonium fuelled fast reactors as well as ^{233}U fuelled thermal reactors using these facilities. These experiments provided a thrust to the development of technology of plutonium fuel fabrication, separation of ^{233}U from irradiated thorium, reactor control instrumentation etc.

PURNIMA (**Plu**tonium **R**eactor for **N**eutronic **I**nvestigations in **M**ultiplying **A**ssemblies) was designed and built as a zero-energy fast research reactor. This was India's first fast critical facility with plutonium based fuel and it attained first criticality on May 22, 1972. PURNIMA (later called PURNIMA-1) had a compact core volume of 3 litres. The fuel pins were made of 11 mm diameter tubes made of type 347 stainless steel. In these pins, the central 180 mm was occupied by sintered PuO_2 pellets, which were held tightly between two 80 mm long axial molybdenum reflectors. The tube ends were sealed with welded end plugs. The fuel pins were contained in an asymmetric hexagonal core vessel made of stainless steel and the fuel pins were held in position from a 40 mm thick top grid plate by retaining nuts. The fuel elements were arranged in a triangular array. The core was radially reflected by 170 mm thick copper reflector followed by 230 mm thick mild steel shield. The copper reflector consisted of two parts: the movable lower part, which was mounted on a carriage drive mechanism along with the entire core assembly and the stationary upper part which was rigidly fixed along with the mild steel shield on a steel platform. Six safety rods and three control rods, all of molybdenum, were located in vertical channels in the copper reflector close to the core.

Fabrication of fuel for PURNIMA provided the first experience in handling and fabrication of Plutonium based fuels. PURNIMA reactor was extensively used for carrying out detailed studies on fast reactor neutronics. Experiments were carried out at PURNIMA to obtain reactor physics data such as critical mass, reactivity worth of various reflector blocks, safety and control rod worth and prompt neutron lifetime.

The core of PURNIMA was dismantled after carrying out all planned experiments. The infrastructure developed for this reactor was later used to house the rest of the PURNIMA series of reactors. Making of PURNIMA was an important step towards development of fast breeder reactor technology in India.

PURNIMA-2

After completion of experiments with PURNIMA-1 reactor, a ^{233}U uranyl nitrate solution fuelled beryllium oxide reflected zero energy thermal reactor called PURNIMA-2 was set up, which attained first criticality on 10 May 1984. One of the major objectives of PURNIMA-2 was to perfect the technique of carrying out criticallity experiments. It was possible to make this reactor critical with 397 gm of ^{233}U. PURNIMA-2 was the first homogeneous reactor built in India. In a homogeneous reactor, the nuclear fuel (in this case uranyl nitrate) is intimately mixed with the moderator/coolant water in the form of a solution.

The reactor core was contained in a cylindrical zircaloy vessel of height 500 mm and diameter 148 mm. This was made from a single block of zircaloy. The core vessel was placed inside a zircaloy box and the interspace was filled with graphite to reduce

View of fuel handling facility of PURNIMA-2

leakage of neutrons. The safety plates were made of cadmium sandwiched in aluminium. For fine control, stainless steel clad boron rods were provided. The core volume was variable between 4 and 8 litres according to experimental requirements. The fuel concentration was 60 to 130 gm uranium per litre of solution. Enriched uranyl nitrate with 98% ^{233}U as a solution in water was used for the experiments.

The main experimental study conducted with PURNIMA-2 was determination of variation of critical mass as a function of fuel solution concentration and determination of worth of safety and control devices. For a fuel concentration of 116.6 g/l, the critical mass was 457 g of ^{233}U. Through these studies the optimal H: ^{233}U ratio required to attain the minimum critical mass for the particular system geometry was established, which helped to freeze the design of KAMINI reactor.

PURNIMA-2 made a significant contribution towards development of technology for ^{233}U fuelled reactors. The reactor was shut down in 1986.

PURNIMA-3

PURNIMA-3 was built and used as a test bed and zero energy critical facility to study the core of the KAMINI reactor. In this reactor ^{233}U– Al alloy (20%) flat plate type fuel subassemblies and beryllium oxide reflectors clad with aluminum / zircaloy were used. The reactor was made critical during 1992.

Experiments were performed in PURNIMA-3 to study various core-reflector configurations of interest to KAMINI reactor. Measurements were done with aluminum / zircaloy canned beryllium oxide reflector modules to determine k_{eff} of different core-reflector configurations, evaluation of reactivity contributions of removable reflector modules, dummies etc. and reactivity worth of safety-control plates. Core of PURNIMA-3 was dismantled and the fuel was transferred to KAMINI reactor in 1996.

Fast Breeder Test Reactor (FBTR) – Heralding the Second Stage of Nuclear Power Programme

View of the FBTR complex at IGCAR

Fast breeder reactors constitute the second stage of India's three-stage nuclear energy programme, for effective utilization of the country's limited reserves of natural uranium and exploitation of its large reserves of thorium. The decision to build a Fast Breeder Test Reactor was taken in 1968. Towards this objective, Reactor Research Centre (RRC), was set up at Kalpakkam, in 1971, totally devoted to the development of fast reactor technology. Initially, an agreement was signed with CEA, France for the design of FBTR, based on the experimental Rapsodie reactor as well as for training of personnel and transfer of manufacturing technology of critical components. In an unprecedented move hitherto in the history of DAE, a team of 30 engineers and scientists were sent to the French nuclear centre at Cadarache, to complete the design of FBTR in consultation with French engineers. However, CEA, France indicated their inability to assume any responsibility for the construction phase as the services of only five French engineers were sought to assist the design and construction team. The Department of Atomic Energy took the bold decision to go ahead, taking it as a challenge. The responsibility of construction of the reactor was thus totally with India, with major participation by Indian industries. This was a radical departure from earlier occasions, considering that the BWR at Tarapur plant was a turnkey project executed by M/s General Electric, USA and that the first PHWR at Kota was constructed by Canada with Indian personnel working as part of the team. In 1985, RRC was renamed as Indira Gandhi Centre for Atomic Research (IGCAR) after the late Prime Minister of India, Smt. Indira Gandhi.

Heat generated in the reactor is removed by sodium. While the Rapsodie reactor had sodium to air heat exchangers as the final heat sink, it was decided to get the know-how on sodium heated steam generators (SG) also from France, based on their experience with the tests on SG for their 250 MWe Phenix reactor. Since a fast reactor does not have a neutron moderator such as heavy water or light water, the reactor core has to be compact resulting in very high volumetric power density. For example, the power produced per cubic metre core of a fast reactor plant is 550 MWt compared to 8 MWt for a PHWR plant of the same capacity. Thus, it is imperative to use a very efficient heat transfer fluid as a coolant, which should also possess favourable nuclear characteristics of low neutron moderation and absorption. Liquid metals, and among them liquid sodium, meet almost all the requirements of a fast reactor coolant with its high thermal conductivity, reasonable specific heat, low neutron moderation and absorption and high boiling point giving a large operating temperature range at near atmospheric pressure.

FBTR was designed as a 40 MWt, loop type, sodium cooled fast reactor. The sodium flowing through the core, called primary sodium, picks up the heat and transfers it to secondary sodium. Secondary sodium in turn transfers the heat to water/steam once-through steam generator (SG) modules. Superheated steam at 753 K & 125 b from the SG modules is fed to a conventional steam water circuit comprising of a turbine-generator. A secondary sodium circuit is interposed between the primary sodium circuit and the steam-water system from considerations of safety. The core consists of hexagonal shaped subassemblies of fuel, nickel reflectors and steel reflectors. The subassemblies are supported at the bottom by the grid plate. The fuel chosen for the nominal core had the same composition as Rapsodie MOX fuel, viz. 30% PuO_2 -70% UO_2, with the latter enriched in ^{235}U to 85%.

Reactor construction was started in 1971, and civil works were completed by 1977. Most of the components were manufactured by the Indian industries, and were installed in 1984. Except for the grid plate, one control rod drive mechanism, one sodium pump and raw materials for critical nuclear

Fast Breeder Reactors for long term energy security

"It is a matter of great pride that DAE is stepping into the commercial phase of the second stage of the power programme, during its 50th anniversary, by commencing the construction of a 500 MWe Fast Breeder Reactor Project (PFBR) at Kalpakkam. Signalling the commencement of FBRs at this crucial juncture, while the first stage based on PHWRs, is at its half way mark, is vital to ensure a smooth transition to result in unhindered and enhanced contribution of nuclear energy in the coming decades.

Looking back, the successful design, construction and operation of the Fast Breeder Test Reactor (FBTR) to demonstrate the technical viability of FBRs, has been a major milestone in the history of IGCAR and DAE. The more than 135 GWd/t burn-up of the unique high plutonium based carbide fuel and mastering the sodium handling technology including 1,50,000 hours continuous operation of the sodium pumps without maintenance are just a few major technological highlights of FBTR. This success has been achieved with intensive intra-departmental collaborations, particularly with BARC and NFC and due to an active industry-academia networking pursued by IGCAR.

Living up to the vision of Bhabha and growing on the seeds sown by Sarabhai, mission-oriented engineering, science-based technology and frontier research activities pursued at IGCAR have provided confidence and impetus to embark on a large scale FBR programme. The design and operational experiences of FBTR along with in-pile data of more than 320 years of world wide experience on large sized FBRs, has provided the necessary confidence towards design and technology development of an indigenous 500 MWe power reactor.

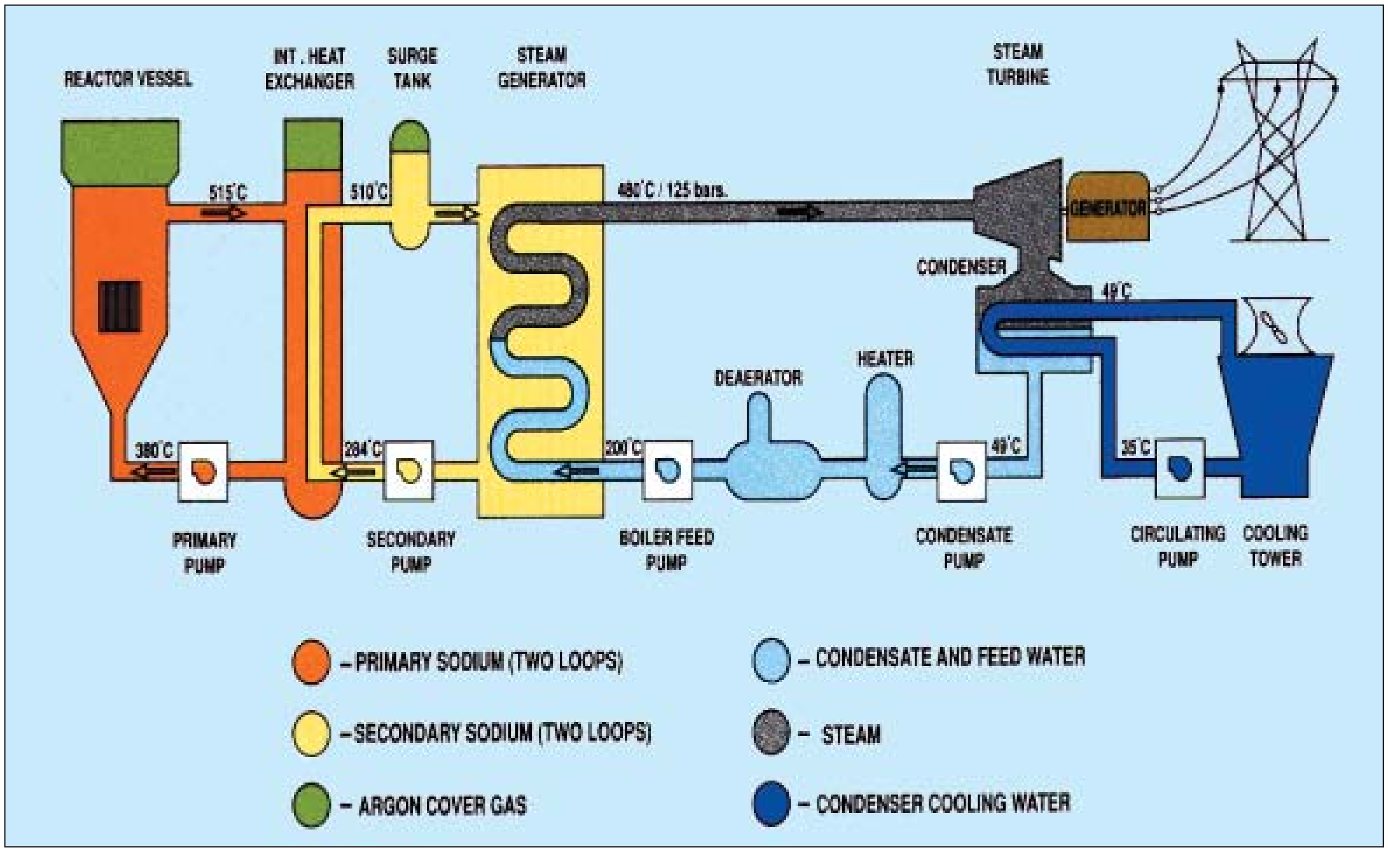
Closing the fuel cycle is inevitable for sustaining the planned nuclear power programme. An integrated fuel cycle facility is being planned at Kalpakkam to cater to the requirements of reprocessing the spent fuel from PFBR and subsequent fabrication of the fuel to be loaded back to FBR. In the coming decades, rigorous R&D efforts necessary for complete fuel cycle of metallic fuels will also be undertaken. Kalpakkam would then benchmark a unique distinction of developing the technology for reprocessing of a wide variety of high burn-up fuels including carbides, oxides and metallic alloys.

Based on the design experiences of FBR and with the confidence in fast reactor fuel reprocessing, a few FBRs of 500 MWe capacity each, will be undertaken with emphasis on improvements in techno-economic aspects and safety. Emphasis will be to decrease the unit energy cost significantly in future FBRs, so as to make them economically competitive. This will be achieved by incorporating a variety of innovative features, such as increasing maximum burn-up of fuel to 200 GWd/t and by increasing the reactor design life to 60 years. These would be possible by the application of advanced core component materials currently under development, improved manufacturing and quality management technologies as well as enhanced confidence in predicting high temperature properties of advanced alloys. These efforts would ultimately result in making fast breeder energy affordable and competitive. Following the first four FBRs, the subsequent series of fast reactors of 700 MWe or 1000 MWe capacity, would be designed based on metallic fuels with emphasis on increased breeding ratio and minimising doubling time, to ensure sufficient availability of plutonium for sustaining the rapid growth of the envisaged FBR programme. Thus, the road map for FBR technology for the next twenty years has been worked out and is being implemented in full earnest.

For efficient utilisation of the limited uranium resources and for ensuring long term energy security of the nation, a larger role of fast reactor technology in the coming decades is inevitable. When the Nation would look back after 50 years, with the status of a developed India, the success of FBR programmes in the Department to provide sustainable and affordable electricity for the industrial and economic advancement of the country, would be acknowledged and appreciated. There is no doubt in my mind that, India is poised to become a global leader in fast breeder technology by 2020."

Baldev Raj

- Director, IGCAR



Schematic showing the various parts of FBTR - the reactor, heat exchanger, steam generator and turbine

components, which were imported from France, all the other components were manufactured in India. The design and manufacture of this non-standard steam turbine was done indigenously. Sodium, because of its good heat transfer and nuclear properties, is used as a coolant in fast reactors. A sodium purification rig was set up at this center and 150 tons of sodium coolant of reactor grade was prepared from commercially available grade. The total indigenous content of FBTR was more than 80%, considered quite high in the light of the standards of Indian industries in the seventies and eighties.

The original design of MOX fuel with 30% PuO_2 - 70% enriched UO_2 to be supplied by the French, was reviewed in the light of the embargo subsequent to the Peaceful Nuclear Explosion (PNE) by India in 1974. Since Indian nuclear programme did not envisage use of highly enriched uranium at that stage, investment in an enrichment facility for the sole purpose of fuelling FBTR core was considered neither expedient nor viable. It was a challenge because completion of FBTR was in question. It was also an opportunity to develop advanced fuels. Hence, the option of using an alternate fuel rich in Pu was studied. Non-compatibility of higher concentrations of PuO_2 with sodium and difficulties in fuel fabrication ruled out the choice of Pu-rich oxide fuel. The carbide option was found feasible and chosen, notwithstanding the extra need for inert atmosphere during fabrication. Being an unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. Hence it was decided to operate the reactor initially with a small core. As against the original design of 65 fuel subassemblies of MOX fuel for the nominal core, the small core had only 22 fuel subassemblies of Mark-I (MK-1) composition (70% PuC - 30 % UC), during its first criticality.

Commissioning and Operating Experience

Commissioning of the reactor system was done in phases. Initially, primary and secondary sodium systems were commissioned, without steam generators in place. The reactor was made critical on 18th Oct 1985 and low power physics experiments were conducted. Steam generator modules were then connected to the secondary sodium circuits.

During an in-pile fuel transfer for performing a low-power physics experiment in May 1987, a major fuel handling incident

First criticality of Fast Breeder Test Reactor - seen in the photo is Dr. Raja Ramanna in the control room in October 1985

Prime Minister Rajiv Gandhi being briefed by C. V. Sundaram, Director, IGCAR in December 1985

took place in which the sturdy guide tube, which guides the fuel subassembly while loading into the reactor, was bent by about 320 mm. Removal of bent tube posed challenge as the operation had to be done with the reactor filled with sodium. A remote cutting tool was developed and with the aid of ultrasonic scanning device working under sodium, viewing of the bent parts was made possible. The guide tube was cut, ensuring that the chips do not fall into the reactor. The fast reactor community the world over appreciated this achievement. All subsequent fuel-handling campaigns (40 nos.) have been smooth.

With the limited charge of the first core, the reactor could be operated only up to a maximum power of 1 MWt till 1992 for intermediate power physics and engineering experiments. Subsequently, the steam water circuit was commissioned and the steam generators were put in service with power being raised to 8 MWt in Jan 1993. The reactor power has been progressively increased, reaching the highest power of 17.4 MWt in 2002. Twelve irradiation campaigns have so far been completed. With a view to enlarge the core to the nominal size, subassemblies of MK-II composition (55 % PuC - 45 % UC) were inducted in 1996. The MK-I fuel was operated at a Linear Heat Rating (LHR) of 400 W/cm, and has seen a burn up of over 135 GWd/t, without even a single fuel pin failure. This has been rendered possible by the encouraging results obtained from Post-irradiation Fuel Examination (PIE) studies conducted on three fuel subassemblies, discharged at burn-up of 25, 50, 75 and 100 GWd/t.

FBTR has operated for 32,967 h. All the sodium pumps have given trouble-free service for more than 150,000 h as on date. Sodium purity has been well maintained. Corrosion is so negligible that the engraved identification numbers on the fuel pins resident in the reactor for 18 years could be easily read in the hot cells during PIE. There was only one significant sodium leak, when 75 kg of primary sodium leaked from purification circuit into the nitrogen filled purification cabin in April 2002. The leak was due to manufacturing deficiency in a bellow sealed valve. There was no radioactivity release. The incident was effectively handled, system normalised and reactor operation resumed in three months. The once-through serpentine steam generators have operated without any tube leak. The steam generator leak detection system has been working trouble-free and is in continuous service since 1993.

In addition to its use as a self-driven irradiation facility for the Pu-rich monocarbide fuel, FBTR is being utilised as an irradiation facility for fuels and materials. With a neutron flux which is one order higher than that in a PHWR, FBTR could be used for studying the irradiation creep behaviour of alloys of zirconium that are being used in Indian PHWR. The creep behaviour was found to be in accordance with published International data. The present mission of FBTR operation is to irradiate the MOX fuel (29 % PuO_2) chosen for PFBR to the target burn-up of 100 GWd/t at the designed LHR of 450 W/cm. To get the required LHR in the core of FBTR, a small quantity of ^{233}U has been added to the test fuel. Since ^{233}U will be used in the third stage of India's nuclear energy programme, the irradiation is expected to provide some insight into its behaviour as well. The irradiation commenced in July 2003 and will require 484 Equivalent Full Power Days (EFPD) of operation, spanning the next five irradiation campaigns to reach the target burn-up. As on Dec 2004, the test fuel has reached a burn-up of 45 GWd/t.

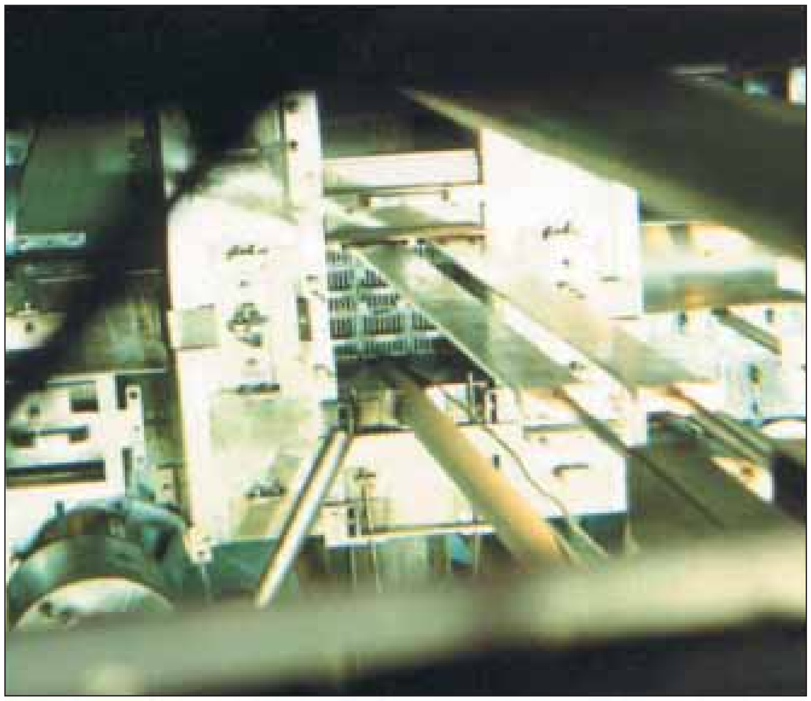
Safety Experience

Several safety related engineering and reactor physics tests have been conducted in FBTR to validate the data assumed and the codes used in design and safety analysis of the reactor. The completed physics tests include measurement of the various feed back coefficients of reactivity, effect of sodium voiding of the reactor, flux measurement above sodium and evaluation of the performance of the failed fuel detection system. The completed engineering tests include study of the evolution of various reactor parameters during postulated incidents of high and low probabilities. A major low probability incident tested was the capability for removal of decay heat from the reactor by natural convection in the primary and secondary sodium loops under conditions of non-availability of all the sodium pumps. All the tests have proved that the reactor is quite safe under all postulated incidental scenarios.

The experience in construction, commissioning and satisfactory operation for the past twenty years have demonstrated the mastering of the multi-disciplinary technology for energy production using a fast reactor and provided sufficient feedback to enable the launch of the work on building a Fast Breeder Reactor project (PFBR) of 500 MWe capacity.

The maximum annual activity released to atmosphere by FBTR is 437 Ci as of March 2005. This is well below the permitted limit by AERB guidelines. Cumulative occupational exposure so far is only 66.8 man-mSv (6.68 man-rem). Most of this small exposure has taken place during fuel assembly operations rather than due to reactor operation. During the past twenty years, there has been no significant event of abnormal radioactivity release, personnel or area contamination thus confirming the worldwide view that the sodium cooled reactor concept gives low radiation doses to operating personnel and low releases to environment.

KAMINI – Opening the door for the Third Stage of Indian Nuclear Power Programme.



View of reactor core through the water shield

KAMINI (Kalpakkam Mini) reactor located at IGCAR, is a special purpose 30 kW research reactor. This reactor is versatile and has facilities for carrying out neutron radiography, neutron activation analysis and radiation physics research. Its location in a post irradiation examination facility enables neutron radiography of radioactive objects like reactor fuel and other components discharged from the neighboring FBTR, without the need for transporting radioactive material through the common domain of the Centre. KAMINI was commissioned in 1996 and attained its full power operation in 1997. The experimental facilities at KAMINI are presently being used for neutron radiography and activation analysis.

The KAMINI reactor system consists of ^{233}U/aluminium alloy plate fuelled core with beryllium oxide encased in zircaloy as reflector. Demineralized light water acts as moderator, coolant and shield. Operation and control of this reactor is done by cadmium safety control plates, which also provide the emergency shutdown. Various neutronic and health physics equipment are provided to monitor the functioning of the reactor. The entire reactor system is housed in a stainless steel tank, which is surrounded by high-density concrete biological shield.

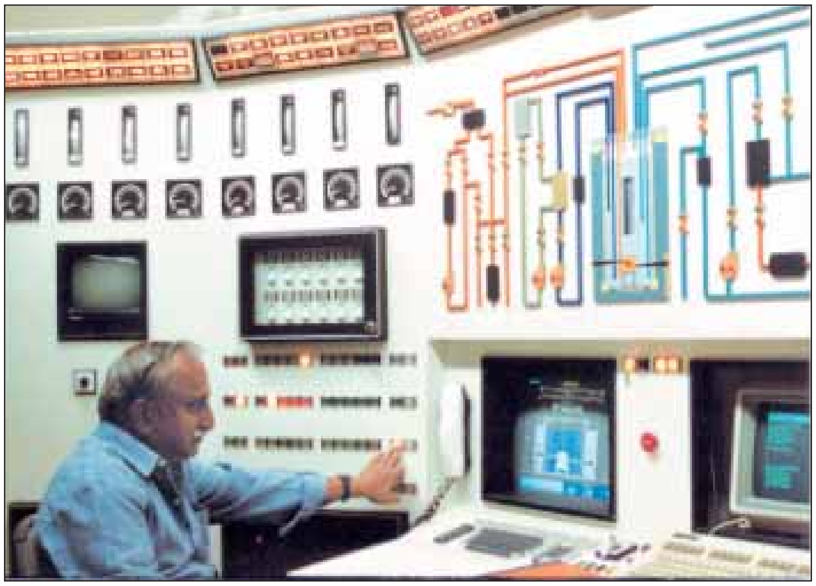
The average neutron flux in the core is 10^{12} n cm^{-2} s^{-1} and at the radiography site it is 10^{6} to 10^{7} n cm^{-2} s^{-1}. The flux at the sample irradiation site is 10^{10} to 10^{11} n cm^{-2} s^{-1}. The development of the components and systems used for building KAMINI involved a large number of agencies and scientists.

The experimental facilities provided in the reactor include neutron beam tubes, sample irradiation tubes, and a pneumatic sample transfer system for sending samples for activation analysis to a location adjacent to the core of the reactor and retrieve them after desired irradiation time. One of the beams is designed for radiography of radioactive objects. This section of the facility has been provided with a special tubular rig and sample drive system extending from a hot cell, where radioactive materials are examined, enabling positioning of the radioactive objects in the beam path for radiography and retrieving them into the hot cell after the operation. The beam can also be used for radiography of non-radioactive objects when the radioactive objects have been retracted into the hot cell. Another beam is available exclusively for radiography of non-radioactive objects. Facilities for direct and indirect methods of neutron radiography, film processing and image enhancement as well as for activation analysis including sample preparation and counting with Ge-Li detector and PC based multi-channel analyzer are available.

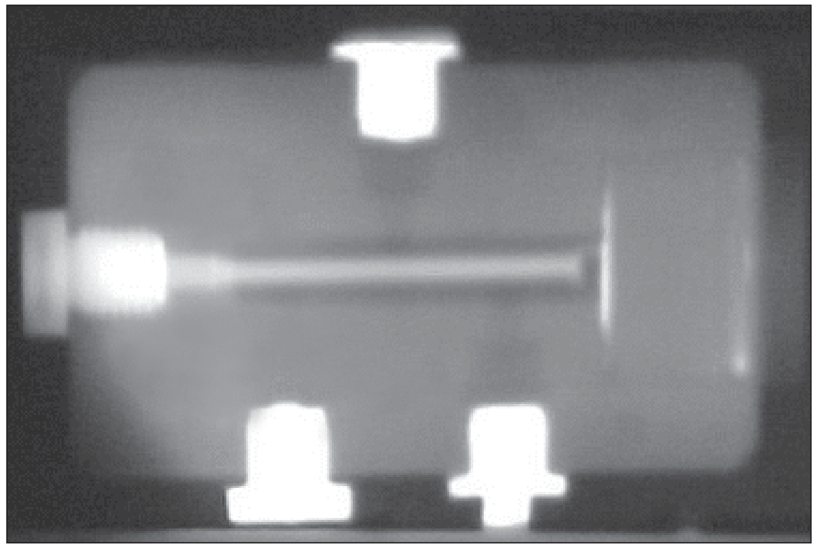
Presently, KAMINI is the only reactor in the world operating with ^{233}U fuel, which is very important for the third stage of Indian nuclear power programme.

A cubical cavity each side of which is 250 mm is available for positioning objects and equipment to be tested in the beam path for exposure to neutron beams. The beam tube locations are provided with shields and handling facilities to enable researchers to carry out the experiments without exposing themselves to radiation.

As an easily accessible and operable research reactor, KAMINI is used for testing detectors, nuclear instrumentation channels, study of composite shields and for radiation transport



View of the control panel



Neutron radiography is an invaluable tool to image structures and flaws within a solid and is a non-destructive flaw detection technique par excellence. The picture above shows a neutron radiograph of an explosive manifold

studies. The activation analysis facilities have been used for radiochemical studies and forensic science applications. These facilities are ideally used for biological/agricultural sample irradiation and material characterization applications. An MOU has been signed with Indian Space Research Organization (ISRO) and a large number of components used in satellite launch systems of ISRO are being regularly examined by neutron radiography using KAMINI.

KAMINI symbolizes a small but significant step in the context of utilization of thorium in the third stage of nuclear energy programme of India.

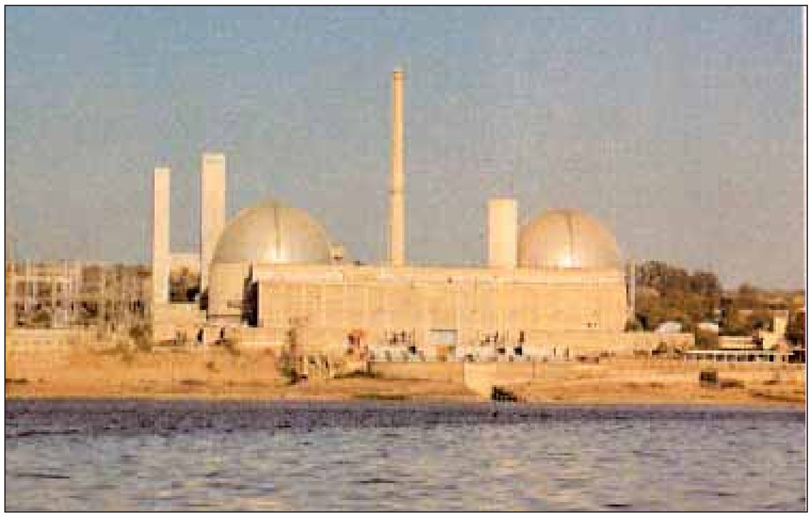
Research Reactor Projects in Progress and Plans for the coming years

Critical Facility

Required for the third stage of the nuclear power programme, a new reactor concept with thorium fuel cycle is being developed to ensure the long-term energy security of the country. This concept, called the Advanced Heavy Water Reactor (AHWR), is a technological development and demonstration prototype system for the large-scale commercial utilization of thorium in the fuel cycle. The innovative design of this reactor, requires validation of the physics design prior to its acceptance as a proven core. As a part of this validation program, a Critical Facility (CF) has been designed and is presently under construction. Detailed reactor physics experiments are planned in this facility to enable validation of the physics design parameters and computational models. These will include basic multi group cross section libraries, lattice cell parameters, pin power distributions, reaction rates, simulation of reactivity devices, core flux/power distribution etc. The CF is a tank type, heavy water moderated reactor using natural uranium metal clusters as fuel. The reactor is designed for a nominal power of 100 Watts and an average flux of 10^8 n cm^{-2} s^{-1}. For AHWR core experiments, the central nine natural uranium metal clusters will be replaced by AHWR fuel consisting of Th-Pu / Th- ^{233}U MOX clusters. The reactor design would enable flexibility for configuring the reactor core at any desired pitch. With this feature, a core with precise geometry at the desired pitch can be achieved. India has also launched a scaling-up programme for the presently operating 235 MWe PHWR designs to higher power levels of 540 MWe and 700 MWe. For accurately estimating the physics design data for these reactors, experiments would be conducted in the critical facility. Hence, the core size is made large enough for conducting experiments with 37-pin natural uranium oxide clusters. The large critical facility is also ideally suited for studying physics of loosely coupled cores such as those of 540 MWe and 700 MWe PHWRs. The CF will be a multi-purpose facility, where one can perform experiments in cold clean condition for heavy water moderated reactor systems, like AHWRs and PHWRs. It is also proposed to use this facility to model sub-critical cores with a source, to simulate an Accelerator Driven Sub-critical System (ADS), which is a new concept under study. For this purpose, a 14 MeV neutron generator would be coupled to the reactor. The facility can also be used to study light water cores in the future by replacing heavy water with light water, after the experimental campaigns for AHWR and PHWR are completed. The CF is expected to become operational by early 2005.

Multi-Purpose Research Reactor

Keeping in view the projected requirements for reactor based facilities for basic and applied research as also the projected demand for radioisotopes in the country, a new facility – a “Multi Purpose Research Reactor” (MPRR) is being designed. The MPRR is a 20 MWt pool type research reactor with a maximum thermal neutron flux of about 4.5 x 10^{14} n cm^{-2} s^{-1}, with epithermal flux of 2.5 x 10^{14} n cm^{-2} s^{-1} and fast flux of 1.0 x 10^{14} n cm^{-2} s^{-1}, at the central water hole. The reactor will be fuelled with low enriched uranium fuel in silicide form dispersed in aluminium matrix. The reactor will use de-mineralized light water as coolant and moderator. An annular heavy water tank providing a D_2O layer of 600 mm will surround the core to extend the usable neutron flux, far beyond the core edge. A large number of irradiation positions and experimental facilities (seven beam tubes and fifteen irradiation thimbles) will be provided in the reflector tank to cater to the various needs of basic research and isotope production. The core is cooled by light water in a closed loop, which in turn will be cooled by a secondary cooling system rejecting the heat to the atmosphere through a cooling tower.



Panoramic view of Rajasthan Atomic Power Station

One of the hallmarks of the Department of Atomic Energy (DAE) has been the meticulous planning marked by foresight, coherence and continuity. Indigenous capability to design, construct, commission and operate nuclear power plants have been developed alongside the necessary infrastructure for the entire process from mining and processing of uranium to spent fuel reprocessing, radioactive waste management, and production of heavy water.

India has thus become one of the few countries in the world that has acquired expertise in the entire range of nuclear fuel cycle activities. As these activities were of special nature and did not form a part of the general industrial domain of the country, DAE established facilities for carrying them out in a proper phased manner. Establishment of industrial infrastructure for nuclear fuel cycle activities was backed by the strong R&D base of DAE. This infrastructure building activity was started well before launching the nuclear power programme. In addition, significant efforts were put in by DAE to develop indigenous manufacturing capability to make various equipment/components conforming to stringent quality standards of the nuclear power programme. In retrospect, this policy of self-reliance has paid rich dividends in the form of negating the effects of restrictive trade barriers and technology denials, prevalent globally in this field. This approach has also resulted in technological spin-offs for the Indian industry.

Studies on nuclear power reactors were initiated in the 1950s. At that time only boiling water and gas cooled power reactors were in operation in USA and UK respectively. Since these power reactors would need to be imported, a group was formed in DAE to prepare tender documents for inviting global tenders for supply of these reactors. Though heavy water moderated and natural uranium fuelled reactors were the first choice, during the initial years these were not considered, since such reactors were then not in commercial operation elsewhere. This indicates the cautious approach taken by the department for acquiring and inducting only tested and established technologies.

The work on the first nuclear power project was commenced at Tarapur, Maharashtra (TAPS-1 & 2) in 1964 and the reactor type chosen was Boiling Water Reactor (BWR). The work on the second nuclear power project at Rawatbhata, Rajasthan (RAPS) was also simultaneously started in 1964, based on Pressurised Heavy Water Reactor (PHWR) technology.

These two projects had separate offices headed by the respective project administrators, considering the different nature of their construction strategy. TAPS-1&2 were turnkey projects from M/s General Electric (GE), USA, which were to be handed over on completion to DAE for operation and maintenance. For RAPS, Atomic Energy Canada Limited (AECL), was responsible for the design and supply of the plant, while, DAE undertook the responsibility for construction, commissioning and operation, under AECL guidance.

> *"...No power is costlier than no power..."*
>
> ***- Homi J. Bhabha***

The Power Projects Engineering Division (PPED), with headquarters in Mumbai, was constituted in 1967, when DAE decided to set up Madras Atomic Power Project with full responsibility for design, construction, manufacture, commissioning and operation of this reactor using in-house capability for nuclear design and Indian engineering consultants for the balance of plant. PPED was entrusted with the responsibility of implementing the Pressurized Heavy Water Reactor (PHWR) programme. A number of engineers and scientists, who were working at

BARC on reactor design, operation and related fields were transferred to PPED. Atomic Power Authority (APA) was formed as a constituent unit of DAE in 1970, after TAPS-1 and 2 were commissioned in 1969. APA was entrusted with the responsibility of operating and maintaining TAPS-1 and 2. In July 1979, the responsibilities of APA were transferred to PPED. PPED was thus entrusted with the responsibility for design, engineering, procurement, construction, commissioning, operation and maintenance of all nuclear power plants.

Birth and Growth of NPCIL

With construction of more PHWRs being undertaken at Narora and Kakrapar and expansion of the nuclear power programme, PPED was rechristened as Nuclear Power Board (NPB) on August 17, 1984 and given wider powers for decision making. The Nuclear Power Board metamorphosed into Nuclear Power Corporation of India Limited (NPCIL), a fully owned company of the Government of India, Department of Atomic Energy and it started functioning from 17th September 1987. Prior to this, the Atomic Energy Act was amended to enable the formation of NPCIL. This was essential for creating a framework for faster decision-making and also for tapping funds from the capital market for the proposed nuclear power programme.

Significant efforts have been made to achieve improved effectiveness, public awareness, media interaction and transparency, enhanced involvement of individuals, improved capabilities for interactions with outside organizations, better time management, creation of highly motivated teams with determination to achieve, better human resource management, modern and "tailor made" methods of management, and improved personnel assessment system.

NPCIL has grown from a modest beginning with a subscribed share capital of about Rs.967 crores in the year 1987-88, to the level of about Rs. 8,632 crores as on 31st December 2003. The capital investment in the company has grown from about Rs. 1,364 crores in 1987-88 to about Rs. 14,542 crores as on 31st March 2003. Presently, NPCIL is a profit making company with a credit rating of AAA. The profits are achieved despite charging tariffs comparable to those of other nearby sources of energy and in some cases, like Tarapur, lower than any other source in the vicinity.

"... The Nuclear Power Corporation of India Ltd. (NPCIL) is unique in the world in having built, under one roof, world- class expertise in the areas of planning, feasibility studies, design, construction, commissioning, operations, maintenance and rehabilitation of nuclear power plants. Several of NPCIL operating plants have been judged to be amongst the best in the world, in terms of safety, performance and environmental impact.

Indian industry has responded magnificently to the technical challenges of the nuclear power programme. It has enabled India to achieve a very high degree of self-reliance in this vital sector.

With sustained efforts put in the last 50 years, the Indian nuclear power programme has grown to be one of the largest and most comprehensive programmes in the world. It provides the Indian people energy security and the choice of an optimal mix of power sources. Nuclear power will continue to play a vital developmental role in the life of the Indian Nation…"

- S.K.Jain,
Chairman & Managing Director,
NPCIL

Human Resource Development

During the early years, BARC formed the core of human resource for the power programme. The training school of BARC, which started functioning in the year 1957, has been the source of trained engineers and scientists. A small core of faculty manages the school and majority of teaching staff is drawn from among the working scientists and engineers at BARC and other units of DAE. After completion of the training course, the trainees are posted to various units of DAE. During late 1960s and early 1970s, the focus of HRD remained on learning and acquiring expertise in the technology required for operation of the nuclear power reactors. Initially, personnel were trained in USA for operation and maintenance of TAPS-1 and 2 and in Canada for RAPS-1. A group of engineers and scientists from BARC were also deputed to Canada for participating in the design of RAPS-1 and 2 and training in quality assurance. Over the years, separate induction and training programmes have been developed by NPCIL to meet its specific needs. A Nuclear Training Centre (NTC) was set up at Rawatbhata, Rajasthan, in 1969 to train and license staff for operation and maintenance of the plant.

In the 1980s more nuclear power stations started commercial operations. Recruitment and training programmes were also enlarged with more NTCs to meet the increased requirement. In 1987, when NPCIL was formed, a new set of initiatives were taken to implement self-awareness and cross functional training to senior executives, team training at all stations to enhance team performance, up-gradation of the training of the trainers, training in root cause analysis, training in human performance fundamentals and good practices. Full scope operator-training simulator has been developed indigenously for 220 MWe PHWR units. Also noteworthy is the development of a simulator for the 540 MWe PHWR units at Tarapur.

Progress on the Power Reactor Front

India figured on the nuclear map of the world in 1969, when two Boiling Water Reactors (BWRs), TAPS-1 and TAPS-2, were commissioned at Tarapur Atomic Power station (TAPS). The main objectives of setting up these units were to prove techno-economic viability of nuclear power and to obtain experience in operation and maintenance of nuclear power plants and to demonstrate technical viability of operating the nuclear power stations in the Indian regional grid system. For Tarapur, all the components of the power plant and nuclear fuel were imported and the role of Indian industries was limited to construction, erection and service contracts.

PHWRs were chosen for the first stage of the three stage nuclear power programme, as these are efficient producers of plutonium required for the second stage and they could be made indigenous to a great extent with the participation of the then Indian Industry. As a part of main thrust of developing PHWR designs, building of the second nuclear power station Rajasthan Atomic Power Station (RAPS) consisting of two units, RAPS-1 and RAPS-2, was taken up as a joint Indo-Canadian venture in the late sixties, at Rawatbhata, Rajasthan. Canada furnished nuclear designs and also supplied all the main equipment for the first unit. India retained responsibility for construction, installation and commissioning activities. For the second unit (RAPS-2), manufacture of components was taken up in India and the import content was reduced considerably. Canada withdrew support for the plant in 1974 due to the Pokhran tests and Indian engineers carried out remainder of the design work, construction and commissioning of the unit.

> *"... The favourable development of the technology and economics of nuclear power made the Government of India decide over a year ago to go ahead with the second nuclear power station with an electrical generating capacity of 200 MWe in one reactor at Rana Pratap Sagar in the state of Rajasthan..."*
>
> ***- H.J. Bhabha***
>
> *7th General Conference, IAEA, 1962*

From the third nuclear power plant, Madras Atomic Power Station (MAPS) consisting of two units, MAPS-1 and MAPS-2, at Kalpakkam, Tamilnadu, India had started carrying out all facets of the work. A notable number of design changes were incorporated at MAPS.

Improved reactor designs were developed to keep abreast with evolutionary changes taking place worldwide and to meet new safety criteria. Since there was a requirement to establish PHWRs at various sites in different electricity regions of the country, it was also necessary to evolve designs that met environmental and seismic criteria. In order to commence construction of PHWRs at a number of sites more or less simultaneously, it became necessary to evolve a standardized design. Improvements were also incorporated to enhance reliability of operation and productivity, reduce costs and attain better capacity factors. The first two units of PHWR using indigenously developed standardized 220 MWe plant design were set up at the Narora Atomic Power Station (NAPS). Subsequently, three more atomic power stations of the standardized and optimised design, with capacity of 2 x 220 MWe each, were built and commissioned at Kakrapar, Kaiga and Rawatbhata using indigenous technology. Successful commissioning of these stations has clearly demonstrated that India has mastered the technology involved and is fully capable of utilizing the same in the commercial domain. This also established nuclear power as a safe, environmentally benign and economically viable source of power that offers energy security for the country.

In order to realize economies of scale, designs for 540 MWe PHWR have been evolved and construction of two such units was taken up at Tarapur. Recently, one of the two units has been made operational. In order to further optimize the cost of electricity production by exploiting economy of scale, NPCIL has initiated design modifications in 540 MWe PHWRs to upgrade the rated capacity to 700 MWe.

To accelerate growth of nuclear power capacity, 2 X 1000 MWe VVER reactors are being constructed at Kudankulam with the cooperation of the Russian Federation. A beginning has been made to launch the second stage of the nuclear power programme and work has been initiated on construction of a Fast Breeder Reactor (FBR) plant of 500 MWe capacity at Kalpakkam. An independent corporation called BHAVINI has been set-up for this purpose. This new venture exemplifies the synergy between the research and development strengths of Indira Gandhi Centre for Atomic Research and the project planning and construction expertise of NPCIL.

India's nuclear power programme has now matured into a safe and economical option for meeting the country's power demands, and is poised for an accelerated pace of growth. Presently, 15 reactors are under operation and 8 reactors are under construction. NPCIL is alive to the need for developing and employing technologies that will further enhance safety, availability and cost competitiveness of nuclear power.

Tarapur Atomic Power Station – The First Nuclear Power Project

Tarapur in Maharashtra was selected for building the first nuclear power station. A sleepy hamlet, Tarapur, located on the West coast of India was to be transformed by this significant

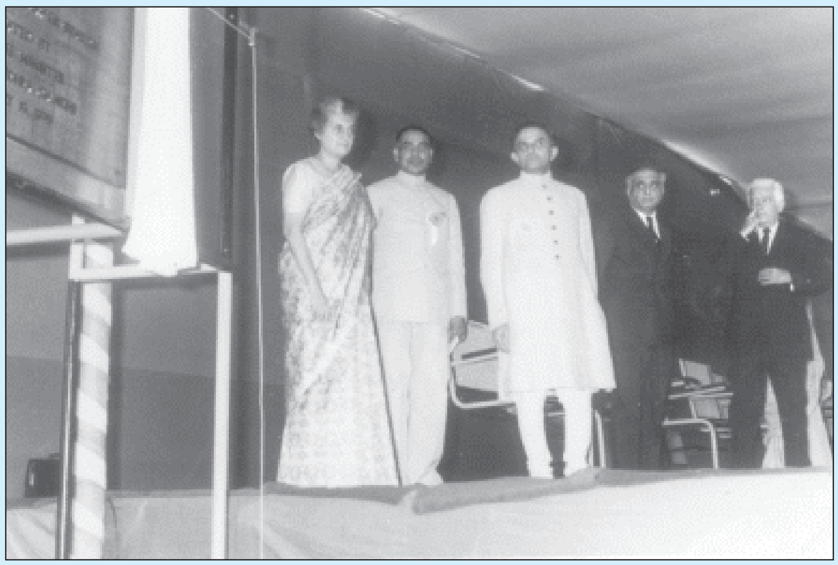
Tarapur power reactors - TAPS 1&2

decision. Construction activity started a Tarapur in October 1964. After intensive testing to demonstrate the safe and reliable operation of all the plant systems and equipment, both of the Tarapur Reactors (2 X 210 MWe) were commissioned in October 1969.

It was a momentous occasion when the first nuclear power plant in the country began generating electrical power and the electricity started flowing to the states of Maharashtra and Gujarat. Thus, a new chapter in the country's nuclear power programme was heralded. The then Prime Minister Ms. Indira Gandhi dedicated the plant to the nation on 19th January 1970. The station has been operating since then. Tarapur has today become an industrial hub with 600 odd industries employing nearly 20,000 workers from all parts of India, thanks to the cheap power supplied to them. The Tarapur units are operating safely and reliably and continue to be viable. The safety aspects like environmental releases, personnel radiation exposures etc., are continuously being improved based on international standards. Several upgradations are being carried out. These life cycle management measures and encouraging performance are likely to result in extended lifespan of the units.

View of Tarapur construction site in 1964

In the unique Indian context, primitive means of transportation did not deter the development of advanced nuclear technologies - Equipment transport to TAPS site in the early sixties

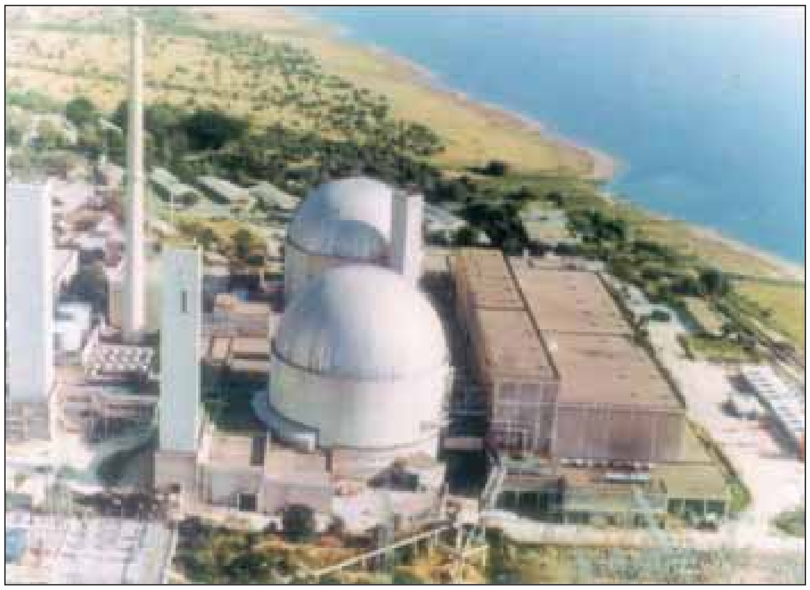


Prime Minister Indira Gandhi dedicating TAPS to the Nation in 1970. Also seen is Vikram Sarabhai

The Early PHWR Projects

Rajasthan Atomic Power Station (RAPS-1 and 2)

The first PHWR project is located at Rawatbhata, Rajasthan. The reactors (2 X 220 MWe) are located on the right bank of Rana Pratap Sagar Lake on Chambal River. The construction of RAPS-1 was commenced on December 1, 1964. Most of the major nuclear and conventional equipment were imported from Canada. However, certain components like thermal shield plates and fuelling machine vault doors were manufactured at BARC. In addition, some of the conventional equipments like compressors, chillers, demineralised water plant, and 250 MVA transformers were made in India. RAPS-1 achieved criticality on 11th August 1972 and started feeding power to the grid on 30th November 1972.



Rajasthan Atomic Power Station - 1 & 2 Plants

For RAPS-2, concerted efforts were made to get major equipment fabricated in India. Critical components like steam generators, calandria, end shields, shield tank, dump tank and Control & Instrumentation (C&I) systems were made indigenously. Other important components taken up for indigenous manufacture were the fuelling machines, shielding plugs and sealing plugs, which required precision machining and special heat treatment. This was the period when heavy engineering industries in the country were either being set up or being developed. The group of engineers located at Bombay (Mumbai) had the challenging task of developing indigenous manufacturers for the equipment. The commissioning of RAPS-2 was entirely carried out by the Indian team. The indigenous content of about 55% in unit-1 increased to 75% in unit-2. RAPS-2 achieved criticality on 8th October 1980 and started feeding power to the grid on 1st November 1980.

The setting up of the RAPS-1 and 2 plants was an important phase when India moved from the stage of obtaining nuclear power projects on a turnkey basis to the stage of entering into collaborative ventures with significant degree of indigenous participation. This effort laid the foundation for the sound indigenous capability, which the country has achieved now. The Rajasthan and Tarapur Units proved that India is capable of absorbing advanced technologies quickly and exhaustively to the extent that it can effect improvements and expand on the same.

Madras Atomic Power Station (MAPS-1 and 2)

The second set of PHWRs, MAPS-1 and 2, were built at Kalpakkam on the east coast about 65 km south of Madras (Chennai). The project marked a milestone in the nuclear power programme whereby total self-reliance for execution of the project including design, engineering, construction, commissioning and operation was achieved. A twin unit PHWR station of 2 x 220 MWe capacity was set up.

Madras Atomic Power Station

Though the basic features of the reactor systems were similar to those of Rajasthan reactors, many engineering design changes were introduced, based on the lessons learnt at RAPS. The most important change was replacement of the dousing tank provided at RAPS with suppression pool in the basement of the reactor. The suppression pool is used for limiting peak pressure in the containment building following a loss of coolant accident and has the advantage of being a passive system. India is the first country in the world to adopt the pressure suppression pool concept for a PHWR. This concept is an improvement over the vacuum building concept used in some other countries. Prestressed concrete was chosen for inner containment design and a rubble masonary outer wall was built around to serve as partial double containment. A submarine tunnel, the first of its kind in India, was provided to bring seawater to the plant from half a kilometer offshore. An indoor switchyard was provided in view of the problems of salt deposition faced at TAPS-1 and 2. The material of end shields was changed to austenitic stainless steel for the second unit to avoid possible cracks developing in tube sheet. The construction of MAPS-1 and 2 involved significant indigenous efforts in the area of nuclear and conventional plant design, indigenous manufacture of components and deploying sophisticated construction technology. The development efforts, though time consuming, yielded remarkable success. MAPS-1 achieved criticality on 2nd July 1983, and was synchronized to the grid on 23rd July 1983 by the then Prime Minister Ms. Indira Gandhi. MAPS-2 unit achieved criticality on 12th August 1985, and was synchronized to the grid on 20th September 1985. This unit was dedicated to the nation by the then Prime Minister Rajiv Gandhi in December 1985.

The setting up of MAPS units accelerated the process of self-reliance.

PHWRs of Standardized Design

Narora Atomic Power Station (NAPS-1 and 2)

By now sufficient indigenous capability was built to take up improvements of significant nature in the PHWR designs for the NAPS-1 and 2. The lessons learnt from Three Mile Island (USA) and Chernobyl (Ukraine) events were incorporated in the design of NAPS. These plants meet the highest national and international standards of safety and performance.

Narora is located on the banks of Ganga in Uttar Pradesh.

Narora Atomic Power Station

The reactor plants at Narora incorporated many new concepts and improved designs as compared to the earlier plants. Since the plant is located in a seismic zone, special design efforts were put in to fully meet seismic requirements. The reactors of NAPS are the first units built to conform to the standardized 220 MWe PHWR design, which was developed indigenously. Further improvements saw provision of full double containment. NAPS-1&2 are the first reactors in the world to have such a provision. Indigenous manufacture of critical nuclear equipment took considerable time as many “first of its kind” equipment had to be developed .

The first unit of NAPS achieved criticality on 12th March 1989 and was synchronized to the grid on 29th July 1989. The second unit achieved criticality on 24th October 1991 and was synchronized to the grid on 5th January 1992. Since then, the units are functioning quite satisfactorily.

Kakrapar Atomic Power Station (KAPS-1 and 2)

The Kakrapar plant, comprising of two PHWR units of 220 MWe each, is located on the left bank canal taking off from Kakrapar weir on Tapi River. The plant follows the layout of Narora with respect to reactor building. It has incorporated all the safety features of Narora Plant. However, in contrast to the alluvial substrata at Narora plant, the site at KAPS has rocky substrata. This called for redesigning of the systems, components and structures of the plant to account for the different seismicity levels. These design changes were fully

Kakrapar Atomic Power Station

tested during the Gujarat earthquake in 2001, when, the KAPS plants continued to operate safely during and after the cataclysmic event.

The construction began in December 1983. The first pour of reactor concrete commenced in December 1984. There was progressive improvement by way of reduction of indigenous

Tested by Nature !

The KAPS plants continued to operate safely after the Gujarat earthquake of 2001. This is a testimony to the safe design of the plant.

manufacturing time cycles of major equipment. The construction period of the project was shortened by deploying heavy-duty crane for handling components and use of advanced welding techniques. The first unit achieved criticality on 3rd September 1992 and was synchronized to the grid on 24th November 1992. The second unit achieved criticality on 8th January 1995 and was synchronized to the grid on 30th March 1995. These units are performing well for the past several years. Kakrapar project represented a phase, where a trend of consolidation of the technology and reduction in manufacturing time cycles of major components emerged as a repetitive success story.

Kaiga Atomic Power Station (KGS-1 and 2)

Kaiga Atomic Power Station, comprising two PHWR units of 220 MWe each, is located near village Kaiga on the bank of Kali River, in Karnataka. The plant draws its cooling water from Kadra reservoir. The design of the plant is of the twin unit module basically followed in the standardized version of the PHWRs built at NAPS and KAPS. The layout of plant was made more compact and full double containment was provided.

The first pour of concrete for Kaiga-1 commenced on September 1989. In May 1994, due to de-lamination of portions of the inner containment of dome in Kaiga-1, further construction activities at Kaiga were put on hold. After evaluation and analysis of the incident, redesign including change in

Kaiga Atomic Power Station

construction methodology, was worked out. As re-engineering of Kaiga-1 dome was expected to take more time, the priority of construction was changed to complete Kaiga-2 first, to be followed by Kaiga-1. Construction resumed initially at Kaiga-2 and entire dome concreting was completed in March 1998. Special high performance concrete of grade M60 was successfully developed as part of re-engineering. Speeding up the downstream activities absorbed a significant portion of the time lost due to the re-engineering of the dome. The time schedule for the construction and commissioning activities was compressed by resorting to parallel working on construction and commissioning activities round the clock and adoption of innovative measures, like provision of complete structural steel cover below dome to facilitate uninterrupted progress of mechanical works in parallel with the dome construction.

Manufacture of the nuclear equipment took a significantly shorter time, as no learning time was required. Kaiga-2 achieved criticality on 24th September 1999 and was synchronized to the grid on 2nd December 1999. Kaiga-1 achieved criticality on 26th September 2000, and was synchronized to the grid on 12th October 2000 within a period of 16 days. Prime Minister Atal Bihari Vajpayee dedicated the Kaiga power station to the nation on 5th March 2000.

Rajasthan Atomic Power Station (RAPS–3 and 4)

With two units RAPS-1 and 2 already operating and the infrastructure already present, decision was taken to set up RAPS-3 and 4 at the same site. RAPS-3 and 4 is a twin unit project with 2 x 220 MWe PHWRs.

Rajasthan Atomic Power Station - 3 and 4 plants

Construction work was started in July 1988. Ordering of equipment and manufacture/delivery also progressed in parallel. Along with civil works, reactor erection, nuclear piping, mechanical, electrical, instrumentation and other works also progressed as per schedule. Due to delamination of Kaiga-1 dome in May 1994, construction activities of RAPS-3 and 4 were also put on hold. The holds were progressively cleared from May 1997 to May 1998. In order to speed up, parallel working methods were chalked out to enable equipment erection while dome concreting was in progress. The entire project management was restructured. Nuclear piping, cable laying and other activities were expedited. Commissioning activities were significantly speeded up by innovative methods, close co-ordination and monitoring. This resulted in cutting down the commissioning time. RAPS-3 achieved criticality on 24th December 1999 and was synchronized to the grid on 10th March 2000. The experience of implementing new concepts in unit-3 provided valuable inputs to cut down commissioning time of RAPS-4. Hot conditioning of primary heat transport system in RAPS-4 was completed within 56 days of hydrostatic test. RAPS-4 achieved criticality on 3rd November 2000 and was synchronized with the grid on 17th November 2000 within a period of 14 days, which was a significant achievement and record. Prime Minister Atal Bihari Vajpayee dedicated the Station to the nation on 18th March 2001.

The year 2000 thus saw commencement of commercial operations of four 220 MWe PHWR units adding a total of 880 MWe nuclear capacity in the country. Setting up of these units has demonstrated NPCIL's capability to shorten the commissioning time periods of projects significantly and to handle a large number of projects concurrently.

Other PHWRs Under Construction

Five PWR units are presently under construction. These are: 540 MWe unit at Tarapur (TAPP-3), 2 x 220 MWe at Kaiga (Kaiga-3 and 4) and 2 x 220 MWe at Rajasthan (RAPP-5 and 6). The project implementation strategy has been modified by adoption of large supply-cum-erection packages, and where Indian industry is capable, even Engineering, Procurement & Construction (EPC) to enable speedy completion of the projects. Recently, TAPS-4 unit has gone critical.

Tarapur Atomic Power Project - Units-3 (540 MWe)

In 1984, DAE envisaged setting up a number of 500 MWe PHWR units and this was brought out in the planned nuclear power profile. A dedicated group was then formed for the design and development of 500 MWe PHWR units.

It was decided to locate two units of 500 MWe PHWR at Tarapur so as to avoid a power vacuum at this site as and when the existing TAPS-1&2 come to the end of their operating life and to utilize the existing infrastructure at the site. The plant is basically a scaled-up version of the 220 MWe PHWR but incorporated many improvements based on operating experience of 220 MWe units. It also included many features specific to a large size unit. Some of the highlights of the design are: 37 element fuel bundle design; two identical valve-less primary coolant system loops; provision of a pressuriser for pressure control of primary coolant system; introduction of liquid zone control system for zonal flux control; two diverse fast acting shutdown systems of adequate worth; in-core flux monitors to monitor the status of the larger core and significant improvements in fuel handling system for reliable on-power fuelling of the larger core. A number of development works, in the areas of shutdown systems, liquid zone control system and coolant channel assemblies were carried out. Based on an in-depth evaluation of the design, the plant capacity has been up-rated to 2x540 MWe.

Construction work at Tarapur Units-3 and 4

India's largest and first 540 MWe nuclear power plant (TAPP-4) achieved criticality on the 6th of March 2005. This completely indigenous nuclear power plant, designed and built by NPCIL, is the 15th nuclear power reactor in the country. The vibrant Indian industry has also contributed tremendously to the making of high quality nuclear standard equipment and construction of the plant. This time of construction compares well with international benchmarks. TAPP-3 is also expected to go critical soon. Successful implementation of the project has demonstrated Indian capability to scale up the unit size of PHWRs and to build nuclear power projects in shorter time frames.

The "first-of-its-kind" Unit-4 of Tarapur (TAPP-4) has achieved criticality in just five years from the date of first pour of concrete and seven months ahead of schedule. This is the shortest time taken to build any PHWR in India.

Kaiga Atomic Power Project - Units 3 and 4 (2 x 220 MWe)

Layout of the Kaiga plant, including water intake arrangement, was planned for four 220 MWe units, right at the beginning. Therefore, setting up of Kaiga-3 and 4 did not call for additional land or displacement of population. The first pour of concrete for the project was made on 30th March 2002. Ordering of major equipment and supply-cum-erection packages has been completed and manufacture is in different stages of progress. Completion of Kaiga-3 unit is scheduled by the end of the X Plan period (March 2007), which is five years from the first pour of concrete. Kaiga-4 unit is scheduled for completion within a period of 6 months thereafter.

Rajasthan Atomic Power Project – Units 5 and 6 (2 x 220 MWe)

At Rawatbhata, Rajasthan, four units (RAPS-1 to 4) are in operation. This site has potential and clearance for additional units. In view of the availability of infrastructure at the site and the fact that construction work could commence quickly, it was decided to set up, two more 220 MWe PHWRs at the site. The project did not call for any additional land or displacement of population. The first pour of concrete was made in September

Construction in progress at RAPP-5 reactor building

2002. Major equipment and supply-cum-erection packages have been ordered and manufacture is in different stages of progress. The units 5 and 6 are scheduled to commence commercial operation in August 2007 and February 2008.

PHWRs on the anvil

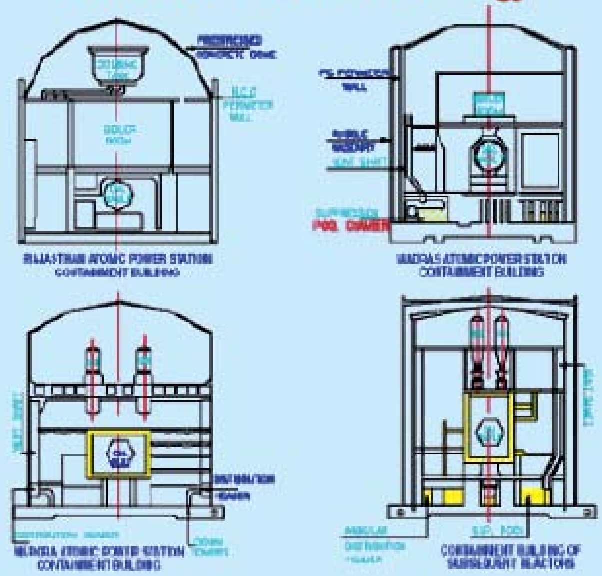
The Indian nuclear power programme has attained a high degree of maturity with a large number of 220 MWe PHWR units in operation. Presently design work pertaining to further upscaling of 540 MWe units to 700 MWe units has been taken up with a view to gain significant cost advantages. The design of 700 MWe units is essentially the same as that of TAPP-3 and 4 units except that partial boiling of heavy water coolant in the fuel channels has been allowed. The design incorporates good features of the 220 MWe and 540 MWe plants. It is planned that the next power projects to be undertaken will comprise of two units of 700 MWe each.

Evolution of the Design of Indian PHWRs

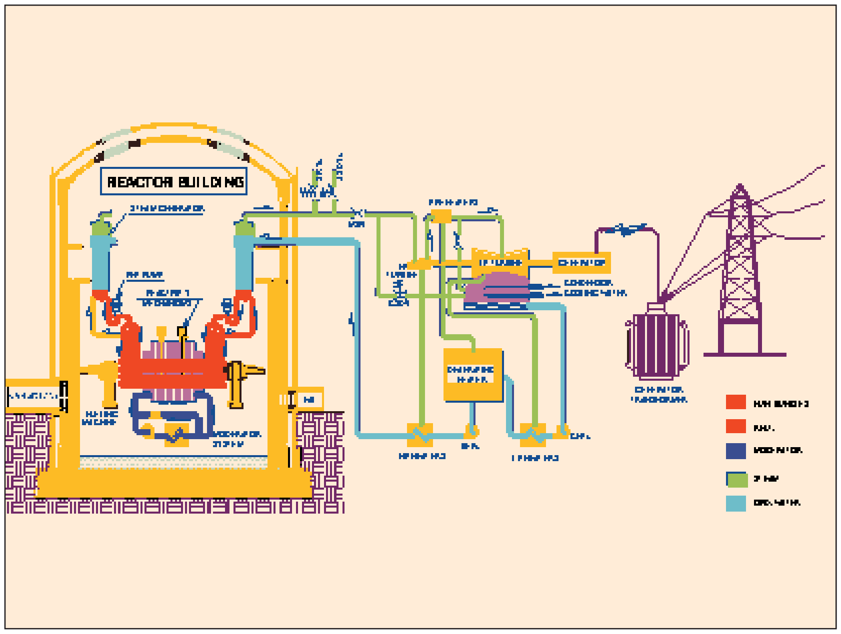
The PHWR is a pressure tube type reactor using heavy water moderator, heavy water coolant and natural uranium dioxide fuel. The reactor consists primarily of calandria, a horizontal cylindrical vessel. It is penetrated by a large number of zircaloy pressure tubes arranged in a square lattice. These pressure tubes, also referred to as coolant channels, contain the fuel and hot high-pressure heavy water coolant. End shields are integral parts of the calandria and are provided at each end of the calandria to attenuate radiation emerging from the reactor, thereby permitting access to the fuelling machine vaults when the reactor is shut down. Each pressure tube is isolated from the cold heavy water moderator present in calandria by a concentric zircaloy calandria tube. During the fission process in which heavy atoms of uranium-235 are split into lighter atoms, a large quantity of heat is produced and neutrons with high energy are liberated. Moderator brings down the energy of these neutrons. These low energy neutrons cause further fission reaction in the fuel and thus the chain reaction continues. During moderation, the heavy water gets heated up. This heat is removed by circulating the moderator through heat exchangers using moderator pumps.

Heavy water in the primary heat transport system provides means for transferring heat produced in the fuel to the steam generators in which steam to run the turbine is produced from ordinary water. On-power fuel-handling system permits the reactor to be fuelled even at full power.

Evolution of PHWR Technology



Fuel
Wire wrap to split spacer graphite coated
19 elements for 220MW to 37element for 500MW
Pressure Tube Material
Zircaloy – 2 to Zr – 2.5% Nb
Reactor Shut – down System
Moderator dumping to Shut off rods/ Liquid poison tube system/ liquid poison addition
End shield/s
Carbon steel slab type to stainless steel, ball filled
PHT
Single loop, 8 pumps/SG to Single loop, 4 pumps/SG to Two loops, 4 pumps/SG



Schematic digram of Indian PHWR

Salient design improvements in Narora reactor design, as compared to earlier units were:

- Two independent, fast acting, shut down systems based on different operating principles.
- Augmented emergency core cooling system, with high pressure injection followed by long term recirculation.
- Integral calandria and end shields.
- Calandria vault filled with light water.
- Elimination of moderator dumping and pump-up system.
- Full double containment with modified vapour suppression pool. The annular space between the two containments is maintained at a pressure less than the atmospheric pressure to prevent release of radioactivity into the ambient atmosphere at the ground level.
- Significant reduction of argon-41 release, due to water filled calandria vault.

Plant Layout and Civil Structures

The plant layout has evolved over the years based on safety requirements, easy accessibility of components, ease of operation and maintenance as also compactness to reduce total requirements of pipes and cables, under ground tunnels and land. Major thrust is given to minimize radiation exposure to plant personnel and general public. Attempts are being made, for the future projects, to provide a common foundation for a group of adjacent safety related buildings. This will minimize the problems of foundation uplift under various load combinations especially under seismic environment and as a result will bring economy in the civil design.

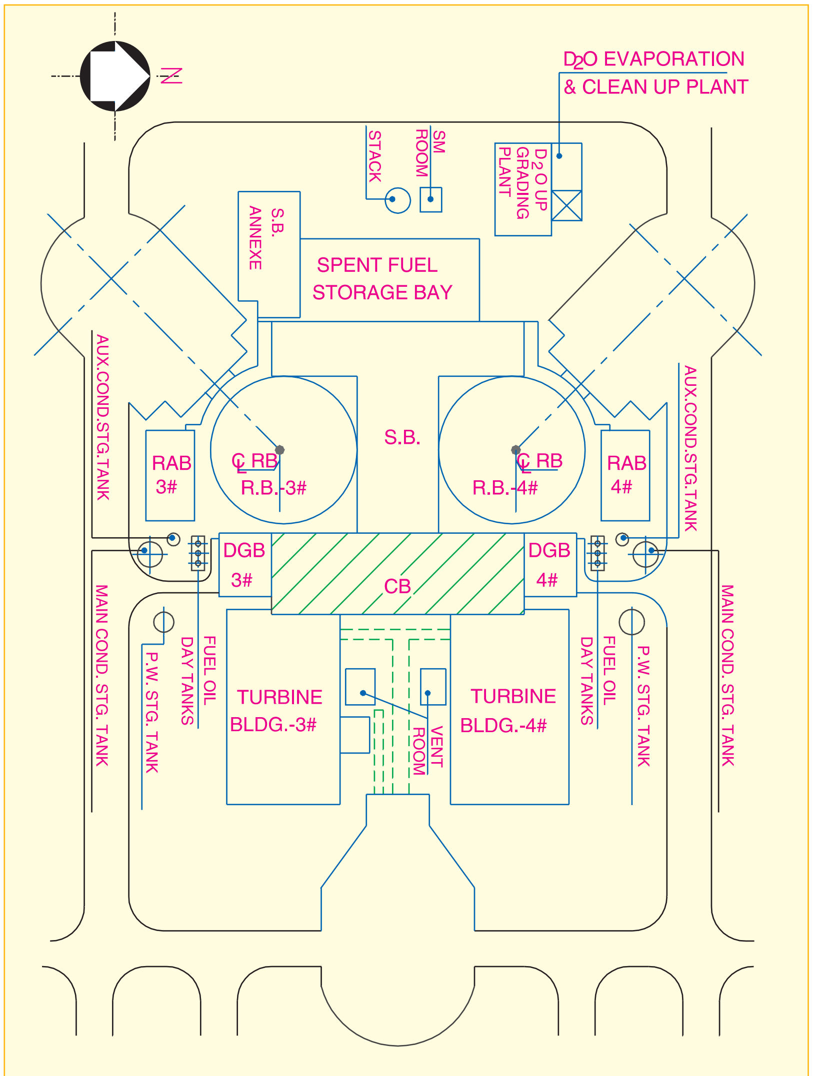
The containment structure, the building which houses the reactor, the primary coolant and moderator systems and other systems related to steam generation, is the most important structure of the plant. The containment structure is required to contain the radioactivity release in the event of any postulated design basis accident (DBA) so that the level of radiation released to the external environment is within acceptable limits. The design of containment structure is the most challenging job for the structural engineers in the nuclear industry.

The containment of the first PHWR in the country, at RAPS, was of Canadian design. The containment structure is 1.2m thick reinforced concrete wall designed essentially to meet the shielding requirement. A pre-stressed concrete dome was adopted as a leak tight barrier in place of the original Canadian design in structural steel.

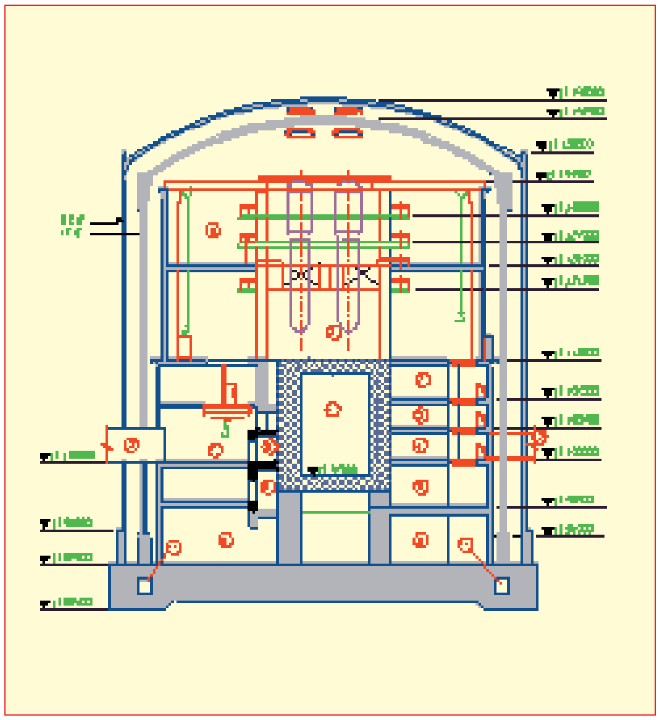
In MAPS, the entire containment with cylindrical wall and dome was constructed using pre-stressed concrete. The concept of double containment, though partially, was introduced for the first time.

The design was further improved in NAPS and KAPS and full double containment was adopted. The designs of containments of NAPS and KAPS are more or less similar except that the height of the reactor building was reduced in KAPS with the provision of openings in the dome of reactor building for erection of equipment.

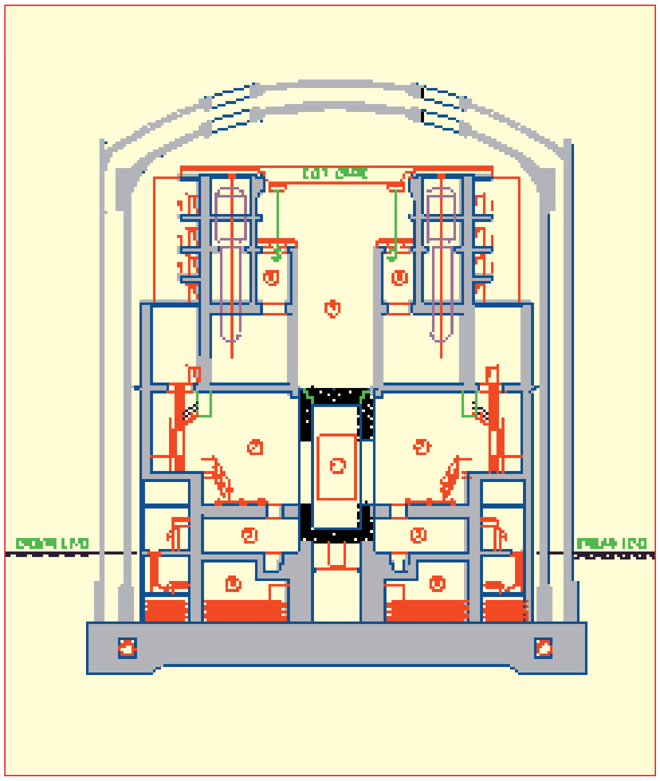
A marked improvement in the containment design philosophy was achieved with the provision of complete double containment having independent domes for both inner and outer



TYPICAL SITE LAYOUT



Typical containment system for the Indian PHWRs



Lowering of steam generator into the containment building

containment walls of Kaiga Units 1 and 2 and Rajasthan 3 &4.

The containment system adopted for 540 MWe PHWR at Tarapur (Units 3 and 4) consists of two steam generator openings in the dome as compared to four adopted for Kaiga (Units 1 and 2) and RAPS (Units 3 and 4). This has been done in order to avoid concentration of pre-stressing cable bands and to have uniform distribution of pre-stressing force on the containment structure.

Fuel for PHWRs

Indian PHWRs use natural uranium dioxide as fuel. The fuel is in the form of small cylindrical pellets, loaded in Zircaloy-4 (Zr-Sn-Fe-Cr alloy) cladding tube and hermitically sealed at both ends by welding with two end plugs. Such elements are assembled in the form of a bundle by welding them to two end plates. Zircaloy is selected as fuel bundle structural component due to its low neutron absorption characteristics and good

Fuel bundles for 220 MWe reactors

corrosion resistance. The zircaloy fuel clad for PHWRs is of collapsible type. Typically 220 MWe bundles are about half a metre in length and each bundle weighs about 16.6 kg.

These bundles are located in coolant channel (pressure tube). The fuel bundle generates heat by nuclear fission and this heat is transported to the primary coolant. On power bi-directional fuelling is carried out with the aid of two fuelling machines, one at either end of the coolant channel.

Nineteen element fuel bundles are used in the 220 MWe reactors. The fuel bundle type used in the first Indian PHWR, RAPS-1 is of 19-element wire wrap design. Based on the performance of the fuel at the different plants, changes have been incorporated in the fuel design, and manufacturing and reactor operating guidelines from time to time. During the eighties the wire wrap bundle design was updated to split spacer bundle design to avoid the possible fretting damage caused by the wires. In order to overcome fuel failures induced due to stress corrosion cracking of zircaloy, development work on graphite coating of the inside surface of the clad tube was taken up. The graphite coating technique has been adopted in fuel manufacturing process since 1989. The UO_2 pellet shape is cylindrical. The pellet shape has been updated to improve manufacturing recovery and reduce stress concentration. The present design incorporates dish on both ends with edges chamfered.

Apart from zircaloy clad natural uranium dioxide fuel bundles, aluminum clad bundles, depleted uranium dioxide as well thorium dioxide bundles were designed, developed and successfully tested in the reactors. The performance of the Indian fuel bundles is comparable to the best in the world. The 37-element fuel bundle designed for 540 MWe and 700 MWe PHWRs is an extension of the close packed 19-element fuel bundle. In this design, one more ring of 18 elements has been added to get more power per bundle.

Reactor Components

The calandria is a horizontal vessel and contains heavy water as moderator. It is penetrated by coolant tubes, which contain fuel and through which flows the primary heat transport heavy water. When the nuclear chain reaction is in progress inside the calandria, some heat is transferred from calandria to reactor vault atmosphere and vault concrete. Heat is also produced in concrete due to neutron and gamma radiations. This heat is removed by cooling systems to limit the temperature. Other penetrations in the calandria contain reactivity devices to regulate power or shutdown the reactor.

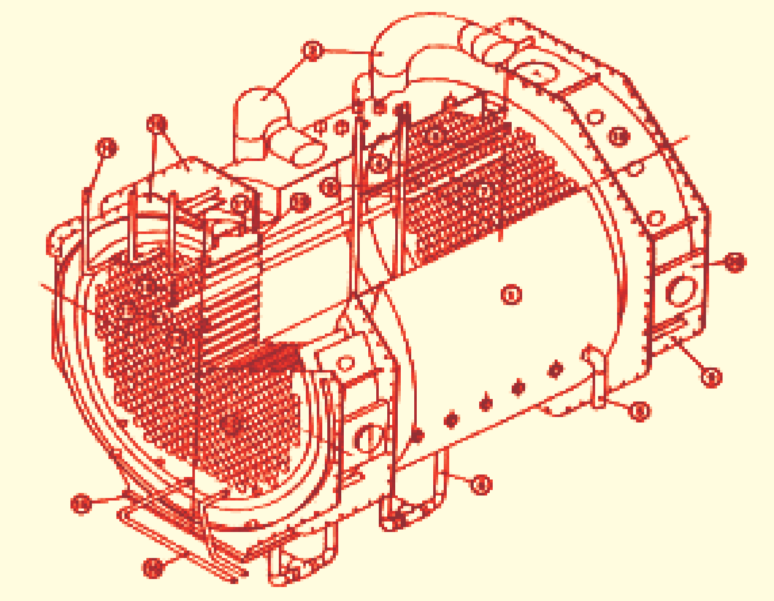
For RAPS and MAPS, calandria design is quite complex. In this, the reactor is shutdown by dumping (fast removal) of about 120 tonnes of moderator heavy water from calandria into another large capacity tank called dump tank. To restart the

Calandria being lowered into reactor building reactor, moderator has to be pumped back from dump tank into calandria, which takes considerable time. From NAPS onwards dumping of moderator for shutting down the reactor has been eliminated. Instead, two independently fast acting reactor shutdown systems have been provided which reduce the time required for restarting the reactor.

Another major reactor component is the end shield. End shields are provided at both ends of calandria and act as radiation shields. Calandria and end shields are housed in reactor vault, which is a concrete structure, lined inside with steel. RAPS-1 and 2 and MAPS-1 end shields are made of carbon steel, but from MAPS-2 onwards stainless steel end shields have been used to avoid the problem of radiation embrittlement. End shields contain water and steel, which act as shields for neutrons and gamma radiations.

From NAPS onwards the calandria and end shields are made integral and the reactor vault is filled with light water, which provides radiation shielding and also cools the reactor vault walls. This has simplified the light water system for cooling the concrete. Also since vault is filled with water, problem of activity due to Argon-41, an isotope formed in air in RAPS and MAPS vaults has been eliminated.

Calandria and end shield assembly



1 CALANDRIA SHELL
2 OVER PRESSURE RELIEF DEVICE
3 MECHANICAL SHUT OFF ROD
4 LIQUID SHUT OFF SYSTEM
5 MODERATOR INLET
6 MODERATOR OUTLET
7 CALANDRIA TUBE
8 COOLANT TUBE
9 END SHIELD
10 CARBON STEEL OCTAGONAL STRUCTURE ASSEMBLY
11 MAIN SHELL ASSEMBLY
12 TUBE SHEET FUELLING MACHINE SIDE
13 TUBE SHEET CALANDARIA SIDE
14 LATTICE TUBE
15 END SHIELD SUPPORT PLATE
16 END SHIELD COOLING PIPES
17 END FITTING ASSEMBLY
18 FEEDER PIPES
19 OUTER SHELL
20 SUPPORT LUG

Once the indigenous manufacture of components commenced, industrial infrastructure was developed and in-house facilities were also built. Technical know-how was developed jointly with industries. Special welding machines, precision machining, and heat treatment were developed. Many special non-destructive examination techniques like ultrasonic, radiography and eddy current testing were also developed for use during manufacture of reactor components. For manufacture of zircaloy components, a dedicated facility viz. Nuclear Fuel Complex (NFC) was established at Hyderabad.

Several R&D programmes were carried out to develop, generate and validate data pertaining to reactor components. Some of the important developments carried out are tri-junction and bi-junction welding for end shields, manufacturing route for coolant tubes, analysis of stress distribution around cluster of nozzles in calandria shell, stress analysis of perforated tube sheets and experimental validation of shutdown systems and generation of data. India has, over the years, acquired self-sufficiency in design, manufacture, erection, operation, in-service inspection and repair of reactor components.

Reactivity Devices

Controlling neutron population in the reactor to maintain chain reaction at the desired rate is known as reactor regulation. Apart from balancing the constituents and configuration of the reactor core, external materials are used to absorb excess neutrons to achieve this. These neutron absorbing materials are known as 'poisons'. Various poison materials used in the Indian PHWRs are steel, cobalt, light water, boron, cadmium, gadolinium etc. The materials are used in solid or liquid state.

In certain situations, due to increase in reactivity, the reactor power may increase to a level that is more than what is demanded. This is not desirable from safety point of view since the chain reactions may propagate uncontrolled. In such situations, the reactor power is automatically brought down rapidly by means of shut down systems. Reactivity devices, which are used for control, monitoring and safe shutdown of nuclear reactors, have gone through a variety of upgrades.

In RAPS and MAPS, control of reactor power was achieved by varying heavy water moderator level in calandria and by insertion / withdrawal of rods containing neutron absorbing material. In standardized PHWR design, from NAPS onwards, power control using moderator level variation has been deleted, as it was quite cumbersome and slow.

Moderator System

The heavy water moderator is circulated through the calandria using low temperature and low pressure moderator system. This system circulates the moderator through heat exchangers in which moderator heat is removed by process water. The cooled moderator is returned to calandria. The chemical purity and the activity level of moderator system are maintained by purification system using ion exchange columns.

Heavy water is an expensive fluid. Moreover, it absorbs some neutrons to form small quantities of tritium and therefore, its leakage from the system causes tritium activity in air. Hence, heavy water leaks from the system are to be minimized. Most of the moderator system equipment and piping is located in area inside reactor building, which is inaccessible during reactor operation. This requires that, the equipment for the system should be reliable.

To minimize heavy water inventory in the system, layout of equipment and piping was made compact. Also purification system has been shifted to outside reactor building. Number of equipment has been drastically reduced. Layout of equipment and piping has been made user friendly.

Welded joints have replaced a number of mechanical joints, the number of valves has been reduced and type of valves has been changed to those with better design. Material of heat exchanger tubes has been changed from cupro-nickel to stainless steel.

These changes resulted in better reliability, ease of operation, maintenance and in-service inspection, and reduced leakage of heavy water from the system.

In KAPS-1and 2, TAPP-3 and 4, Kaiga-3 and 4 and RAPP-5 and 6 canned rotor pumps have been used instead of mechanical sealed pumps for moderator circulation. This change has further reduced heavy water leaks from the system.

Primary Heat Transport System

This system transfers heat produced in the fuel to the steam generators, where steam is produced from light water to run the turbine. Primary coolant pumps circulate pressurised heavy water. Major equipment of the system like steam generators, pumps, piping are not accessible during reactor operation. The heavy water has high radioactivity content due to corrosion products and tritium. Hence, leakage of fluid from the system is to be minimized.

In RAPS and MAPS, eight steam generators of hairpin design were used. From NAPS onwards, the steam generator design was changed to mushroom type. Also the number of steam generators has been reduced to four. The new design has enabled provision of facilities for in-service inspection as well as cleaning. The number of circulating pumps has been reduced from eight to four, reducing maintenance efforts.

Steam generator for TAPS-540 MWe reactors

From Kaiga-1 and 2 onwards, a number of valves in the system have been eliminated. A conscious effort has been made to minimize the weld joints and in turn the in-service inspection requirements of the joints.

Turbine Generator and Condenser System

The turbogenerator system converts energy of the steam to electrical energy. Turbine blading components and turbine drain systems were modified based on feedback regarding turbine blade failures in earlier plants.

In condenser and feed water systems copper bearing alloys were used in RAPS, MAPS and NAPS. From KAPS onwards, copper free materials like stainless steel or titanium have been used to gain better control of the water chemistry.

Cooling Systems

Cooling systems provide cooling water for condenser, cooling of various equipment and reactor process systems. Heat is dissipated into water bodies like lakes, rivers, and sea or atmospheric air through cooling circuits using heat exchangers or cooling towers. The major heat load in the power station is from the condensers.

In RAPS-1 and 2, Kaiga-1 and 2 and Kaiga-3 and 4, condenser heat is dissipated through open circuits into lake/ reservoir that are available as heat sinks. Similarly in MAPS and TAPP-3 and 4 heat is dissipated into sea. In NAPS, KAPS, RAPS-3 and 4, and RAPP-5 and 6 natural draft cooling towers are used and the heat is dissipated to atmosphere. This is due to non-availability of reliable heat sinks for these units.

For rest of heat loads, once through cooling has been used in RAPS-1 and 2. For MAPS and TAPP-3 and 4 heat from rest of the loads is removed by intermediate closed loop process water. Heat from this loop is extracted through heat exchangers and dissipated to the sea through once through circuit. Intermediate closed loop process water circuit acts as a barrier for activity release to environment in case of failure of tubes of heavy water heat exchangers.

From NAPS onwards, heat from reactor process loads is removed by intermediate closed loop process water. Heat from this loop is extracted through plate type heat exchangers and dissipated to atmosphere through induced draft cooling towers by a tertiary loop.

For rest of the loads, heat is dissipated to atmosphere through cooling towers or to water bodies directly depending upon site conditions.

Electrical Systems

Electrical power system in a nuclear power station is divided broadly into two major parts. They are the main power output system and station auxiliary power system. The main power output system helps in evacuating the power generated by the turbine-generator set and transferring it to the state or regional electricity grid.

Improvements and changes have been effected in the electrical system design and equipment over a period of time. Some of these are:

- Use of static uninterrupted power supply system in place of motor generator sets.
- Use of better insulation materials of class F and H for rotating machines which has contributed to longer life of machines.
- Use of fire resistant low smoke cables for control and power cables which has helped in reducing fire incidents as well as consequential damage.
- Use of Programmable Logic Controller (PLC) based systems for supply, transfer and restoration schemes resulting in improved reliability and reduced maintenance work compared to relay-based schemes.
- Introduction of computer based supervisory control and data acquisition system (SCADA) for data logging, metering and control of selected feeder circuit breakers which is found to be useful in collecting data on electrical system of plant for analysis.
- Provision of indoor type switchyard at MAPS to overcome problems due to salt pollution in switchyard equipment located close to sea.
- Provision of gas insulated switchgear (GIS), which are compact, maintenance-free and cost effective, in the Tarapur units 3 and 4 which are also located close to sea.

Several modifications were also carried out to match the Indian Nuclear Power Plants to the existing electrical grids. Examples of these include, islanding schemes, designing equipment including turbine generator for grid frequency and voltage variations, station blackout capability, modified relay coordination, improved designs of diesel generators, UPS etc. Due to these improvements, on several occasions, the nuclear power stations continued to operate safely and uninterruptedly even when the grids collapsed.

Control and Instrumentation System

Control and Instrumentation (C&I) system monitors and controls the plant parameters. If the critical parameters deviate beyond acceptable limits, safety action is taken through the protection system.

In the first generation plants, C&I consisted predominantly of hard-wired analog electronic control and monitoring loops, relay based logics and dedicated control room indicators / recorders. In the later reactors, hardwired control and monitors were replaced by digital computer based systems for a number of applications. Also, relay logics have given way to Programmable Logic Controllers (PLC) and dedicated control room indicators / recorders are being replaced by shared Visual Display Units (VDUs) and computerized logs with enhanced Man Machine Interface (MMI) features. To have flexible approach, mosaic control panels are used in new 540 MWe reactors at TAPP 3 and 4.

Obsolescence and need for improved MMI and ergonomics for plant performance and safety was a concern for older plants. During En-masse Coolant Channel Replacement (EMCCR), the C&I systems of earlier plants, RAPS/MAPS, were also modernised.

Information Systems

A cautious and systematic approach has been adopted in computerization of control and protective systems, so as to acquire sufficient confidence in software reliability to satisfy safety concerns. Information systems that were the first to undergo computerization were Channel Temperature Monitoring (CTM) and Digital Recording System (DRS) in MAPS, Control Room Computer System (CRCS) in NAPS, Event Sequence Recording (ESR) in KAPS, and Window Annunciation logger (WAL) in RAPS. ESR and DRS are very useful for post-event analysis and have helped in rectifying many site problems. Introduction of failed fuel detection system has reduced the detection time substantially. Kaiga-1 and 2 and RAPS-3 and 4 onwards, the mainframe based CRCS system was upgraded to redundant PC based system and renamed as Computerised Operator Information System (COIS). This system has much enhanced capacity and features like display and printing of plant information in various formats, data storage for analysis, mimic representation of processes etc. Various stand-alone systems are connected through serial links to COIS for man-machine interface through VDUs in main control room. In TAPP-3 and 4 and future plants, it is planned to connect all the stand-alone computer based systems to COIS via gateways so as to have centralized plant monitoring facility. In future, this will be further enhanced to provide Centralized Operating Plants Information System (COPIS).

Control Systems

Pneumatic transmitters and controllers were used, up to KAPS, in some areas where local monitoring and control were considered adequate. For remote control areas, analog controllers were used up to NAPS. Microprocessor based single loop controllers for analog control loops were introduced in KAPS. However, to keep pace with technology, while improving the reliability and availability, in Kaiga-1 and 2 and RAPS-3 and 4, fault tolerant micro-processor based system (Dual processor hot standby system) for process controls and reactor control were introduced. The configuration consists of two independent and identical computer systems, a main and a hot standby system. Dual processors in each system detect the healthiness of that system and if one system is declared faulty, all the controls are switched over to the hot-standby system. There are exhaustive online diagnostics performed by the system to monitor its healthiness.

Keeping in line with the technological advancements, in the new reactors, the control systems were changed from hard-wired systems to computerized systems. The sequential logic as well as safety interlocks for the devices were provided in the software. As a backup, safety interlocks for the devices were hard wired also. The new control system provides automatic mode, step mode and manual mode of operations. The system also provides automatic logging of all operations, which helps in analyzing events and taking corrective actions. Also, the changes in the operating programmes can now be made with suitable modification of the software, without having to make any changes in the hard wiring of the system.

Protective Systems

Another major step was to do away with cumbersome relay logics and exhaustive wiring, which was prevalent up to KAPS. Programmable Logic Controllers (PLC) were used to replace relay logics for safety related systems and non-safety systems from Kaiga-1 and 2 and RAPS-3 and 4 onwards.

Dedicated indicating alarm meters were used for generating contact outputs for reactor protection, process logics and window annunciations up to NAPS. At KAPS a bold step of computerising the alarm generation function was taken. The system called as Programmable Digital Comparator System (PDCS) was installed. This was used for comparing the signal with set point and generating contact outputs thus replacing discrete indicating alarm meters.

The safety systems which are required to shutdown the plant, remove the decay heat, keep the plant in safe shut down state and isolate the containment so as to prevent release of radioactivity to the outside environment are still controlled using hardwired relay based logics. The availability of these systems is ensured by separate test and monitoring systems. Up to KAPS, these test and monitoring systems used to be hardwired panels and systems were tested manually. However from Kaiga-1 and 2 / RAPS-3 and 4 onwards these are replaced by user-friendly computerized testing systems.

Software Verification and Validation

Along with hardware, software also forms the backbone of any computer based system. Based on various international guidelines, procedures were evolved in the field of software Verification and Validation (V&V). All the computer-based systems were successfully verified and validated. Development of V & V techniques is being continued in collaboration with many organizations including Indian Institutes of Technology.

On-Power Fuel Handling System

On-power fuelling is a feature of all PHWRs, which have very low excess reactivity. In this type of reactor, to compensate for fuel depletion and for overall flux shaping to give optimum power distribution, refueling is carried out with the help of two Fuelling Machines (FM). The two FM work in conjunction with each other on the opposite ends of a channel. One of the machines is used to fuel the channel, while the other one receives the spent fuel. In addition, the fuelling machines facilitate removal of failed fuel bundles.

The fuelling machines of RAPS and MAPS are mounted as bridge and carriage assemblies. From NAPS onwards a fuelling machine bridge and column concept is being used to make the system safe under seismic conditions also. Various mechanisms provided allow tri-directional movement (X, Y and Z direction) of fuelling machine head and make it possible to align accurately with respect to the channels. Mechanisms provided enable clamping of fuelling machine head to the end fitting, opening and closing of the respective seal plugs, shield plugs and performance of various fuelling operations viz. receiving new fuel in the magazine from fuel transfer system, sending spent fuel from magazine to transfer systems and then to inspection bay and from inspection bay to spent fuel storage bay.

In earlier reactors there was a common fuel transfer system for both machines. In subsequent reactors, the fuel transfer system has been modified and made independent for each fuelling machine, to enable continuation of refuelling, even when one of the fuel transfer systems is unavailable. This design also reduces the possibility of bundles remaining dry as bulk of the spent fuel operations are done with bundles immersed in cooling water. In addition, the design provides emergency cooling for fuel through heavy water spray. The spent fuel movement now is in a totally closed system, which prevents any possibility of fission products coming into open atmosphere. This helps in reducing internal dose to the plant personnel. The shuttle moves from Shuttle Transfer Station (STS) to Shuttle Receiving Station (SRS) through a Shuttle Transport Tube (STT), which is now in a horizontal plane and without any bends. This design ensures cooling for the bundle right through all operations.

R&D Work in Support of PHWR and Development of Technology

BARC has been pursuing a wide range of R&D in applied engineering to cater to the needs of NPCIL for improved operation of PHWRs and to solve technical problems. Expertise for design, analysis and development in the field of mechanical component design, thermal hydraulics, vibration diagnostics, repair technology and process instrumentation, backed up by the infrastructure built-up over the years in the form of a variety of experimental facilities, test loops that enable long or short term testing of components under simulated conditions, exist at BARC. Large numbers of reactor process equipments and components have been tested at BARC to assess the behaviour under simulated conditions before putting these into regular operation. A large variety of baseline data on the process variables and performance parameters have been generated through operation of the test facilities. These data were further used in development of various codes as well as for verifying results of design analyses. Some of the major R&D activities related to development of technologies for use in PHWRs are described below.

Refueling Technology

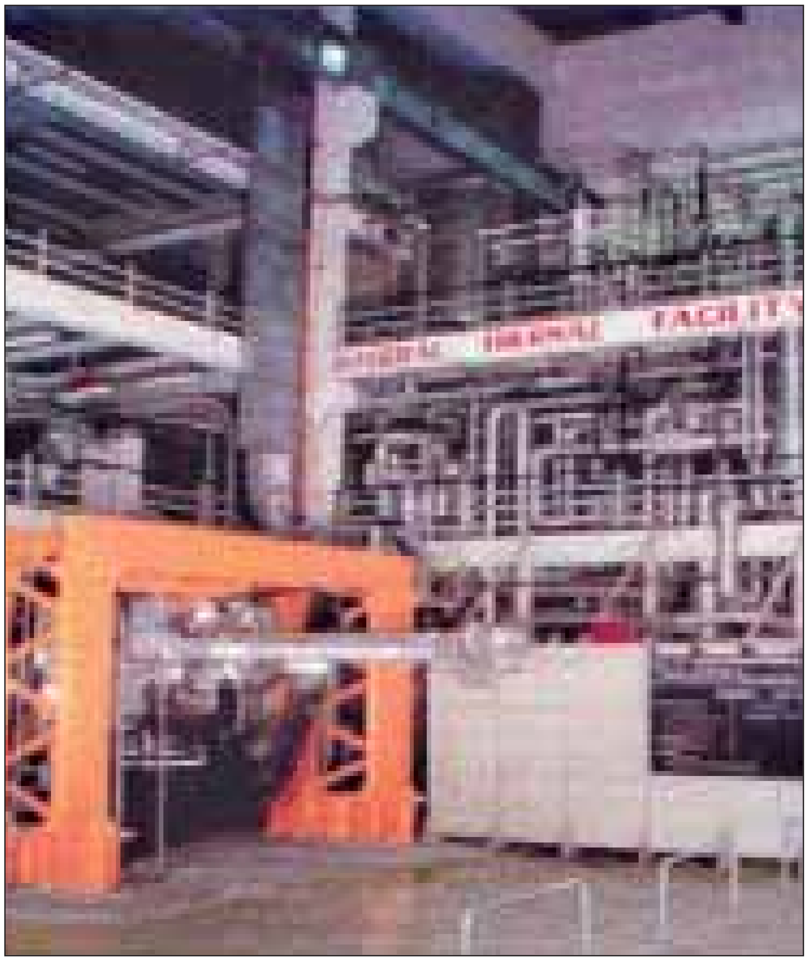
Refueling technology is one of the main thrust areas necessitating intensive R & D activities. Support is provided to the operating PHWRs in innovative design, indigenisation, as well as in trouble shooting and solving some of the complex problems in the area of fuel handling. The expertise generated in this process is utilized to develop man-rem saving tools for monitoring and life management of coolant channels of 220 MWe PHWRs. PHWR fuel was subjected to type testing to qualify the design as well as manufacturing parameters. Support is also provided for design and development of some of the systems and fuel-handling components of the 540 MWe PHWRs.

Developmental Activities for Reactors

Pressurised Heavy Water Reactors (PHWR) - 220 MWe

Valve Stations for RAPS 2

The fuelling machine valve station for RAPS-2 consisting of high pressure and high temperature piping was developed for the first time in India. This required development of special welding procedures, which was successfully achieved.



Fuelling machine test facility

Test Facility for Fuelling Machine

A test facility comprising of a coolant channel with simulated parameters was established. The facility was used to test 3 fuelling machines for RAPS-2 and 6 fuelling machines for MAPS-1&2. At the same time, to meet the ever-growing demand of fuelling machine testing a second facility called integral thermal facility was established. This facility has 2 channels with simulated parameters and facility to have one machine on either side of coolant channel. The new computer controlled system for standardized 220 MWe PHWRs (i.e. NAPS-1 onwards) was developed and used for this facility. Six fuelling machines for NAPS-1 and 2 were tested using this facility. The facility was also used to train the power station operation staff.

The older single channel test facility was renovated to accommodate the fuelling machines of standardized PHWRs. Both the test facilities were utilized to test fuelling machines of all remaining power stations namely KAPS-1 and 2, Kaiga-1 and 2, RAPP-3 and 4. Totally 33 fuelling machines have been tested and supplied to all operating PHWRs.

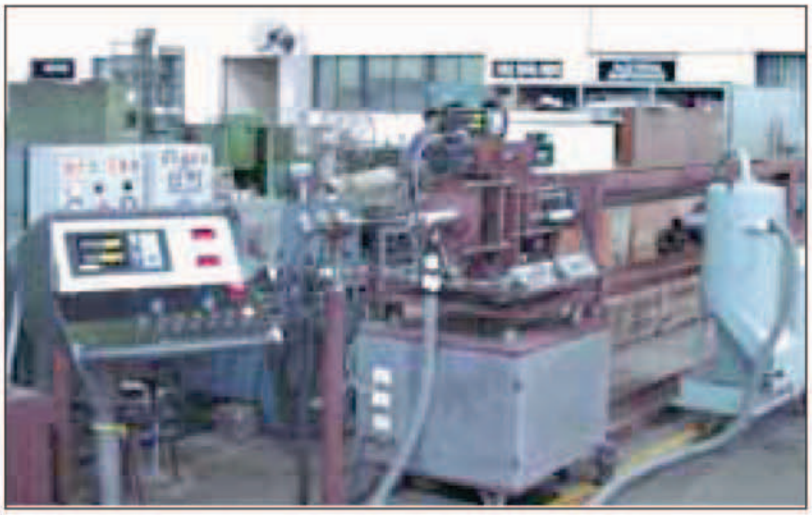
During the testing of fuelling machines, many problems related to design, manufacturing and assembly were faced. Sustained efforts were made to analyze these problems and solutions were found and expertise in this field has been developed. Various methodologies like calibration procedures, assembly-disassemby procedures etc., were developed. Also fuel handling components like extractor assemblies, separator calibration plug etc., were developed for use at the power stations. The acceptance testing of fuelling machine before supply to the power stations has resulted in reducing the commissioning time for the reactors. The expertise has resulted in reducing the downtime of the fuelling machines at the stations.

Type Testing of 19 Element Fuel Bundles for 220 MWe PHWRs

To qualify the design and manufacturing procedures of the fuel, which was fabricated in India for the first time, prototype 19 element wire wrapped fuel bundles were subjected to the type testing in the test set up established for the purpose. 19 element split special fuel bundles and spacer less fuel bundles were also subjected to the type testing in the test set up. Important data like pressure drop, drag force, friction factor etc. were generated experimentally in the test set up. Detailed calculations were carried out to quantify the amount of force exerted on the fuel bundles during residence in channel as well as during transfer operations.

Development of Tools for Handling Emergencies

During refueling, malfunctioning of guide sleeve, separator tips and snout plug may create difficult situation to handle when fuelling machine is either on the channel or having irradiated



Remote cutting machine for guide sleeve

fuel bundles. This can be overcome by using a special purpose cutting machine, which can cut the above mentioned components to solve the problem. Guide sleeve and separator cutting machines have been developed at BARC and their performance has been demonstrated.

Excessive seal plug leakage is observed some times in operating power plants. In such cases, it is required to carry out maintenance work on the closure seal face of coolant channels. This needs reactor shut down, channel isolation and draining. An End Fitting Blanking Assembly (EFBA) has been developed which can be installed on the end fitting for temporary blockage of the leaky seal plug, while continuing reactor operation. During subsequent shutdown this assembly can be removed for repair of damaged seal face. Thus it helps to postpone the repair work. A Channel Isolation Plug (CHIP) has

Special closure plug

Channel isolation plug

Front openable snout plug

been developed which enables repair of the damaged seal face without draining the channel. CHIP is installed and removed by fuelling machines.

A Snout plug boxes the FM. A difficult situation occurs if the snout plug gets stuck and FM rams are not able to operate from inside. The presence of irradiated fuel bundles makes the approach to the FM difficult. A front openable snout plug has been developed which helps in dealing with such situations. A prototype has been fabricated and demonstrated.

Magazine Auxiliary Drive

In PHWRs of 220 MWe capacity , the FM does not have redundant magazine drive. If the drive fails because of motor failure or oil system failure, and if magazine is having irradiated fuel bundles, it will be difficult to retrieve them. An auxiliary drive has been designed for use in such situations to facilitate magazine rotation from a distance behind the shielding.

Pressurised Heavy Water Reactors - 540 MWe

Development and Testing of Fuel Handling Components

New designs of ram head, special extensions, and water-lubricated bearings were subjected to elaborate testing for proving the design. A special test set up was made for this purpose. A fuel locator test facility was established for subjecting the fuel locator and liner tube to various qualification tests including endurance test.

A test facility has been established for testing the fuelling machine of 540 MWe reactors. The facility comprises of oil hydraulic valve station, water hydraulic valve station, test carriage, coolant channel, control system etc. The facility has been commissioned and testing of the first fuelling machine has been taken up.

Testing of Prototype 37 Element Fuel Bundles

37 element fuel bundles, fabricated for the first time in India, were subjected to type test as well as endurance test in the fuel locator test set up. Experiments were carried out to determine the terminal velocity of the fuel bundle in the channel. The fuel bundles were tested for more than 7000 hours to qualify the performance.

Design of Spent Fuel Transfer System

Spent fuel transfer system is required to receive irradiated fuel bundles from the transfer magazine located in fuel transfer room within the reactor building and to transport them to the storage bay located outside the reactor building. The system comprises of shuttle transfer station, shuttle transport tube, bay receiving equipment, shuttle transport system etc. The design of the system has incorporated many new ideas evolved like use of mechanical stop for fuel safety, use of rope drive for simplicity, shuttle design for better fuel safety and reliability, simplification in the system from O&M point of view etc. The detailed design has been completed. The designs were meant for manufacturing of critical components for the Tarapur- 3 and 4 reactors.

Fluid Power & Tribology

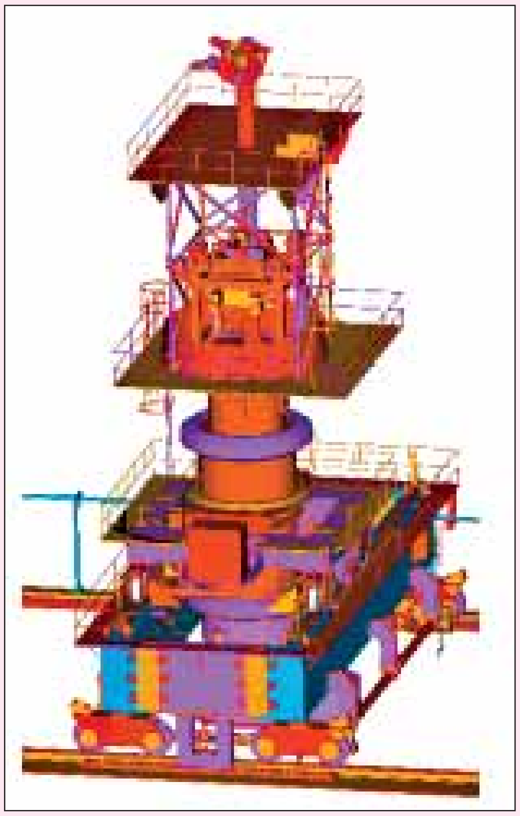
A full-fledged fluid power laboratory has been established with facilities for development and testing of hydraulic components like valves, motors, cylinders etc. As a part of the indigenisation programme and to take care of embargo, special water hydraulic valves were developed for fuel handling system.

A remotely operated electro-hydraulic servo manipulator with six degrees of freedom, having total reach of 2 m and handling capacity of 50 kgf was developed. Remote operation for cutting bolts using hydraulic tools handled by the manipulator to dismantle the bolted joints, handling of 220 MWe PHWR fuel bundle and manual override of directional valve of MAPS fuelling machine were carried out to demonstrate the capabilities of the manipulator. A tribology laboratory has been established and development of water-lubricated bearing is in progress.

Design and Development of Fuel Handling System for AHWR

In AHWR, the fuel is required to be handled in vertical condition for refueling operation. The refueling is planned to be carried out on power. For this purpose a fuel-handling system has been conceptualized and detailed design is in progress. The system is one of the biggest and dynamic systems of the reactor. It comprises of the fuelling machine, carriage and trolley, under water fuel transfer equipment, inclined fuel transfer machine, fuel storage bay, fuel handling equipment/tools,

Servo-hydraulic 6-degrees of freedom manipulator



Artist's view of AHWR fuel handling System

shipping cask etc. The fuelling machine design has been completed and order has been placed for manufacturing of prototype fuelling machine.

Coolant Channel Replacement Machine

The in-reactor service life of the Zircaloy-2 pressure tube is expected to be around 8 to 10 full power years. In order to enhance the life of the reactors using these pressure tubes, it is necessary to replace all the tubes when their service life is over. To carry out large-scale replacement of these channels, a semi-automatic, remotely operable Coolant Channel Replacement Machine (CCRM), consisting of large number of subsystems, tools and components, has been developed. The most important element of the design of CCRM is the Remotisation of tool alignment and positioning, and channel removal operations. This helps in grossly reducing the man-rem expenditure.

The CCRM consists of a self-elevating work table, tool positioning and aligning system, servo manipulator, remotely operated EOT crane and a large number of tools to perform various tasks for component removal and refurbishing. The prototype of the machine, will be used as test facility for future developmental activities pertaining to repair and replacement technology.

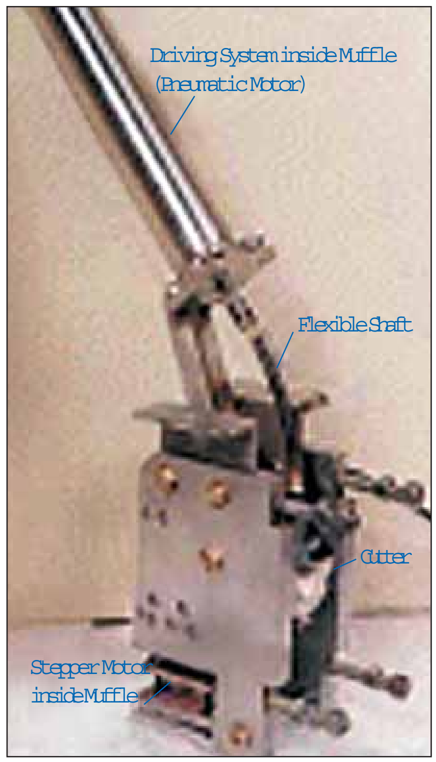
Coolant channel replacement machine

Boat Sampling Technique

A non-destructive testing technique namely, the Boat Sampling Technique (BST), has been developed for obtaining in-situ metal samples from the surface of an operating component. The metal samples could be used for metallurgical analysis to confirm the integrity of the component. This technique obtains boat shaped samples without plastic

CBN coated cutter

Scooped region and sample



Sampling module

deformation or thermal degradation of the base material of the components.

This technique can be used remotely and under water to obtain samples from a desired location. The BST incorporates a sampling module and mechanical, pneumatic and electrical sub-systems. The weight of boat sample would be 4 grams and the time required for obtaining each sample is estimated to be 180 minutes. The technique has been qualified in a mock-up to obtain samples from core shroud of TAPS reactor.

3 MW Boiling Water Loop

The 3 MW Boiling Water Loop is an experimental facility for carrying out heat transfer and fluid flow experiments relevant to nuclear reactor systems. The loop is designed for carrying out both single-phase and two-phase flow studies on blowdown from high pressure systems, heat transfer associated with emergency core cooling, dryout and post-dryout heat transfer studies, sub-channel mixing in rod cluster experiments etc. The loop has provision for 3 MW DC power supply to its test sections. Two test sections can be accommodated both horizontally and vertically in the loop.

3 MW boiling water loop

Facility for Integral System Behaviour Experiments (FISBE)

The Facility for Integral System Behaviour (FISBE) closely simulates the primary heat transport system and associated components of the secondary system of the Indian PHWR. Power to volume scaling philosophy is followed in the design of this facility. The elevations, pressure and temperature are maintained at the same levels as in the reactor. This facility will be used to understand the physical phenomena that occur during accidental conditions and operational transients.

Vibration Laboratory

Realizing the importance of in-depth study of vibration behavior of machinery, structures and components, action was initiated in 1982 to set up a vibration laboratory. The required front-end hardware and the back-end analysis tools were procured on a small scale. With growing demand for large-scale measurement and analysis the facilities were augmented in a phased manner. Today the laboratory is equipped with a large number of measurement and analysis equipments that can cater to vibration testing of large components, dynamic characterization of structures and components, development of on-line monitoring systems, advanced diagnostic tools,

Calibration facility established for pressure, temperature and flow validation of analytical models, study of rotor dynamics and in-situ estimation of rotor unbalance. Under MOU or technical collaboration, the vibration laboratory is closely working with DRDL, RCF, Konkan Railways and Tata thermal power in developing advanced diagnostic tools for missile components, plant machinery, railways and turbo-generators.

With the expertise and state-of–art equipment, the laboratory is capable of undertaking any challenging task within and outside DAE. Vibration laboratory investigates actual vibration problems on a day-to-day basis. For example, detailed analysis of Sparger tubes in the moderator system installed at MAPS was carried out to enhance the power of the plant from 170 MWe to 225 MWe.

Stress Analysis

Stress analysis plays very important role in the design and development of many critical parts of the reactor assembly. Apart from theoretical analysis based on existing model codes, it is also necessary to validate these results with experimental simulations. As a part of the component development programme, theoretical and experimental stress analysis were carried out for many complex shapes like end shield of NAPP and diaphragm support of the integrated calandria-end shield assembly.

Facility for earthquake engineering studies

Nuclear plants have to meet seismic design requirements to ensure safety in case of earthquakes. As part of efforts in this regard, methodologies were developed to generate design basis ground motion for nuclear power plants. Response of various systems, structures and components were evaluated and they were designed to withstand the design basis motion. Several simple, accurate and computationally efficient techniques have been developed for seismic analysis. Devices to control the seismic stresses in the Structures, Systems and Components (SSCs) like isolators, elasto-plastic dampers, lead extrusion dampers and friction dampers have been developed. Older plants were requalified to present day seismic design requirements.

Fracture Mechanics

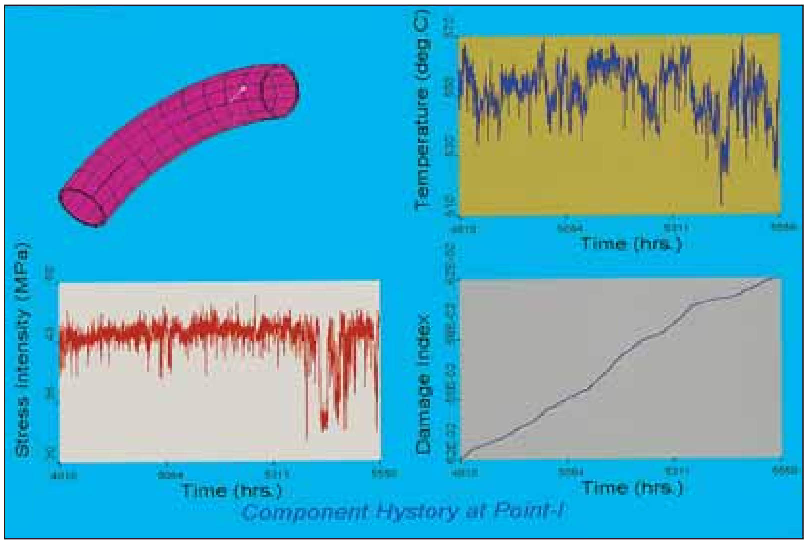
Various analytical and experimental fracture mechanics studies have been carried out for assessing component integrity. Studies are presently being carried out on effect of earthquake loads on Leak-Before-Break (LBB) assessment of high energy piping and development of advanced codes.

The current LBB evaluation is based on the monotonic tearing instability criteria. However, during this event the NPP PHT piping experiences large amplitude reversible cyclic loads, which are known to significantly decrease the fracture resistance. A cracked component, which is safe for monotonic load, may fail in limited number of cycles when subjected to fully reversible cyclic load of the same amplitude. Keeping this

PHT elbow having circumferential flaws being tested in view a series of cyclic tearing tests have been carried out on cracked straight pipes subjected to cyclic pure bending loading. Analytical investigations have been made and cyclic fracture assessment procedure has been established to make the LBB design more realistic.

Advanced Fracture Studies

Comprehensive fracture mechanics studies have been pursued over many years to understand the behavior of PHT piping components under severe accident scenarios. These studies also helped to generate a material database and many close form solutions which are immensely helpful for leak-before-break qualification of PHT piping. The studies have



Temperature/stress history experienced by HRH pipe elbow at NTPC plant

experimental as well as theoretical fronts. The experimental work involves specimen level as well as component level testing under monotonic and cyclic loads. Theoretical studies primarily involve finite element analyses of the components using advanced fracture mechanics concepts such as constraint dependent J-R curve.

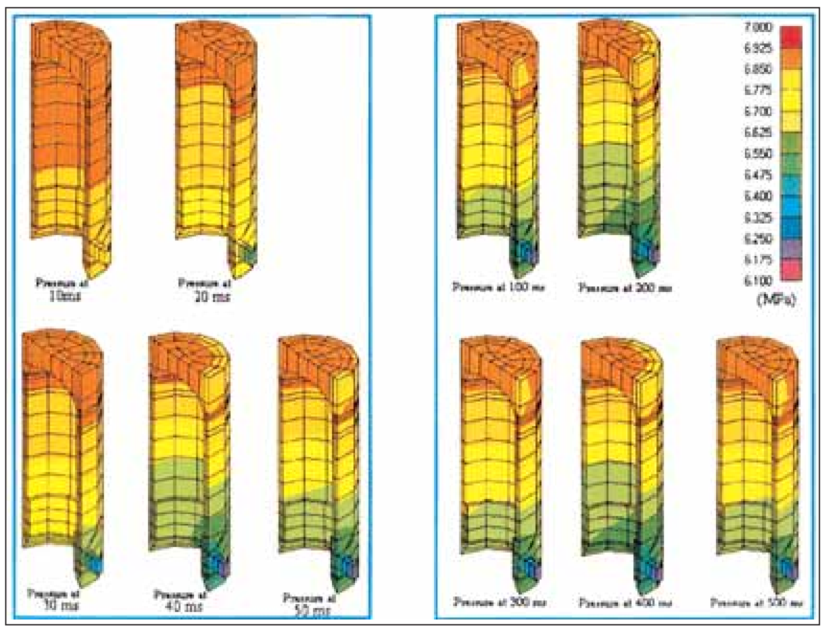
Provision of real time creep-fatigue monitoring system is a regulatory requirement in power plants of many countries. Such a system is helpful during in-service inspection, life extension program, and generation of database for future design. A finite element based in-house system 'BOSSES' has been developed for such purpose. The system is in operation in heavy water plants and NTPC, Dadri, plant for many years. The system has three main modules, namely, data acquisition, FE solver and visualization.

Component Development

Technical support has been provided for design and development of reactor structures and components for RAPS, MAPS, NAPS and 540 MWe reactors by carrying out stress analysis. Several developmental tasks were performed using analytical methods, photo-elastic stress analysis methods, strain gauge experiments and radiometric testing. These studies helped to evolve new design for the PHWR projects and in addition some of the challenging problems of on-site component installation, fabrication and commissioning were solved.

Thermal Hydraulics/Reactor Safety

Postulated pressure tube failure accident inside the calandria is one of the serious design basis accidents for calandria. In the early seventies studies were carried out to obtain the limiting pressure on the calandria shell due to addition of the flashing fluid, bubble growth, shock wave propagation, bubble condensation and bubble collapse with a one dimensional model. This study helped to identify the influence of the above parameters. Detailed studies were carried out with in-house finite element codes FLUSOL and FLUSHELL. Predictions were verified with the reported experimental results of the simulated pressure tube failure accident experiments. Subsequently the

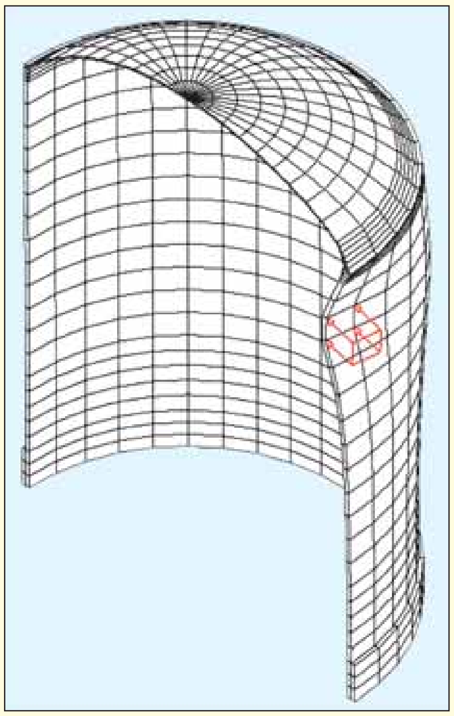


TAPS core shroud acoustic load analysis code FLUSHELL has been up-graded for acoustic load evaluation and associated fluid-structure interaction analysis within two-phase medium for Loss Of Coolant Accident (LOCA) due to postulated recirculation line break of TAPS reactor.

Nuclear Containment Safety Research

Containment safety research is being carried out to enhance the performance of nuclear containment with improvements in engineered safety features and to demonstrate safe technology for public acceptance of nuclear power programme. Thermal hydraulic, hydrogen distribution and structural safety assessment, for design and beyond the design basis accidents, have been carried out for the Indian reactor containments. In-house computer codes CONTRAN, HYRECAT, BOXCAT, ULCA and ARCOS3D have been extensively bench marked with experimental test results. Recent studies for accidental impact of aircraft on PHWR containment have helped to demonstrate the adequacy of the ultimate barrier.

Methodologies for safety analysis of different reactors were evolved. Some of the important phenomena identified during analysis were stagnation channel break, cold pressurization during LOCA and non-LOCA events, blind LOCA and special phenomena during defueling and refueling for AHWR. The maturity of the R&D was demonstrated by participation in various international standard exercises for validation of codes for various phenomena occurring during small break, large break, ECCS injection, single and two-phase natural circulation under depleted inventory condition and pressurized thermal shock. Several computer codes were indigenously developed and validated and various international codes were adopted and used for several applications. Some of the important analyses carried out are TAPS core shroud analysis, TAPS MOX fuel analysis, Retrofitting analysis for RAPS reactors, PWR-KK Analysis and complete safety analysis for AHWR reactor. Assessment of thermal-hydraulic safety after design basis events and beyond design basis events was carried out for different reactors like TAPS-BWR, PHWR, PWR-KK, CIRUS, DHRUVA, PFBR and AHWR.



Deformed shape of outer containment wall

Various supporting experimental studies for different phenomena like leak before break, blow down, molten fuel coolant interaction, channel heat up, vapour pull through are being carried out at various universities as collaborative projects. Two computer codes DYNA220 for 220 MWe PHWR units and DYNA540 for 540 MWe PHWR units have been developed. These are full scope plant specific computer codes to simulate the whole nuclear power plant with all the process parameters with their associated control. These two codes are being extensively used by NPC to carry out the design modifications and to study the transient analysis of our PHWR units for meeting regulatory requirement of AERB.

Risk Analysis

Fire PSA studies have been carried out for MAPS and, based on these, re-routing of safety related cables, installation of fire barriers, etc., have been carried out to reduce vulnerability of

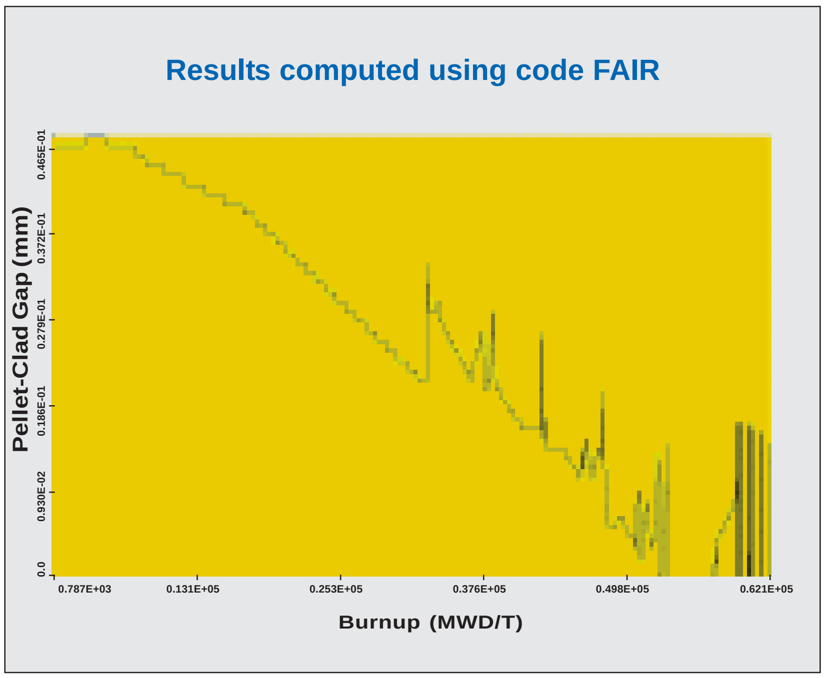
LOCA simulator

the MAPS systems to fire. Facilities have been designed, developed and are in operation to carry out thermal ageing, radiation ageing and Loss of Coolant Accident (LOCA) environment qualification studies. Effects of temperature and radiation on performance/reliability of the hardware systems/ components/materials are also being studied at the PANBIT facility setup in BARC.

New Analytical Models/Codes

A finite element based code FAIR has been developed to analyze the integrity of clad under various reactor scenarios. This code models complex thermo-mechanical and chemical processes occurring in a nuclear reactor fuel pin.

High performance equation solvers and parallel processors are being developed for very large scale computations involved in the rigorous safety analysis of reactor components under various postulated scenarios. The direct sparse solver under use consists of four stages, namely, processing, ordering, symbolic factorization and cholesky decomposition. A great degree of success has also been achieved in parallelizing complete finite element in-house codes along with solvers. Through such efforts the speed of computation has been increased few hundred times over last few years.



Variation of pellet-clad gap with burn-up in a fuel pin with a free standing clad

> *"…From an industry perspective, the Departments partnership approach has greatly facilitated synergising of efforts to achieve National goals…"*
>
> ***- B.N. Kalyani, CMD,***
> *Bharat Forge, Pune*

Quality Management Systems

Indian nuclear power programme can rightly claim credit for ushering in a quality management system in many industries. It laid emphasis on a planned and systematic approach to build quality into a product at every stage. The focus was shifted from the regime of inspection oriented quality control, which depended on defect detection, repair and rectification, to defect prevention at the source. Structured programmes of quality assurance were initiated covering areas of design, manufacture, testing, commissioning, operation and maintenance. The important steps introduced to achieve good products were: qualification of all products, personnel, equipment, consumables and even contractors; traceability of material and consumables by positive identification; mock-up and trials before commencement of actual production; testing of simulated coupons; qualification of products for environmental and seismic requirements; and documentation of all steps and process parameters including non-conformance.

In the field of inspection and testing, various techniques including non-destructive examination, leak testing, optical and laser methods have been successfully developed and put to use. In development of these technologies, DAE provided the lead by procuring the necessary equipment, training the personnel, developing procedures, conducting the tests, and sharing the knowledge with the industry. Training of industry personnel was also undertaken.

The expertise in Total Quality Management, Quality Assurance, Inspection, and high quality culture in DAE have enabled it to obtain contracts from many private industries for providing these services. Profit is not the primary objective of DAE in providing such services but to disburse spin-offs to others and take part in the nation building activities. DAE has also picked up some good practices from other organizations, as this is part of its continuous march towards excellence.

Nuclear Grade Fabrication

Fabrication of nuclear pressure vessels, heat exchangers and components is an area where there has been fruitful co-operation. Design and fabrication of nuclear components have to conform to mandatory codes, and stringent methods of quality assurance. Indian industry needed considerable inputs for taking up such jobs. Since the mid 70's, DAE has made available its R & D and design facilities as also funding through development contracts and came forward to assist and fund the Indian industry to create necessary facilities. A state-of-the-art workshop at BARC was established and technologies for fabricating some of the initial components were developed here and later on passed on to the industries.

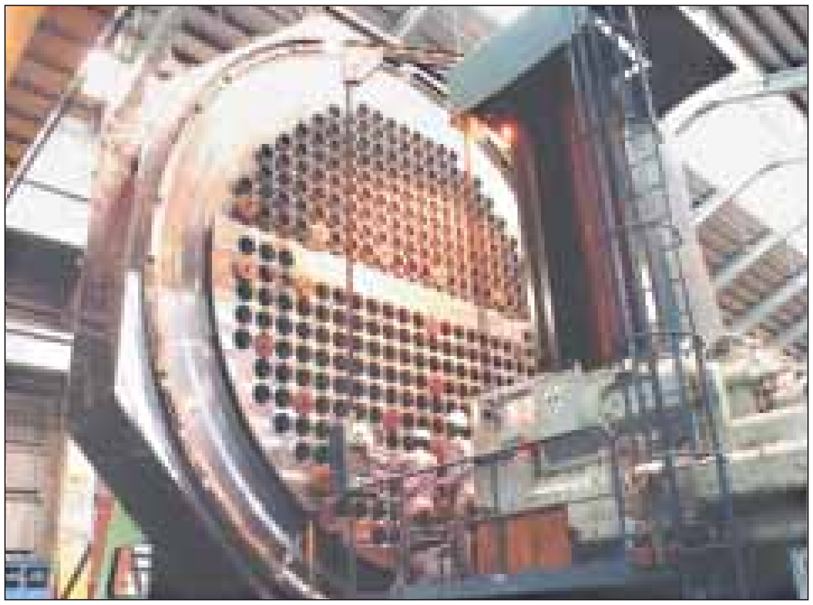
Development of dedicated nuclear fabrication bays and clean room conditions, installation of high-technology machinery, introduction of advanced methods of welding, development of welding consumables, welding procedures, training and qualification of welders and inspection engineers, adoption of modern methods of metrology and non-destructive evaluation techniques are some of the areas where there has been constant and continuing co-operation. Many industries have benefited from this transfer of technology and its spin-off effect.

Reactor components

Large reactor components like calandria and end shield, with a diameter of about 6 metres, need to be fabricated and precision machined. Engineers from Indian industries and DAE jointly took up this challenging task. Concerted efforts were put in to develop welding techniques to achieve welds of required quality standards and distortion control to achieve dimensional stability requirements. Handling and precision machining of such large components was a new experience.

Coolant Channel Components

Developing manufacturing facilities for coolant channel components made from zirconium alloys was one of the most challenging tasks. DAE decided to develop in-house facilities



Machining operations on end shield

for this purpose. Overall capability for carrying out the work involved in all stages from treatment of ore and production of zirconium alloys to machining of finished products such as coolant tubes, calandria tubes, fuel tubes and garter springs has now been achieved.

Quality requirements for these products are very stringent. Chemical composition is required to be controlled at various stages. The manufacturing processes influence material properties of the end products and thus control of manufacturing processes at each stage is important. The challenge has been successfully met and these coolant channel components were made available to all the PHWR units except the first one at RAPS. The coolant tube material was changed to zirconium-niobium alloy from KAPS-2 onwards. The complete manufacturing cycle of these tubes has been evolved in-house.

In addition to coolant tubes and calandria tubes, zircaloy components for reactivity and shutdown systems, which require development and use of special welding techniques for zirconium alloys have also been manufactured.

Manufacturing of stainless steel end fittings of coolant channels was also a difficult task. This was because of the requirement of large number of precision-machined dimensions at various locations. Initially this task was entrusted to the Central Workshop of BARC. End fittings up to KAPS were machined there. After successfully establishing the manufacturing cycle this technology has now been passed on to the Indian industry.

Precision machining and assembly

Precision machining of critical components is another area of fruitful co-operation. Often the components are very large in size and the design and application demand very close tolerances, high surface finish and surface treatment requirements, complicated assembly procedures, machining of odd sized components, heat treatment between various stages. DAE has established a continuing relationship with industry and has actively participated in development of capabilities in this regard. The indigenous manufacture of several critical components like end shields, fuel handling equipment etc. is evolving continuously with adoption of better technologies as time progresses. Some examples of modern technological processes/equipment presently adopted for work on reactor components are multiple axis numerically controlled machine tools, co-ordinate measuring machines, laser devices, vacuum bonding, vapour deposition, investment castings, powder metallurgy, wire cutting electric discharge machines, computerized control systems etc.

Machining of the end shields at manufacturer's works

Fuel Handling Equipment

The fuelling machines, for on-power fuelling of PHWRs, are among the most complex equipment and are comparable to aircraft or space equipment. When the work of manufacture of the first fuelling machine head and carriage for RAPS-2 was taken up, there was a considerable gap between the technology required for their manufacture and that commercially available. The technology required for manufacture of fuelling machines was gradually built up from the grass root level with the help of entrepreneurs/manufacturers who had the potential as well as the will to take up the challenge.

The complete technology of manufacture, inspection and testing was developed by close interactions between the manufacturers, designers and quality assurance personnel. All the raw materials and standard proprietary items were imported for the first project. Design and material of many items were changed to suit established manufacturing routes. Alternative sources for supply of raw materials and proprietary items were

Fuelling machine head

also located in the meanwhile. All the fuel handling machine components required for the first indigenously built project at MAPS could be successfully manufactured in India following this multi-pronged strategy.

Steam Generators and Heavy Water Heat Exchangers

Steam generators and heavy water heat exchangers required for the RAPS-1 were imported. In all subsequent reactors, these were manufactured indigenously with imported raw materials, such as special tubes, forgings, etc. Overlaying, drilling, tube-to-tube sheet welding, etc., involved in the manufacture of steam generators and heavy water heat exchangers required considerable developmental efforts. These are to comply with the requirements of various sections of ASME Code.

The steam generators, up to MAPS-2, consisting of a horizontal drum welded to a bank of hairpin heat exchangers, are not amenable to in-service inspection of tubes, as called for in the current codes/regulations. Mushroom type steam generators have now been adopted for the later plants.

Electrical System

In the late 1960s, Indian electrical industries were mainly catering to cement plants, chemical plants etc., apart from 110 MWe thermal power stations. They were not geared to meet equipment ratings for 220 MWe units coupled with stringent requirements due to factors like environmental qualification of equipment, radiation resistance, stage wise inspections, higher rating of equipment, higher short circuit rating of switchgears, motor control centres, and requirement of reliability data etc.

Also, the auxiliary power supply system of a NPP is quite different from that of other utilities because availability of power is essential even when the plant is under shutdown. Some of the electrical equipment, which were developed by industry to meet the requirements of nuclear power stations, are motor-generator sets, automatic constant voltage rectifiers, batteries with high discharge capability, motors suitable for and housing special bearing design and lubrication system to take care of operation under severe ambient conditions and longer life, transformers with higher MVA capacity were designed to cater for 220 MWe units, underwater and high radiation field lighting system and static UPS system.

Qualification of Products/Components

NPCIL and the industry also interact in the area of setting up appropriate facilities for qualification of components and products for environmental, radiation, fire, seismic and reliability requirements.

Control and Instrumentation (C&I)

Over the years, the contribution of Indian industry to the C&I of NPPs has grown on two fronts. Due to the general development of indigenous C&I industry, as well as special indigenization efforts taken by DAE in specific areas, an increasing number of instrumentation items became indigenously available for the NPPs. Also, the C&I systems have been continuously evolving in step with the advancing technology in the international field.

In RAPS, while the main control room panels were indigenously fabricated, all the operator interface units (indicators, recorders, controllers, hand switches, pushbuttons) were imported as also most of the field instrumentation items. Some of the systems required for nuclear application, such as reactor regulation and protection systems, fuel handling control logic system, channel temperature monitoring system etc. were designed and manufactured indigenously using imported components.

Apart from the above, a number of instrumentation items became indigenously available due to the general development of Indian C&I industry and collaborative ventures with foreign firms involving technology transfer. However, of late, continued availability of instruments from Indian industries with foreign tie-ups is becoming more uncertain due to embargo and restrictions imposed by some countries. Some of the C&I items developed indigenously are: Resistance temperature detectors, high pressure-high temperature solenoid valves, venturi tubes and nozzles for flow measurement, bellows sealed instrument isolation valves, photo coupled indicating alarm meters, and microprocessor based single loop electronic controllers.

The control centre instrumentation of Indian NPPs, mostly supplied by Electronics Corporation of India Limited (ECIL), has undergone upgradation over the years, following the global trend. The analog control and monitoring loops and relay-based systems were replaced by digital computer based systems and programmable logic controllers in a number of applications for later plants. Dedicated control room indicators and recorders used in earlier plants are being replaced by shared CRT based displays and computerized logs with enhanced man- machine interface features.

Overall Impact of the Interaction with Indian Industries

Indian industries have successfully developed and adopted various technologies for manufacturing processes, inspection, testing and quality surveillance to meet the critical requirements of various international nuclear codes and have supplied components and equipment for most of the systems for Indian nuclear plants.

In the early sixties, when the nuclear power programme was initiated, Indian industrial experience was limited to manufacture of cement kilns, boiler drums, and simple chemical reactor vessels. It can be seen that Indian industry has made significant contribution to the nuclear power programme over the years. Today the Indian industry contribution covers the entire range of products from civil structures and materials, nuclear equipment, turbo-generator and related equipment, process equipment and electrical items to control and instrumentation systems.

The success of the partnership between DAE and industry can be gauged from the fact that the foreign exchange content of Indian nuclear power plants has been reduced to 10%. There are now more than a thousand organizations both private and public, which have risen to the occasion and are capable of manufacturing internationally comparable, nuclear grade equipment. This has enabled India to operate NPPs reliably and safely, despite technology control regimes - all credit to the Indian industry.

"...We at the Bharat pumps and compressors limited have been very closely knit with DAE since our inception in 1971. Over the period of 33 years, we have been associated in supply of centrifugal and reciprocating pumps, reciprocating compressors and industrial gas cylinders to various projects under DAE. Due to our long association and patronage received from various DAE officials from time to time, we have immensely benefited specially with regard to establishment and standardisation of our quality systems in conformance with the quality requirements stipulated by DAE..."

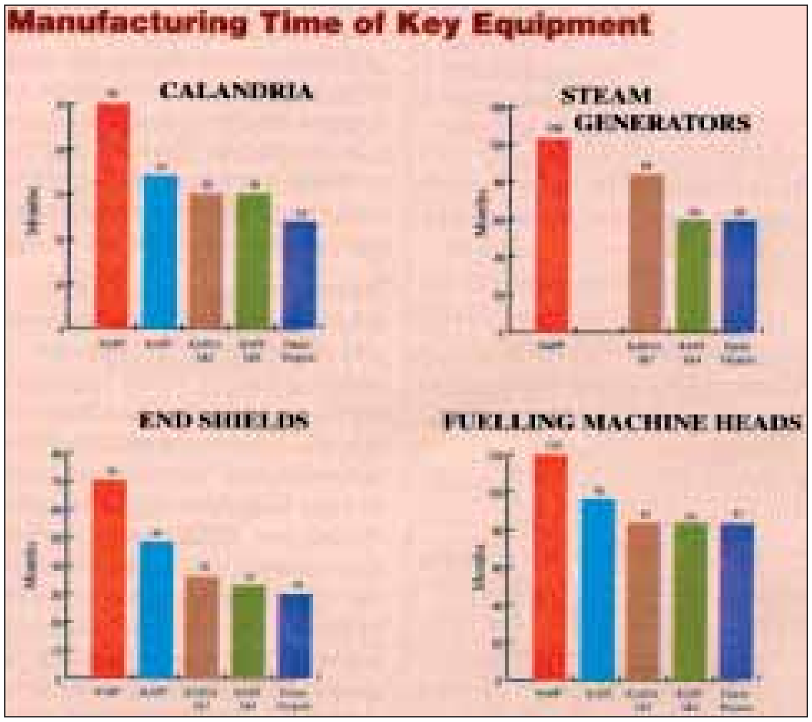
- ***V.S. Singh, Managing Director,***
Bharat Pumps and Compressors Limited

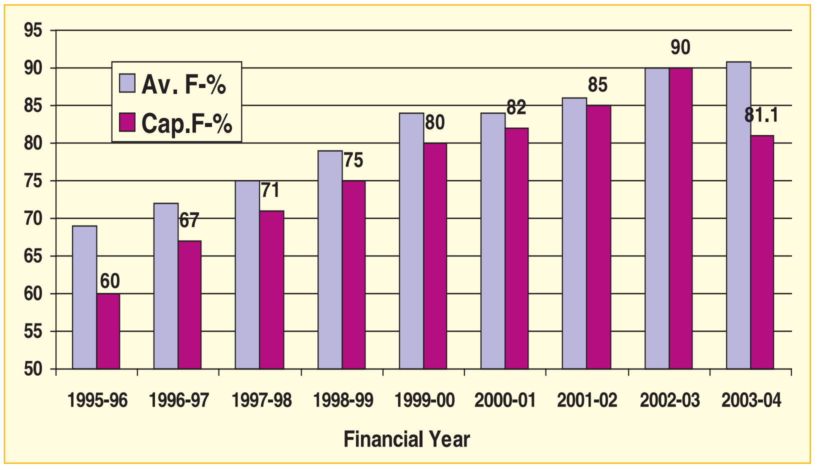
Project Implementation

In the competitive environment prevailing in the power sector and related reforms in the country it has become imperative to bring down the tariff by enhanced capacity utilisation, reduction in capital cost and project construction time. Project construction schedule is important for nuclear power projects, as shorter construction schedule will result in lesser escalation of project costs and interest during construction. In the past, due to the technology development phase the programme had to go through, the construction schedules were relatively longer. With the gaining of experience and attaining a stage of maturity of technology, reduction of construction times of the ongoing projects is planned and is being achieved. The salient areas on which actions have been taken for shortening the construction time and also achieving faster growth of nuclear power, are standardization of design, going in for larger capacity units, well-defined implementation of pre-project activities ahead of the project financial sanction, adopting the project implementation strategies of going in for large supply-cum-erection packages, and where the industry is capable, Engineering, Procurement and Construction (EPC) contracts to reduce interfaces and number of contracts, carrying out mechanical erection activities in parallel with civil construction, using advanced project management techniques, and computerized project monitoring and accounting systems.

Historically the material supply to construction sites has been arranged by procurement agency through large number of purchase orders for individual set of items. For the PHWRs under construction, radical and significant changes have been introduced in the procurement and erection contracts for power plant materials and equipment by adopting what is popularly known as "work package concept". It has the basic advantage of fixing single point responsibility.

The work package approach has resulted in reduction in number of purchase proposals resulting in optimal use of engineering manpower allowing strengthening of core competence in engineering and analysis, single point responsibility and hence focused coordination work, expeditious placement of purchase orders by the contractor, reduction in supply and erection work and time schedule, and reduction in construction manpower. (from earlier over 1000 for a project to





Performances on Capacity (Cap-F) and Availability Factors (Av.F) of nuclear power

less than 350 personnel). The best of Public Sector Units are effectively spending Rs. 500 crores/year but NPCIL has crossed more than Rs.1000 crores/year.

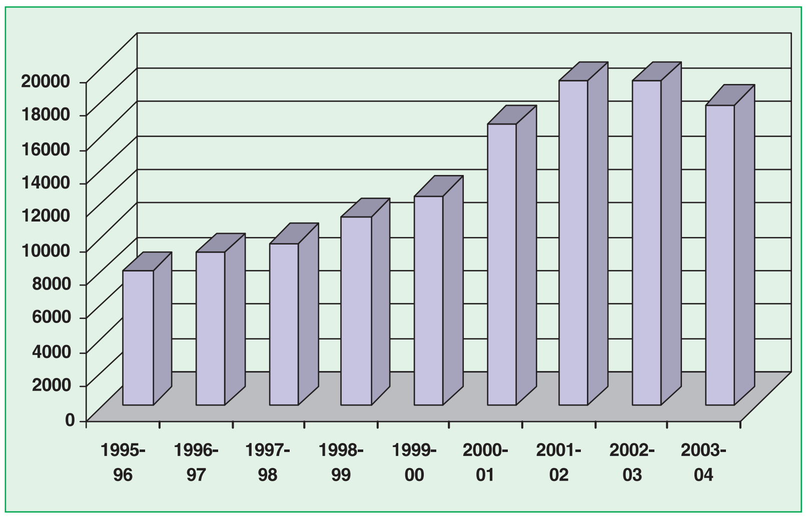
The above measures are yielding results in the projects under execution. The construction work on 8 reactor units with installed capacity of 3960 MW is proceeding at a rapid pace with project schedules of less than five years from first pour of concrete.

Operation of Nuclear power Plants

India has acquired an experience of about 220 reactor years of commercial operation of nuclear power plants. In the 1970s, three units (TAPS-1 and 2 and RAPS-1) were in operation. In the 1980s, RAPS-2, MAPS-1 and 2 and NAPS-1 were added to the capacity. In the 1990s and up to the year 2000, remaining units were added to constitute the 15 units presently in operation.

The radiation exposure to public due to operation of NPPs is demonstrated to be less than 5% of the conservative permissible limits set by AERB and adds less than a few percent to the already existing natural background. For example, the natural background is around 200 mrem/year around our NPPs (which the public will receive whether NPPs are there or not) and NPP operation will add 1 or 2 mrem/year i.e. 200 will become 201 or 202 mrem/year. Continuous reduction is also taking place in the exposure of staff involved in the operations to radiation. The personnel dose limits set by AERB are being adhered to strictly for all the operations.

In the early period of operation up to 1987, the overall annual capacity factors were in the range of about 50%. Problems related to grid, turbine and generally conventional equipment contributed to the non-availability periods. A real turnaround in the performance of the operating nuclear power plants started from 1995-96. The overall annual capacity factor of the NPCIL units in operation has progressively improved to 90% in recent years. From a level of 3000 Million Units (MUs) of nuclear electricity generation in 1981-82, NPCIL's generation has now grown to the present level of about 19250 MUs. Correspondingly the financial performance of NPCIL has also been very good. It is now a dividend paying company. It is selling electricity to the State Electricity Boards (SEBs) on a competitive commercial basis for the past several years.



(Generation in million units vs financial year)

The turnaround in performance of the nuclear power stations has been possible due to structural changes, motivation, managerial training, enhanced technical training, systematic root cause analysis of the events, effecting improvements, and avoiding repeat incidents. Special emphasis was laid on outage management, which led to reduction in outage periods. Preventive and predictive maintenance and in-service inspection techniques have been strengthened. The state-of-the-art tooling, operation and maintenance procedures, manpower training and surveillance practices have been incorporated. Improvements in the plant systems have been progressively carried out based on the experience gained over the years. Above all, the management practices have also been strengthened to enhance commercial orientation while at the same time nurturing safety culture. All these have led to a maturing of technology by which such a remarkable improvement in performance has been achieved.

Generating Performance of NPCIL Nuclear Power Plants

For the year 2002, KAPS-1 achieved the distinction of being the best performing unit in the world, in PHWR category. It was ranked first in the world amongst 31 operating CANDUs / PHWRs, with a Gross Capacity Factor (GCF) of 98.4%. For the calendar year 2002, three NPCIL plants were amongst the best five PHWR plants in the world. Annual outage time has also been reduced progressively. KAPS-2 completed its annual maintenance shutdown (ASD) in a record time of 18 days for Indian PHWRs.

NPCIL has technical cooperation with the following international organizations as a member:

- World Association of Nuclear Operators (WANO)
- Candu Owners' Group (COG)
- International atomic energy Agency (IAEA)
- World Nuclear Association (WNA)

The above organizations offer various programmes with an aim to improve safety and reliability of NPPs. Enhanced interaction with other NPPs in the world through these organizations has played an important role in the overall performance improvement achieved by NPCIL. NPCIL has also provided services of its experts to these organizations for the success of their programmes.

Challenges Faced Successfully Over the Years

The Indian Nuclear Power Programme faced many challenges. Each of these challenges was met successfully and the programme emerged stronger. It would be worthwhile recalling some of these exciting experiences.

Replacement of Feed Water Spargers in TAPS-1

Cracks in feed water spargers and nozzles (components of the reactor that are subjected to maximum cyclic loading) were detected in similar reactors operating abroad and this was a subject of concern for TAPS. To meet this challenge, necessary inspection and repair equipment was developed. During 1987, the feed water spargers were inspected and, in the case of TAPS-1, replaced with spargers of modified design. This was a major engineering challenge as it involved working from inside the reactor. The entire action plan for assessment, design and replacement of feed water spargers of TAPS was carried out indigenously. This was a major experience in modification of vital reactor internal components.

Core Shroud Inspection in TAPS-1 and 2

Commercial operation of TAPS began in the year 1969 and the station has been operating satisfactorily ever since. Initially many teething problems were faced, which were critically reviewed and overcome by suitable design changes. In order to improve the station's performance several engineering modifications have been incorporated.

The BWR reactor internals support the core, direct the water flow, and separate steam from water. In addition to the core support structure, the internals comprise of the feed water spargers, the jet pump assemblies and the steam separator and dryer assemblies. Severe degradation of the BWR reactor internals due to inter-grannular stress corrosion cracking (IGSCC), irradiation assisted stress corrosion cracking (IASCC) and fatigue has been observed at operating plants abroad. The core shroud is the most critical component among a list of internals susceptible to IGSCC. IASCC affects sections subject to high neutron flux even at relatively low stress level. Identification of cracking at the circumferential belt line region welds in several plants during 1993 led to most BWR plants in the world carrying out inspection of their core shrouds and other internal components during planned outages, and some BWRs identified extensive cracking.

A failure of reactor internals could affect safety functions controlling the power and cooling of the fuel may be in jeopardy. Extensive developmental work was carried out and special techniques were developed for the inspection of core and core shrouds of TAPS-1 and 2. The inspection was carried out successfully and no damage to core shrouds was found in both TAPS-1 and 2.

Repair of End Shield in RAPS-1

The end shields, about 6m in diameter and weighing about 120 tonnes, are installed at both ends of the calandria. The end shields are provided to support the coolant channel assemblies, and to provide shielding from high radiation fields within the reactor core so that fuelling machine vaults are accessible during shutdown.

After satisfactory operation for some years, the south end shield of RAPS-1 developed leaks in September 1981. Detailed remote examination using periscope, bore-scope, fibre-scope, helium leak test etc. was carried out. Investigations revealed leaks in the calandria side tube sheet (CSTS). CSTS is made of 3.5% nickel carbon steel and this material became brittle because of high energy neutron bombardment. Calandria side tube sheet is located towards calandria and hence is inaccessible. More over, there is high radiation field in the fuelling machine vault. Hence, repair is very difficult. Various leak sealing techniques were worked out. Initially chemical plugging method was tried to seal the leaks but it did not succeed. Later, various repair techniques were evolved and mock up trials were carried out. The repair scheme adopted finally involved use of electric discharge machining (spark erosion) for cutting and removal followed by use of specially developed boring machine, boring bar guide for boring CSTS and Fuelling machine Side Tube Sheet (FSTS) bores and installation of mechanical closure plug assembly with suitable strap to seal the opening and the ligament cracks.

Special remotely operated precision tools and guide facilities were developed to work in the high radiation areas. Various precision tools and guide facilities were required to carry out the job. Grafoil was used as gasket material. Plug installation was done with the help of a special plug manipulator. For radiation shielding inside the slab bore, a steel shield plug was assembled. The repair work was completed successfully and reactor was re-started in February 1985. However, power generation was limited to 100 MWe to restrict the stresses in the end shields to a lower value. Recently leaks developed in north end shield also. A similar repair method was adopted and the repair was carried out. Both end shields are now repaired successfully and RAPS-1 is operating satisfactorily. Further analysis enabled increased power generation up to 150 MWe.

Repair of Over Pressure Relief Device (OPRD) in RAPS-1

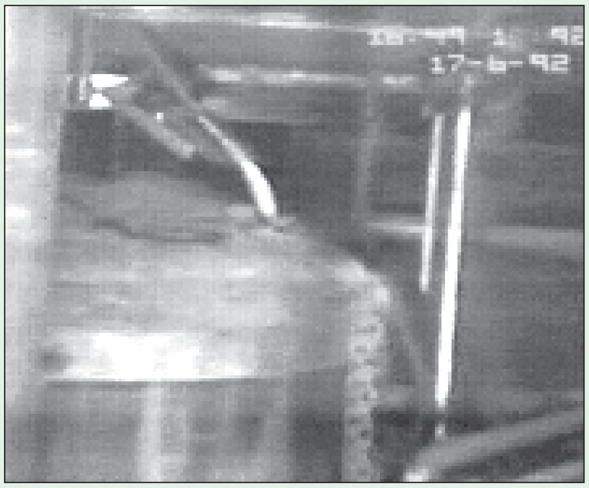
The Over Pressure Relief Devices (OPRD) installed on top of the calandria vessel in RAPS–1 developed a moderator heavy water leak in the year 1992. After identification of the leak using remotely operated CCTV system, the reactor was operated for some time with reduced helium pressure at the top of the calandria but, eventually, the reactor had to be shutdown in early 1994. Many schemes were proposed and a significant amount of development work was required to be carried out before finalization of an acceptable repair method.

The OPRD is located in an inaccessible area with high radiation field. The location of OPRD inside calandria vault and availability of approach through a very small 95 mm diameter opening made it difficult for designing and developing remotely handled tools, tackles and manipulators which could go to the inaccessible locations and perform the blind operations in a satisfactory manner from a point about 3.5 m above on the reactor deck. Several manipulators were developed.

The chosen repair technique consisted of heating of the top of OPRD cover to about 443 K and then pouring of molten Indium into the water seal cavity and subsequently forming it using specially designed tools. After ensuring the reliability of various tools, tackles, and manipulators and after demonstration on a full-scale mock-up, the work of forming a cast-in-situ seal inside the calandria was carried out in February 1997. After normalizing the system, helium leak detection test and helium pressure run down examination showed the effectiveness and sound health of the new seal. It was confirmed that the leak had totally stopped. The reactor was re-commissioned and started on 31st March 1997.

It is to be noted that a similar leak in a reactor in North America was plugged using a different technique and the time,

Visual inspection of calandria vault and cleaning of debris



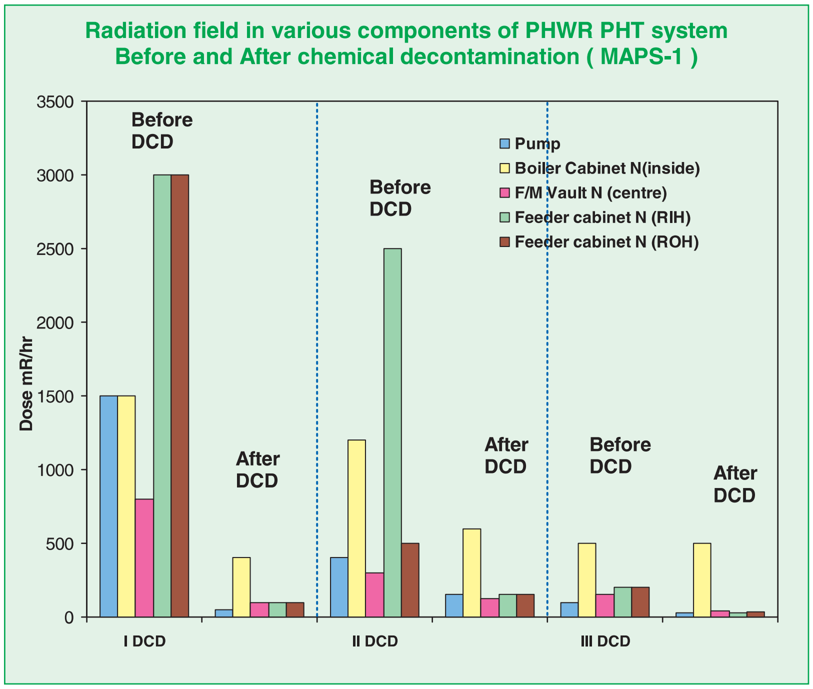
Over-pressure rupture disc repair at RAPS

radiation exposure to personnel and cost involved in that case were much higher than those in the case of work carried out at RAPS.

Control of radiation field by chemical decontamination

Chemical decontamination of the Primary Heat Transport (PHT) system

Over a prolonged period of operation, the radiation field in the PHT system increases due to the leakage of fission products from the fuel elements to the coolant system. In order to minimize radiation dose levels to occupational workers, chemical decontamination processes are used for the removal of the deposited radionuclides. This decontamination process is carried out prior to every annual shutdown in order to minimize man rem expenditure for the plant. Whenever a major maintentance of the PHT is envisaged as in the case of the replacement of a large number of coolant channels, such a decontamination procedure assumes particular importance because the total radiation exposure for such maintenance jobs has to be minimized. Chemical decontamination removes the radioactive deposits by dissolving the deposited oxide layer itself in a controlled manner. In order to do this an acid, a reductant and a complexant are required respectively to dissolve the oxide, reduce the metal ions in the oxide for enhancing the destabilization of the oxide and to solubilise the metal ions by complexation. Usually organic reagents are used for this purpose and they are employed at a concentration level of 1.0 g/lit to minimize base metal corrosion and waste volume; hence the name dilute chemical decontamination. Water and Steam Chemistry Laboratory (WSCL) carried out experimental work in the last few years in order to ascertain quality assurance for dilute chemical decontamination technology in various areas like material compatibility, kinetics of oxide dissolution, method of chemical injection, radiolytic decomposition, ion exchange processing of waste and quantification of the efficacy of the process. Several campaigns were carried out in the PHWRs at MAPS and RAPS in the past decade with good decontamination factors. The formulation presently standardized is based on a mixture of nitrilo triacetic acid (1.0 g/lit) with a provision to regenerate the chemicals through strong acid cation exchanger. The large scale replacement of coolant channels which were carried out at the MAPS and RAPS reactors, mentioned in the



next section, were preceded by successful campaigns of decontamination, thereby greatly reducing the radition exposure to workers.

The final purification and termination of DCD campaign is carried out by mixed bed ion exchanger. The waste so produced is disposed off in a retrievable sub-surface disposal yard called tile-hole which has multiple impervious barriers. The DCD technology developed and standardized for Indian PHWRs has enabled saving of man-rem, foreign exchange and made available a technology that can be used elsewhere in the world.

Coolant Channel Replacement and Upgradation in RAPS-2 and MAPS-2

In India the first 7 PHWRs use zircaloy-2 as the pressure tube material. The remaining PHWRs in operation and others under construction use zirconium-2.5% niobium (Zr-2.5% Nb) pressure tubes, which have considerably superior properties. The pressure tubes contain fuel bundles and heavy water coolant of primary heat transport system. At operating condition, the pressure tubes are subjected to high pressures, temperatures and radiation flux. The zircaloy-2 pressure tubes undergo degradation in material properties caused due to aging phenomenon associated with the effect of pressure,

Table 1. Dilute Chemical Decontamination (DCD) of Indian PHWRs: Performance Statistics

	MAPS #1	MAPS #2	MAPS #1	MAPS #2	MAPS#2	RAPS #1
Date of Campaign	Dec-93	Apr-95	Nov-97	Feb-99	Jan-02	May-02
Duration (h)	19.00	21.50	22.75	20.50	18.5	22
Formulation	EAC	EAC	EAC	EAC	NAC	NAC
IX Rad.waste (m^3)	4.5	5.2	5.0	6.5	7.0	7.0
Fe removed (kg)	233	203	163	254	222	232
^{60}Co (Ci)	346	121	155	46	12.9	40.3
Total activity (Ci)		138			35.8	50.8
DF on Pumps	7- 30	2 – 6	1.3 - 2.7	1.2 - 5	1.3 - 6.0	3 - 5.4
DF on Boiler cabinets	1- 4	7 – 2	1.3 - 10	1.8 - 8.0	2 – 12.5	1.3 – 2.4
DF on Feeder cabinets	7 - 30	10-17	3.3 - 20	4.5 - 15	2 - 7	4 – 5
Dose Averted (MANREM saved)	400	490	300	~300	50% for EMCCR	80% for ISI

temperature, fast neutron flux and contact with hot pressurised heavy water coolant flowing through these channels. The degradation takes place with respect to dimensional changes and material properties. The pressure tubes need to be replaced en masse once it is established that irradiation induced embrittlement causing loss of fracture toughness is at a stage such that the leak before break criterion can not be assured and / or dimensional changes are beyond acceptable levels.

Of the seven PHWRs that have zircaloy-2 pressure tubes, two reactors (RAPS-2 and MAPS-2) have been re-tubed with new pressure tubes of Zr-2.5% Nb material, to ensure extension in life by 30 years. Since operations of cutting the coolant channels and their removal and further re-tubing operations had to be carried out in hostile radiation environment, it required detailed planning to ensure uninterrupted progress of work coupled with low radiation exposure and very low cost. The En Masse Coolant Channel Replacement (EMCCR) has been carried out in four phases viz. defuelling, removal, reinstallation and recommissioning.

During EMCCR of first reactor i.e. RAPS-2, the complete job was divided into several packages and carried out sequentially using labour contracts, with close supervision by skilled NPCIL technicians and supervisors. The entire execution of EMCCR for the second reactor, i.e. MAPS-2, was awarded to a reputed contractor, under NPCIL supervision. For the execution of these works, the manpower was trained with regard to radiological safety and access control to minimise the man-rem expenditure. The crew assigned for specific assignments were given theoretical and practical training on full-scale mock-ups established in the component workshop at site. The EMCCR work has been completed on these two reactors so far. In completing these operations the time period and cost and man-rem were lower than the budgeted estimates.

Coolant channel replacement operation

As can be seen, significant advancement has been made in life management of coolant channels and retubing of pressurised heavy water reactors in India. Simple manual / semi automatic remote operable tooling for use in high radiation area have been successfully developed and deployed. It is assessed that these re-tubed reactors can be safely operated for the next 25 to 30 years.

Moderator Inlet Manifold Failure in MAPS-1 and 2

In MAPS, cold moderator is supplied to calandria through an inlet manifold designed to introduce the heavy water with a very low velocity. In the calandria, moderator gets heated during the process of moderation. Hot heavy water is taken out from calandria through an outlet manifold and cooled and sent back into the calandria. For shutting down the reactor, moderator is dumped from calandria via dump ports into dump tank.

During 1989, failure of calandria inlet manifolds was noticed in MAPS-1 and MAPS-2. Under the circumstances it was not possible to operate the moderator system in normal mode. The reactors were brought to power after carrying out a series of short-term rehabilitation measures. In the new mode of moderator circulation, calandria inlet line was blanked, outlet was converted into main inlet and dump ports were utilized as outlet. However under this mode of operation power was to be restricted to 75% of full power.

Prior to this, video (visual) inspection inside calandria vessel in both the reactors of MAPS was carried out to ascertain health of the calandria internals. An inspection system was devised based on miniature radiation resistant camera. Extention tubes having tilting arms for carrying camera module, lighting and other fixtures were designed and assembled. After performance qualification of these on a full-scale mockup, actual inspection of calandria internals was carried out. The areas covered included moderator inlet manifold, other calandria tubes, rolled joint grooves of calandria tube sheets, dump ports, moderator outlet manifold and adjuster rods. Precise assessment of condition of calandria internals was made. Loose parts of inlet manifold were identified. Size and weight estimates for these parts were also made.

Before the reactor was started again, it was necessary to move the loose parts of inlet manifold (identified during video inspection) to corner pockets inside calandria so that they do not come in the flow path of moderator. With a view to reduce downtime of the reactors to the minimum possible, a scheme based on telescopic manipulator for remote handling of the loose parts was quickly finalised. The limited access through 118 mm diameter lattice bore, ease of insertion and withdrawal during regular operations, safe removal in case of snapped wire ropes of the manipulator and maximum weight of the loose parts to be lifted were taken into account while designing the manipulator. A special pneumatically operated gripper was devised to lift the loose parts. Provision of telescopic arm enabled the manipulator to reach all the loose parts of different dimensions lying at different locations.

As it was not possible to replace or repair the damaged calandria inlet manifolds, various options were considered and analysed in detail to bring the reactors back to rated capacity. One of the best possible options was introduction of moderator inlet spargers. This option involved replacement of 3 calandria tubes by 3 perforated tubes, which introduced moderator inside calandria at low velocities. Sparger Channel with whirlers at the ends is a unique device for introducing moderator into the calandria of a PHWR. The design and development of sparger channel was done by BARC. This option was analysed in detail with regard to feasibility in implementation and safety aspects. Design of spargers was evaluated and optimised by carrying out analysis and conducting a series of experiments.

As implementation of sparger option involved long shutdown, it was decided to carry out the same during En Masse Coolant Channel Replacement (EMCCR). In the meantime, design, analysis, development, and testing were carried out. Elaborate planning and mock-ups were done, and documents were prepared to minimize time, effort, and radiation exposure to personnel during installation of spargers in calandria and moderator piping.

After installation of spargers, testing and commissioning of system was successfully carried out and capability to achieve

Inside view of pressure tube

Loose part of manifold

Sparger channel mock-up test facility at BARC full power was demonstrated. Installation of sparger channels involved very critical activities like machining of calandria tube sheets and rolling of sparger tubes on to calandria tube sheet.

The mission of introducing the sparger channels was accomplished successfully, once again proving India's technological excellence in the field of nuclear power.

Rehabilitation of NAPS-1 after Major Fire

There was a fire in Turbo Generator (TG) area of NAPS-1 on the 31st of March, 1993. In this fire, the major equipment which suffered extensive damage included turbo generator, turbine and its bearings, isolated phase bus duct (IPBD), earthing transformer of TG, excitation transformer, excitation panels and cables in the area below the generator.

The fire, which initiated at IPBD, spread and also damaged the cables emanating from class-III power supply panels leading to non-availability of emergency power supply. This created a station black out like situation. This called for extensive fire safety up-gradation including cable route segregation and provision of proven fire barriers at floor and wall penetrations.

All fire barriers/fire doors were up-rated to 3 hour fire rating. Specifications for firebreaks (coating of fire retardant paint on cables at specified interval) were prepared and firebreaks were qualified by actual testing. These firebreaks have a 30 minutes fire rating. In addition to application of fire retardant paints along the run of cables these firebreaks are applied on entire vertical length of cables and on full length of cables where they are emanating from safety related electrical panels. With this, an additional level of fire protection would be available while fire barriers would prevent propagation of fire from one room to another. To isolate the excitation transformer and the cables a fire barrier wall has been built. Further, cable trays in this area have also been covered by GI sheets.

Hydrogen leakage detection system has also been provided near the generator and isolated phase bus duct connection. The hydrogen addition system which was earlier located near the affected area has also been shifted out of turbine building. A fire hydrant has been additionally provided near the generator. Diverse detection system based on optical detectors has been additionally provided in cable galleries and switchgear room.

Modification in last stage turbine blades (LP turbine) was carried out. Provision of main oil tank and turbine oil tank drain off system was provided in addition to existing CO_2 fire protection system already provided. Provision of high velocity water spray for TG bearing and hydrogen seals has been provided. Provision of emergency hand switches to stop control room and control equipment room air handling units/fans and dampers to avoid ingress of smoke in control room in case of fire have been provided.

Lessons learnt from NAPS-1 fire have been incorporated in all the plants. The plants now meet international industrial and fire safety requirements.

At the second conference on fire safety organized by Nuclear Engineering International in 1997, the paper describing the handling of the Narora fire incident by the plant and organization was applauded. Similar presentations at IAEA meetings and meetings with Regulatory bodies have elicited appreciation of operator action, training, post-accident rehabilitation, design capability and frankness in sharing information. These international fora served as a window to project India's capability in the field of safety, training, and emergency preparedness.

Partial Delamination of Inner Containment Dome in Kaiga-1

A portion of the underside of the inner containment (IC) dome of Kaiga Atomic Power Project, Unit-1, got delaminated and fell down in 1994 when the pre-stressing operation was going on. Based on the recommendations of the investigation committees, the re-engineering of the IC dome was taken up. The re-engineered IC dome was constructed using high performance concrete of M60 grade, which was specially developed for this purpose. The pre-stressing of the inner containment dome was optimized such that the design requirements under load combinations pertaining to construction condition as well as under abnormal and abnormal plus extreme environmental condition at the end of design life are met.

One of the major issues that emerged during the investigation of de-lamination was the design of inner containment dome against radial tension, which arose mainly due to stressing of curved cables, change of membrane stress trajectories due to sudden change of thickness and presence of discontinuity due to un-grouted holes of pre-stressing ducts. Computation of these stresses was made based on the philosophy evolved during re-engineering and radial reinforcements were introduced to carry such tension. The radial stresses developed during the process of pre-stressing were compared with the analytical

Re-engineered inner containment dome under construction

results with the help of vibrating wire strain gauges embedded in the concrete, which indeed showed very good correlation between the theoretical and actual observation made during the process of pre-stressing and proof testing

The ring beam of IC structure houses anchorages of the post-tensioned cables coming from dome and wall. Ring beam being a zone of discontinuity, was analyzed using a detailed 3D shell-solid model consisting of general shell element representing the shell portion and brick element representing the ring beam portion. The most challenging task of such modeling was the simulation of prestressing load in the 3D model.

The pour-sequence of the dome was developed based on the available construction means while considering the effect of shrinkage and heat of hydration. Another important aspects of pre-stressed concrete structure design is the sequence of pre-stressing. A fresh sequence of stressing of pre-stressing cables was developed in an iterative manner so as to minimize development of undue stresses during the process.

In order to validate the assumed design parameters and also to demonstrate proper constructability / concretability, mock-ups were carried prior to taking up construction of the re-engineered IC dome. Some of the mock-up studies were related to determination of co-efficient of friction and wobble of pre-stressing sheath, groutability of long prestressing ducts, concretability of certain critical areas of the dome such as a portion of the thickened area around SG opening, a segment of the IC ring beam, ability to pump M60 grade concrete with micro-silica etc.

Finally, strain measurements in the IC dome were carried out during the process of pre-stressing and proof testing and the measured strains were found to be very close to the analytically predicted values.

PRESSURISED WATER REACTORS (PWRs)

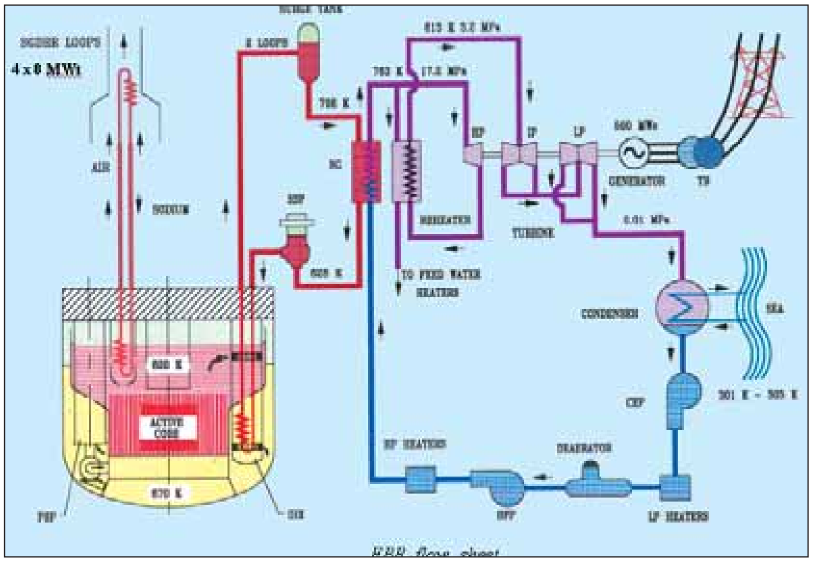
Kudankulam Atomic Power Project (2 x 1000 MWe)

Nuclear Power Corporation of India limited is constructing a nuclear power plant at Kudankulam in collaboration with Russian organization- Atomstroyexport. The project comprises two units each of 1000 MWe VVER type reactors. The entire design of the plant and supply of all the major equipment is in the scope of the Russian Federation while development of infrastructure and project construction is in Indian scope of work. The design of this reactor was assigned to Atomenergoproekt, (AEP) as a nodal agency, the nuclear reactor being designed by Gidropress.

The Kudankulam site cleared for setting up an installed capacity up to 6000 MWe, is located on the coast of Gulf of Mannar, 25 km Northeast of Kanyakumari in Radhapuram Taluk, Tirunelveli District of Tamil Nadu. The nearest town is Nagercoil, which is about 35 km West of Kudankulam village.

VVER is an acronym for “Voda Voda Energo Reactor” meaning water-cooled, water moderated energy reactor. The VVER type reactor belongs to Pressurized water reactor (PWR)

Concreting of core catcher in progress at Kudankulam



FBR flow sheet

family. VVER-412 being supplied to India is an advanced reactor with a double containment and passive safety features. Some of its main passive systems include fast boron injection system, passive heat removal to the atmospheric air, and melt fuel catcher.

The design, construction and operation of the plant meets the regulatory and licensing requirements of Russian regulatory body "GAN" as also India's Atomic Energy Regulatory Board. This would enable incorporation of some of the good practices of India in the design and operation of these reactors of foreign design. The first pour of concrete was carried out on 31st March 2002. Unit 1 is expected to be completed by Dec 2007 and unit 2 by December 2008.

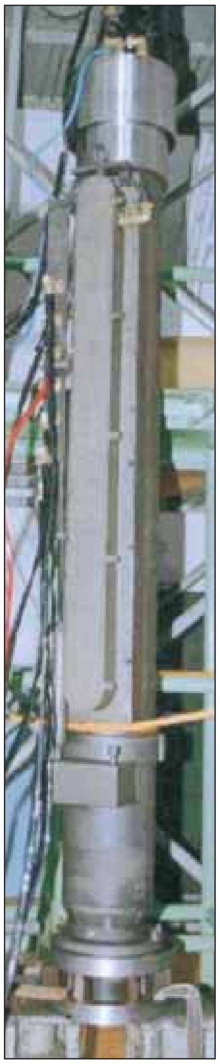
Fast Breeder Reactor (500 MWe) Project

As a logical follow-up to Fast Breeder Test Reactor (FBTR) and keeping in view the need for commercial deployment of Fast Breeder Reactors (FBRs), the design of a Fast Breeder Reactor (FBR) of 500 MWe was initiated in the 80's as a step towards technical demonstration of FBR for power generation. The reactor power was chosen as 500 MWe considering the indigenous Turbine Generator availability. Considering the designs of Phenix and Super Phenix FBR's in France, PFR in UK and BN-600 in Russia, the PFBR layout was selected to be of pool type. In this layout, the core, primary pump and Intermediate Heat Exchanger(IHX) are located in a single large vessel, with an inner vessel to separate the hot and cold pools, unlike loop type where each of these components is a separate entity joined by pipes. The major advantage is the higher thermal inertia of primary sodium resulting in slower temperature rise in case of cooling failure, giving enough time to shutdown the plant. Another advantage is the absence of active primary sodium leak outside the vessel.

A first reference design was realized in 1983. The design was conceived to accommodate either mixed carbide or mixed oxide fuels. In this design there were four primary pumps, 8 IHXs, 4 secondary loops each having 3 SG units. Each SG unit comprised of an evaporator, superheater and reheater. The steam from the superheater after passing through the high pressure turbine unit needs reheating and this reheating was to be carried out by a part of the hot secondary sodium stream from IHX.

Materials Development

The development of indigenous materials for core structures, SG and other components of the reactor is a vital activity for the success of the programme. IGCAR has set up a comprehensive range of facilities for studies on the metallurgical properties of variety of materials. Systematic and exhaustive studies have been carried out to understand these properties for materials to be used in FBTR and materials selected for PFBR. Processing maps and instability maps have been generated for the materials used in reactor systems such as stainless steel 304, 316, 304 LN, 316 LN, modified 9-Cr 1-Mo steel etc. These materials are exposed to sodium at different operating conditions and the effect on corrosion of these metals have been studied in experimental facilities. Also, to understand the behaviour of carbon in the secondary circuits of PFBR, wherein 316 LN and modified 9-Cr 1-Mo steels are involved, a bi-metallic sodium loop has been built and operated. Weldability of steels is another important area, where extensive R&D has been carried out. Studies have been carried out to develop a complete understanding of the mechanical behaviour of welds under monotonic and cyclic loadings. To study mechanical behaviour such as low cycle fatigue and crack growth and creep-rupture in the temperature range of 823 to 923 K, two sodium facilities were constructed and operated. The initial results from these tests have validated the performance of indigenous materials planned to be used in PFBR.



Diverse Safety Rod Drive Mechanism (DSRDM)

Manufacturing Technology Development

Though most of the critical components for FBTR were manufactured indigenously, it was felt there could be some problems when large scale components are to be manufactured for PFBR. In view of this a programme of manufacturing technology development for components like steam

Main vessel under fabrication for FBR

"…In these 30 years, we had the opportunity to develop large vertical bowl type circulating water pumps in stainless steel construction with radiographic quality castings for MAPS, NAPS, KAPS etc… We were the first to introduce in India a condensate extraction pump with double suction first stage impeller for extra low NPSHR in Tarapur. We have successfully developed indigenously the canned motor pumps in ammonia towers for Heavy Water Board and large capacity 200 KW canned motor pumps for moderator duties in reactors. We are the ones who successfully developed sea water handling pumps for test reactors like Dhruva. We successfully made one lot of five fuelling machine heads. In power station requirements, we have import substitutions by developing new pumps like auxiliary boiler feed, reheater drain pumps, canned motor pumps in many hazardous applications. This has been possible because of the pragmatic approach the DAE has always taken. Where any other customer would have gone for developed and proved equipment, DAE has always shown willingness to go for indigenous development. They carried the process of partnering to the hilt. The amount of hand holding they did, the encouragement they gave to the developer, the pains they took for the technology, knowledge, experience and skill transfer is commendable..."

Sanjay C. Kirloskar, CMD,
Kirloskar Brothers Limited, Pune

generator, main vessel, control rod drive mechanism, roof slab, grid plate, transfer arm, fuel cladding tube and hexagonal wrapper were undertaken in cooperation with selected Indian manufacturers. Suppliers of raw materials such as plates, SG tubes, IHX tubes, forgings and welding consumables have been developed. The manufacturing technology for pump components such as shaft, impeller, hydraulic bearings etc. has been developed indigenously. All the sub-components such as long shaft have been tested for performance. Raw material required for making the components is available in the international market and not presently affected by embargo. Nevertheless, thrust was given to indigenous material productionand today all required grades of stainless steel for sodium systems and ferritic steel for SG are available from indigenous sources. Experience in manufacturing technology development has boosted our confidence in manufacturing components for FBR.

Engineering Development

The Programme of engineering development for FBRs commenced in the 70s. In view of the collaboration with CEA, France, for the design of FBTR, there was minimal engineering development. The focus was on sodium purification and safe handling of sodium, testing of sodium-to-sodium and sodium-to-air heat exchangers. By 1985 when FBTR went critical, the experimental program had matured enough to tackle development works in support of the 500 MWe Prototype Fast Breeder Reactor (PFBR). The approach to R&D has been to develop analytical/numerical models in 1, 2 and 3 dimensions as a first step. From these studies the important parameters that have influence on the design/process are identified. Experiments were then planned to understand impact of the important parameters. In this way the number of experiments needed for design validation were optimized. A twofold approach was adopted for carrying out the experiments. All experiments requiring liquid sodium facilities and those experimental facilities using water and those required to be retained for a considerable amount of time were planned as inhouse R&D works. The other R&D works were carried out in R&D institutions like the Fluid Control Research Institute, Palghat or industries like M/S Kirloskar Bros. Ltd.

Major facilities built and operated include Large Component Test Rig (LCTR) which is a test facility with 80 t sodium hold up, Sodium – Water Reaction Test Facility (SOWART), Steam Generator Test Facility(SGTF) of 5.5 MWt, 500 kW Sodium Loop, etc. Besides these many small and medium size loops for studies in liquid metal corrosion, liquid metal heat transfer and calibration of sodium instrumentation have been built. The design, construction and operation of these facilities have given enough confidence in the design of heat transport systems for reactors and other facilities.

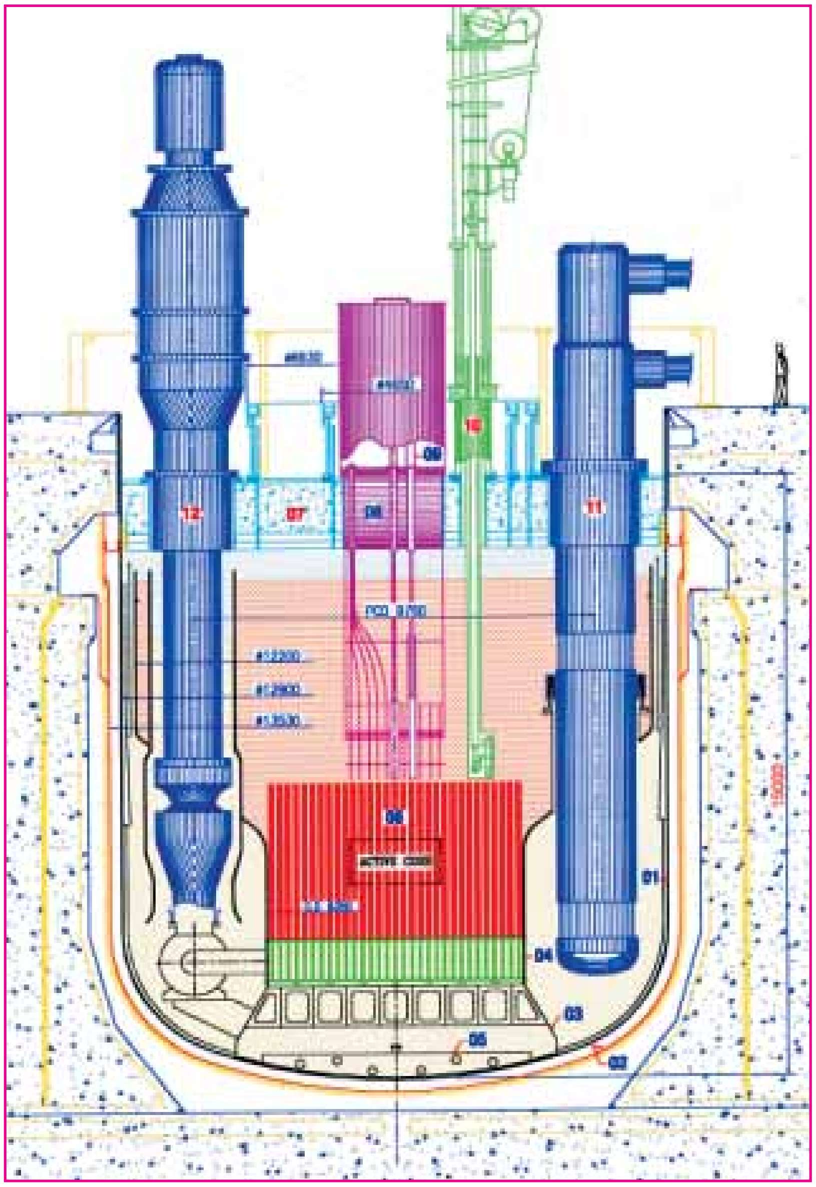
Reactor Assembly

Arrangement of the primary circuit of PFBR is typical of a pool type fast reactor, in that the primary heat transport components are submerged in primary sodium. The loosely coupled primary pumps and IHXs are contained respectively in

Rig for testing of large components

the cold and hot sodium pools separated by the inner vessel. All the submerged components are subjected to strong temperature gradients and fluid-structure interaction.

To study aspects such as heat transfer from hot sodium to the upper cover gas regions, temperature distribution in critical areas and convection currents in annular spaces in cover gas region of the reactor systems, experimental studies have been carried out in LCTR. Regeneration of the FBTR secondary cold trap loaded heavily by hydrogen, diffusing from water in the SG



PFBR reactor assembly

(**Legend**: 01 - Main Vessel, 02 - Safety Vessel, 03 - Core Support Structure, 04 - Grid Plate, 05 - Core Catcher, 06 - Core, 07 - Top Shield, 08 - Control Plug,09 - Control & Safety Rod Drive Mechanism, 10 - In-Vessel Transfer Machine,11 – Intermediate Heat Exchanger, 12 – Primary Pump & Drive)

through the tube wall has been demonstrated. Also in the regions above the free level of sodium in the reactor, large convection currents are set up. This carries large sodium aerosols to the cooler regions at the top of the reactor and also causes temperature dissymmetry along the circumference in large diameter parts. To understand and to reduce / avoid the temperature difference experiments were carried out and methods to successfully overcome problems identified.

From the point of view of flow induced vibration, the assessment of flow patterns and velocity distribution near the control plug and IHX are needed. Assessment of free level fluctuations and cover gas entrainment in the coolant are needed as input for design analysis. Studies have been carried out using water in transparent models of the PFBR primary circuit and some insight has been gained regarding the the velocity distribution in the hot pool. Results with water can be easiliy transposed to reactor design using similitude criteria. The thermal baffle in the main vessel cooling path is provided with a weir crest. The shape of the weir-crest is optimized such that there is no flow separation and the depth of gas penetration is minimum. Experiments performed have validated the design.

Due to the fact that velocity measurements would be more accurate in larger scale models and also the need for larger scale model for study of gas entrainment, a ¼ scale model of reactor assembly (SAMRAT) was designed and erected. This is planned to be used as a permanent model to conduct various thermal hydraulic and flow induced vibration studies, such as optimum location of core monitoring thermocouples, thermal striping of control plug, vibration behaviour of inner vessel and thermal baffles, flow pattern and temperature profile in hot pool, free level fluctuation, gas entrainment etc. The model has been in use since its commissioning in 2003 and has given a very good insight into the velocity patterns in the hot pool especially in and around IHX's.

The flow of sodium through different zones of the PFBR core is varied to obtain maximum outlet temperature. Orifice plates and other flow restrictions placed in the foot of the subassemblies achieve flow zoning. Different combinations of flow restrictions were tested with water in a Hydraulic Test Rig and the results transposed to that of sodium using similitude criteria. From the results cavitation free flow restrictions were

SAMRAT - ¼ scale model of PFBR reactor assembly

selected. Testing of a full scale dummy fuel subassembly (overall height 4.5 m, pin bundle 217 pins x 2.5 m long) for pressure drop vs. flow characteristics in water has given the input for head to be developed by the primary pump. Hydraulic tests on a truncated 19 pin assembly have been carried out to study the influence of fuel pin spacer wire pitch on both pressure loss and flow induced vibrations and this has led to improved understanding of the fuel subassembly hydraulic design. A 1/3 scale model of the grid plate with simulation of individual subassembly flow through restrictors has been tested at the Fluid Control Research Institute, Palakkad. Studies undertaken using air as simulating fluid have confirmed the adequacy of design.

"…MTAR was started in the year 1970 primarily to undertake developmental jobs for DAE. It is worthwhile to mention our interactions particularly with IGCAR, which helped nurturing different important industrially viable technology solutions for changing requirements of IGCAR (Control and safety rod drive mechanism, Diverse safety rod drive mechanism and technology development of grid plate for PFBR). It is not out of place to mention here that MTAR engineers interaction with engineers of IGCAR was technically intense, analytical and eventual agreements in the final design determination. In our own humble way, our contribution to PFBR activity is immense and was well received by IGCAR engineers…."

P. Ravindra Reddy, Director,
MTAR technologies Pvt. Ltd., Hyderabad

Absorber Rod Drive Mechanism

Two types of absorber rod mechanisms of diverse designs are to be used for PFBR. While the Control and safety Rod drive mechanism (CSRDM) is used for startup, control and shutdown, the Diverse Safety Rod Drive Mechanism (DSRDM) has only the safety function. DSRDM utilizes a Curie point magnetic switch, a passive safety device which gets demagnetized when sodium temperature crosses 893 K , and the control rods drop into the core. Full scale prototype CSRDM and DSRDM have been subjected to air testing. The CSRDM has already undergone sodium testing and DSRDM will be tested shortly.

Pump Development

Primary Sodium Pumps circulate the sodium at 670K through the reactor core thereby extracting the nuclear heat generated in the core. These pumps are also in the technology denial list. The pump size should be close to that of the IHX as otherwise it will influence the size of the reactor vessels. Development

Electromagnetic pump developed indigenously work on primary sodium pumps involved extensive hydraulic and cavitation testing of a 1 /2.75 vertical model in water. Cavitation testing involved visual cavity length measurement under stroboscopic lighting, paint erosion test and buffed SS specimen erosion test. Similar studies were carried out for secondary sodium pump also. These development works were carried out jointly with M/S.Kirloskar Bros. Ltd. A Test Rig was built to study behaviour of primary sodium pump rotor assembly and to validate the spherical seat support concept, hydrostatic bearing leak flow and the performance of all rotating parts like seals and coupling.

For low flow rates, electromagnetic pumps are used even though they have poor efficiency compared to centrifugal pumps, due their non-intrusive design. Small capacity (20 cu.m/h) sodium pumps of Flat Linear Induction type (FLIP) were developed to cater to pumping requirements of experimental sodium loops and reactor auxiliary circuits. Likewise, annular linear induction pumps (ALIP) of capacities 5 and 170 Cum.h were designed. The first prototype ALIP underwent performance tests in sodium exhibiting robustness of its design. Further a DC conduction pump that can be immersed in sodium was designed, fabricated and tested for use in the FBR.

Fuel Handling Components

Transfer arm is used for handling the core assemblies in the reactor during fuel handling campaign. It is a 25 m high unit. There is utmost need to check the operation of this machine in sodium before installing it in the reactor. With this in view the full size mechanism was fabricated, assembled and tested in air at room temperature. Operations such as gripping, lifting, moving to another radial position, lowering, releasing the subassembly, positioning etc. have been checked. The mechanism has high reliability and safety features. The sodium test facility for testing transfer arm at high temperature is setup in LCTR. The sodium vessel holds forty cubic metre of sodium. The Inclined Fuel Transfer Machine (IFTM) performs the role of taking the irradiated subassemblies from inside the reactor vessel to external storage and further actions towards reprocessing. The design of this machine is a complex one. This will be taken up for sodium testing after the testing of transfer arm is completed.

Intermediate Heat Exchanger

The pressure loss characteristics and flow induced vibrations in the IHX tube bundle have been studied using a full scale 60 deg. sector model in water. Assessment of flow distribution on the tube side to minimise temperature differences at tube outlet was made with devices tested in air. In addition a mixing device has been designed and tested in air. During non-availability of an IHX its inlet window in the hot pool is closed by a sliding sleeve valve. Theoretical estimate of leakage flow proved difficult and a full scale model was tested in water and leakage found to be acceptable.

Steam Generator

Steam Generator (SG) is such a critical component that it is described as Achilles' heel of a fast reactor plant. Though operating experience of FBTR SG has been good, it is difficult to extrapolate its design to large size. FBR uses a vertical, shell and tube SG, with hot sodium on the shell side and water/steam at 175 bar and 525°C on the tube side. The tube material is Modified 9 Cr-1 Mo. The design of the steam generator takes into account sodium-water reaction and resistance of tubes to its effects, adequacy of heat transfer area and reduction of excessive heat transfer margins to reduce the cost of SG, flow instability problems related to once through type design, flow induced vibrations and several manufacturing considerations such as inspection of long tubes (23 m) by eddy current methods, tube thermal expansion design, tube-tube sheet joint and so on.

The velocity at tube bundle entry is important from flow induced vibration considerations. Studies of velocity distribution in the inlet plenum and flow induced vibration were conducted on a sector model and cylindrical model of SG. The velocity distributions matched well with 3D code predictions and no flow induced vibrations were noticed upto 120% of full flow.

There is enough conservatism in the design to ensure safety. Conservatism can be reduced by carrying out tests on an instrumented SG. With this in view, a major test facility with a steam generator of 5.5 MWt which simulates all the conditions of FBR SG tube, was designed and constructed at IGCAR. Experimental studies with this facility are in progress. The data obtained will be used to validate predictions of multi dimensional codes and also to provide actual operation data under sodium

Steam generator test facility

to design cheaper SG.

Sodium reacts readily with water or steam to form sodium hydroxide and hydrogen. This reaction is highly exothermic. The reactions of water / steam and sodium have implications in the design, material considerations and protection system. The implications in the SG design arise due to high temperatures produced in the reaction zone, propagation of leaks in the same tube or adjacent SG tubes by material erosion / wastage and due to possible high pressures generated in the sodium side of the steam generator. To detect the sodium – water reaction at an early stage, hydrogen monitoring instruments such as Diffusion Type In-Sodium Hydrogen Meter and Cover Gas Hydrogen Meter were developed and are being used in FBTR. Considerable work has been done in the area of acoustic leak detection method, wherein the acoustic noise produced during the reaction is detected by a piezoelectric sensor, analysed on-line and alarm is initiated. Another method of detection by observing the presence of hydrogen bubbles in the electromagnetic flowmeters installed downstream of the SG has also shown promise. Experimental studies in this area are carried out in the specially constructed Sodium – Water Reaction Test Facility (SOWART). During sodium-water reaction, the wastage of the leaking tube and the tube opposite to the leaking tube takes place due to the high temperature and corrosive reaction products in the vicinity of the reaction zone. Even though data is available regarding the steam generator materials in published literature, wastage in the SG material (modified 9 Cr-1 Mo steel) is being studied in SOWART facility to generate indigenous data.

Sodium Instrumentation

Sodium being metallic, it has high electrical conductivity which is exploited in the design of sensors for sodium. Flow is measured by permanent magnet flow meters and eddy current flow meters. Large numbers of such flow meters were designed and are in use. Measurement of sodium level is carried out by resistance type and mutual inductance type level probe. The technology for such instruments has been passed on to Indian industries to enable them to use such instruments in other liquid metal systems like aluminum foundry. Sodium being opaque, viewing the internals in the reactor systems is not possible by conventional methods. Ultrasonic techniques for viewing objects under sodium have been developed and demonstrated in FBTR. This required fabricating sodium immersible ultrasonic sensors to work upto 473 K. These were not, however available commercially. The intricacies of sensor construction involved brazing of pizeoelectric crystal to nickel diaphragm, leak-tight body and arrangement for signal transmission. A decade of concerted efforts in this direction have borne fruit. Sensitive leak detection methods such as sodium ionization type leak detectors have also been developed with the detector sensitivity that responds to low sodium concentrations in the range of nanograms /cc. Many of these sensors and instruments are tested and calibrated in a specially designed and constructed sodium system called ‘SILVERINA’ which was commissioned in 1996 at IGCAR.

Sodium ionization detector

Sodium Fire, Safety

When liquid sodium leaks from the system, it catches fire, due to reaction with oxygen in the air. Even though this reaction is less exothermic than petrol fire, it produces a very dense opaque smoke thus reducing visibility. Much work has been carried out to understand the behaviour of sodium fire and methods to extinguish the same. Fire in a pool of sodium and from a hot spray of sodium was studied to understand the effect on temperature of the surroundings and the nearby structural materials.

In the event of a sodium fire, damage to the components and the equipment has to be reduced by suitable design measures. Even though leak detectors have been developed to detect minute leaks of sodium at an early stage, the extinguishers to be used have been validated and deployed. To contain a large leak, collection trays are kept underneath the systems carrying sodium, so that any sodium that leaks collects in these trays, which aid in self-extinguishing of the fire by oxygen starvation. Such trays have been developed and tested. They are of a corrugated shape with entry holes precisely sized and spaced so that almost all sodium that leaks from the system is collected within, with little burning. Special safety apparel is used during handling of sodium. Concrete is used in construction of the reactors. In the event of an accidental leak of sodium over concrete, the sodium-concrete reaction takes place leading to exothermic reaction of sodium with the water content in concrete and liberation of hydrogen. Development of sodium resistant concrete and schemes for protection of concrete surfaces have, therefore, been carried out.

Removal of sodium from Components

The components in sodium systems need to be cleaned free of sodium before attempting their repair, reuse or dismantling. Various cleaning techniques employed are alcohol dissolution, water vapour-nitrogen process, water vapour - CO_2 - nitrogen process and vacuum distillation process. In all these processes hydrogen is one of the reaction products, which if present in concentrations above 4% in air can form an explosive mixture. Large components such as sodium pump, heat exchangers etc. are cleaned by water vapour- CO_2 in inert gas atmosphere. This method is preferred over the moist-inert gas cleaning method where the sodium hydroxide formed could lead to caustic stress corrosion of the component during reuse. The CO_2 present in the cleaning mixture reacts with sodium hydroxide to form sodium bicarbonate and sodium carbonate which do not cause caustic stress corrosion. These methods were studied and adapted in Engineering Loops and in FBTR. Based on results obtained from experimental methods, the process for PFBR components, fuel sub-assemblies etc. were finalized.

Structural Integrity Assessment

PFBR components operate at high temperature (820 K) with a large temperature gradient (150 K). High temperature failure modes such as ratcheting, creep and fatigue damage mainly under thermal loadings decide the plant design life. Tests were conducted on models of shells subjected to axially varying temperature gradients to simulate thermal ratcheting which happens on the vessels subjected to free level variations of sodium. Creep-fatigue damage assessment methodology was validated based on tests on specimens having component features. The rolled and welded tube-to- tubesheet joints of IHX were qualified by carrying out tests for pull out strength after simulating creep relaxation. The SG tubes made of modified 9Cr-1Mo were tested at high operating temperature under internal pressure with axial deformations to simulate the creep rupture. All these tests have provided results that assure the operation of components at high temperature, for the design life of 40 yr.

Buckling Investigations

Reactor vessels are basically large diameter thin walled shells, which are prone to buckle. Geometrical imperfections imposed during manufacturing stage and subsequently by progressive deformations during operation, plastic deformations, seismic loadings and their dynamic effects reduce the buckling strength significantly. Buckling of vessels have been investigated extensively with the help of numerical analysis and validated thoroughly based on tests on scaled down models. Tests were conducted on 1/30 and 1/18 scaled down models of main vessel under bending moment and shear forces and on 1/13 models of inner vessel under internal pressure and concentrated loads transmitted through stand pipes.

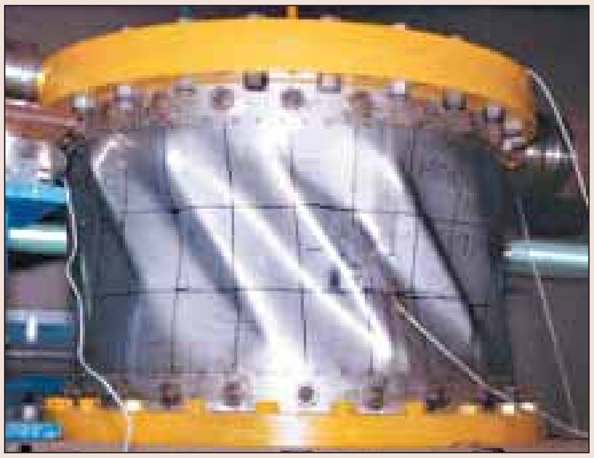
Fracture Assessment

Fracture assessment including analysis for Leak Before Break (LBB) justification and crack propagation behaviour has been studied on 316LN and G91 plates by subjecting them to bending loads. Also a series of fatigue and fracture tests conducted on 1/5 scale models of primary sodium pipe and SG tube bends, subjected to internal pressure and bending moment, confirmed the applicability of RCC-MR:A16(2002) rules and validated the design. Fatigue and fracture tests were conducted on typical large size tees, bends and SG nozzle junctions with the objective of demonstrating LBB.

Structural Dynamics

Structural dynamics problems are very critical in FBR because of the use of thin walled large diameter shell structures with associated fluid effects. Typical examples are pump induced as well as flow induced vibrations, seismic excitations, pressure transients in the intermediate heat exchangers and piping in case of a large sodium water reaction in SG and Core Disruptive Accident (CDA). The structural dynamics problems have been analysed in detail to demonstrate the structural integrity. A few critical aspects have been validated through experiments.

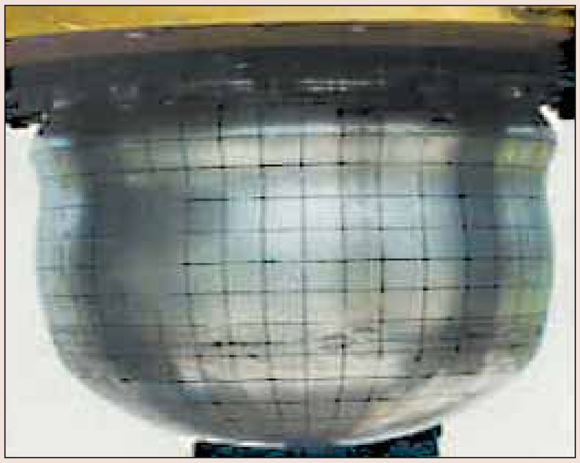
The experimentally measured dynamic behaviour of FBR core subassembly models (CSA) subjected to simulated base excitations matched well with predictions. The seismic response of Reactor assembly components predicted by computer codes, were validated by tests on shake table. A series of tests conducted on scaled down mockup of main vessel with its internal structures has helped to have a thorough understanding of the entire mechanical consequences involved in a CDA, apart from validation of the predictive tools. The CDA experimental programme, carried out at Terminal Ballistic Research Laboratory, Chandigarh, involves a series of tests (TRIG) to characterize explosives, tests on 1/30 scaled down MV models without internals and 1/13 scaled down reactor assembly mockups with internals. From the test series, the maximum potential of main vessel upto rupture was found to be 1200 MJ (the design value is 100 MJ), structural integrity of IHX and decay heat exchangers is assured and sodium release to reactor containment building has been quantified.



Shear buckling of main vessel model

Buckling of inner vessel model

Seismic tests on reactor assembly model



A tested vessel of TRIG series

FBR -500 MWe Project Design - Final option

The manufacturing technology development, engineering development and component testing over a long period have given a wealth of confidence on the design of PFBR. Fast reactors would be acceptable only if they were cost competitive with thermal and hydro plants in operation. Besides they should be constructed in similar timeframes for acceptability. It became clear that larger size components would minimize fabrication time. At the same time, there were similar cost reduction exercises taking place in other countries interested in fast reactor and this provided a good fillip, especially to reduced conservatism in design which is backed up by latest design rules for high temperature design. Also the difficulties experienced with sodium reheat in PHENIX and PFR reactors led to the option of reheating HP turbine outlet steam by Live/ bled steam.

The above resulted in evolving a compact design for FBR with 4 IHX's , 2 primary pumps, 2 secondary loops with 4 integrated SG modules. With improved knowledge gained in the structural damage assessment, data regarding material properties and codification of high temperature design and experiences of life extension in fossil and nuclear power plants employing similar materials, confidence has been gained in designing the FBR plant to work for 40 years instead of 30 years without any economic penalty. This has also helped to reduce the projected unit energy cost.

For higher overall thermal cycle efficiency, it is essential that higher operating temperatures be chosen for the plant. The systematic study undertaken based on improved knowledge base, has led to the enhancement of the operating temperatures for the primary sodium by 17 K i.e. the reference core inlet / outlet values 653 / 803 K are increased to 670 / 820 K. The corresponding secondary sodium inlet / outlet temperature for the intermediate heat exchanger (IHX) is 628 / 798 K. These changes also resulted in an increase of steam temperature at TSV by 10 K i.e. from 753 K to 763 K. The improved cycle efficiency with higher temperature difference across HX results in reduced unit energy generation cost (UEC).

Excavation of nuclear island of FBR in progress at Kalpakkam

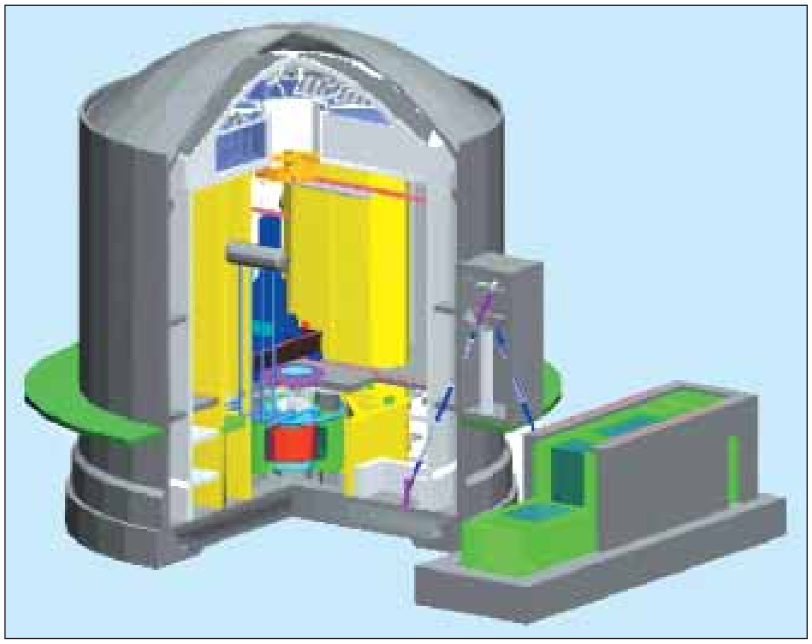
With respect to the layout of the plant, the endeavor has always been to achieve a compact layout to make it more economical. In the present design Reactor Containment Building (RCB), SG building and fuel building are kept on a common raft so as to minimise risk of secondary sodium pipe leak and fuel handling incidents in case of an earthquake. Also, the safety related building should be kept outside the trajectory of any missile from turbine. Construction of this 500 MWe FBR has already commenced in 2004.

THE NEXT GENERATION OF POWER REACTORS

India has vast resources of thorium. It is the endeavour of DAE to develop suitable technologies for utilization of this resource to bridge the gap between the country's energy needs and energy resources. Towards this goal, development work has been initiated on reactor systems for utilization of thorium and breeding of fissile material from it.

Advanced Heavy Water Reactor (AHWR)

Development of AHWR was started in the late eighties. The motivation was to design and develop an indigenous reactor system, which is economically sustainable and environment friendly, uses indigenous fuel resources – particularly thorium - effectively, is suited to serial building in large numbers and meets higher levels of safety by relying on inherent and passive safety features to an optimum extent. Broad requirements for an Indian advanced nuclear reactor were worked out and presented for the first time at an IAEA meeting held at Chengdu, China in October 1990.



Artist's view of proposed AHWR

Major design features

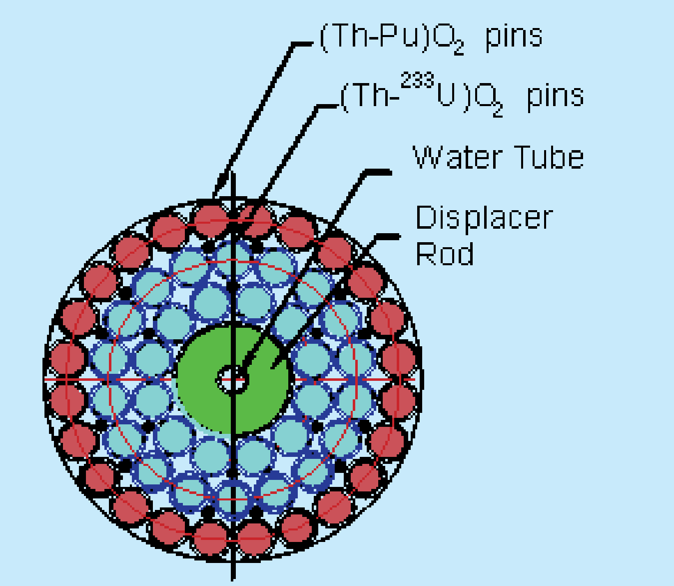
AHWR is a 300 MWe, vertical, pressure tube type, boiling light water cooled, and heavy water moderated reactor. The reactor design incorporates a number of passive safety features and is associated with a fuel cycle having reduced environmental impact. Several of its design features will enable reduction of capital and operating costs. The reactor system is designed to achieve large-scale use of thorium for the generation of commercial nuclear power. This reactor will produce most of its power from thorium, with no external input of ^{233}U, in the equilibrium cycle.

AHWR employs natural circulation for cooling the reactor core under operating and shutdown conditions. All event scenarios initiating from non-availability of main pumps are, therefore, excluded. The main heat transport system consists of a common circular inlet header from which feeders branch out to the coolant channels in the core. The outlets from the coolant channels are connected to tail pipes carrying steam-water mixture from the individual coolant channels to four steam drums. Steam is separated from the steam-water mixture in steam drums, and is supplied to the turbine at a pressure of 70 bars and temperature of 558 K. The condensate is heated in moderator heat exchangers and feed heaters and is returned to steam drums by feed pumps. Four downcomers connect each steam drum to the inlet header.

Safety Features

AHWR is designed to have a negative void coefficient of reactivity and its passive safety systems are designed to work on natural laws. A large heat sink in the form of a Gravity Driven Water Pool (GDWP) with an inventory of 6000 m^3 of water is located near the top of Reactor Building. Removal of heat from the core is by natural circulation. Emergency core cooling system injects water directly inside the fuel cluster. The reactor is provided with two independent shutdown systems. The Passive Containment Cooling System provides long term containment cooling following a postulated Loss of coolant Accident (LOCA). The GDWP serves as a passive heat sink yielding a grace period of three days. The core gets submerged in water long before the end of this period.

Consistent with the approach used in standardized Indian PHWRs, AHWR is provided with a double containment. For containment isolation, a passive system has been provided in AHWR. The reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height. In the event of LOCA, the containment pressure acts on the water pool surface and drives water, by swift establishment of siphon, into the U-bends of the ventilation ducts. Water in the U-bends acts as a seal between the reactor containment building and the external environment.



Fuel Cluster Cross-Section

Thorium Based Fuel of AHWR

The AHWR fuel cluster contains 54 fuel pins arranged in three concentric circles surrounding a central displacer rod. The inner two circles contain thirty (Th-^{233}U) O_2 fuel pins and the outer circle contains twenty-four (Th-Pu) O_2 fuel pins. The central rod contains dysprosia in zirconia matrix. The fuel cluster also incorporates a water tube for the spraying of emergency core cooling system water directly on fuel pins during a postulated LOCA. AHWR fuel is currently designed for an average burn-up of 24,000 MWd/t. Its design makes it amenable for reconstitution, if desired to facilitate a further extension of burn-up in the (Th-^{233}U) O_2 fuel pins in future.

Some Distinctive Features of AHWR

The design of AHWR has several distinctive features as compared to the PHWR systems. Some of these features are: Elimination of high-pressure heavy water coolant resulting in reduction of heavy water leakage losses and eliminating heavy water recovery system; recovery of heat generated in the moderator for feed water heating; elimination of major components and equipment such as primary coolant pumps and drive motors, associated control and power supply equipment and corresponding saving of electrical power required to run these pumps; shop assembled coolant channels with features to enable quick replacement of pressure tube alone without affecting other installed channel components; replacement of steam generators by simpler steam drums; use of non-nuclear grade equipment for all systems required to back-up the passive systems; higher steam pressure than in PHWRs; production of 500 m^3/day of demineralised water in multi-effect desalination plant using steam from LP Turbine; one hundred year design life of the reactor; and a design objective of requiring no exclusion zone on account of its advanced safety features.

The AHWR is a product of systematic endeavour to realize these broad requirements effectively and efficiently. The design has been developed to a level needed for initiating a detailed safety review followed by construction of an AHWR based nuclear power plant. In the process of developing the reactor design, a large number of design options were evaluated and suitable options were selected on the basis of rigorous analysis and experimental work.

Compact High Temperature Reactor (CHTR)

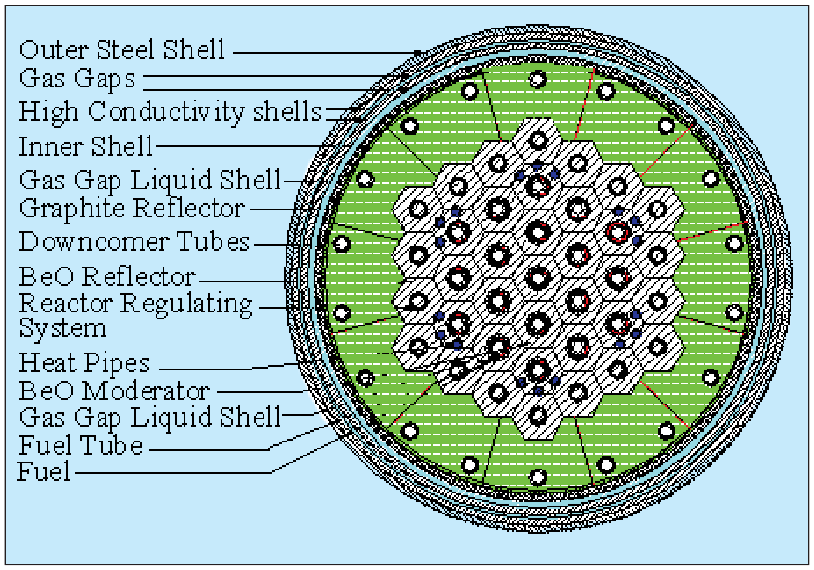
The motivation for development of CHTR is two fold. One is to meet the need for a source of heat at temperatures close to 1000° C for application in production of hydrogen for use as clean fuel and the other is to develop compact nuclear power stations for supplying non-grid-based electricity in remote areas, which are difficult to access. This reactor is being developed as part of the "National agenda on utilization of nuclear technology for developmental applications"

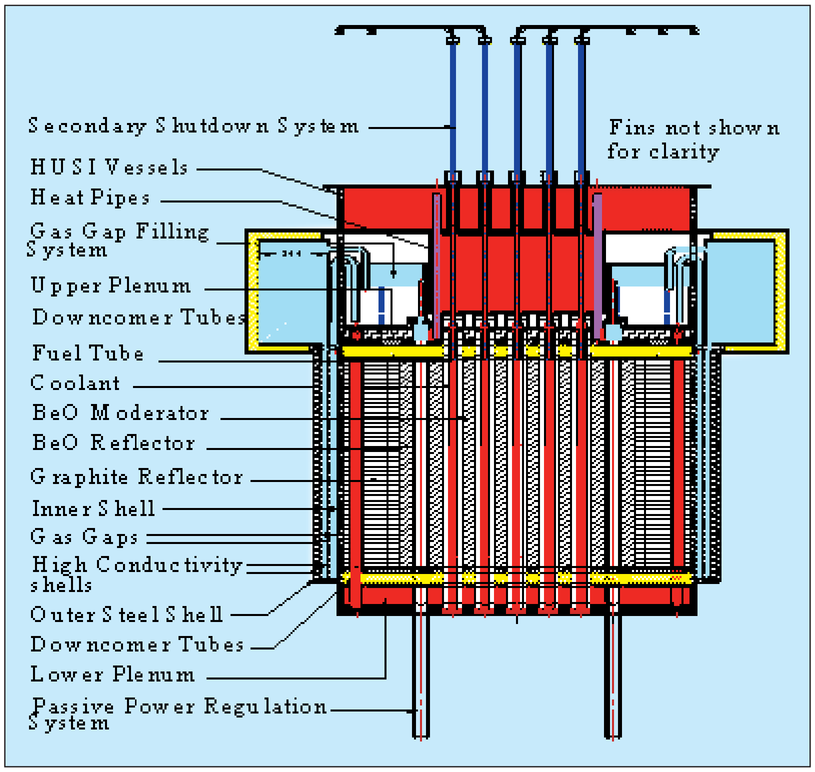
The conceptual design of CHTR envisages use of thorium-based fuel, BeO moderator, BeO/Graphite reflector and molten lead based coolant. In order to understand and develop the technologies involved, a small reactor with 100 kWt power and coolant outlet temperature of 1000° C is being designed. CHTR design provides for passive means of heat removal in normal and accidental condition. This reactor also has passive means of power regulation and shutdown. The conceptual design of the reactor has been completed and feasibility of developing the major systems has been established. Detailed design, computational modeling and analytical studies of systems are in progress. Setting up of experimental facilities to study various systems is in progress. It is estimated that CHTR, would consume 0.5 kg of ^{233}U in 5 years, when used for production of hydrogen by the iodine-sulfur process (η=47%) and would produce approximately 0.3 g of hydrogen per second (9.5 tons per year).

Some of the major design features of CHTR are: use of thorium based fuels with low fissile inventory; passive core heat removal by natural circulation of liquid heavy metal coolant; passive rejection of entire heat to the atmosphere under accident condition; passive power regulation and shutdown systems; and compact design to minimize the weight of the reactor.

General Description of the Design

The reactor core consists of nineteen prismatic beryllium oxide (BeO) moderator blocks which have centrally located fuel tubes made of graphite. Each fuel tube carries fuel inside 12 equi-spaced longitudinal bores made in its wall. The fuel tube also serves as coolant channel. Eighteen blocks of beryllium oxide reflector surround the moderator blocks. These blocks have central holes to accommodate passive power regulation system. This system works on temperature feedback,





Lay out of components in CHTR

and in case of rise of coolant outlet temperature beyond design value, inserts negative reactivity inside the core. Graphite reflector blocks surround these beryllium oxide reflector blocks. This part of the reactor is contained in a shell of a material having high temperature resistance. Top and bottom closure plates of similar material close this reactor shell. Plenums for hot and cold coolant are provided at top and bottom of the reactor core respectively.

Nuclear heat from the reactor core is removed passively by a lead based liquid metal coolant which flows due to natural circulation between the plenums, upward through the fuel tubes and returns through down corner tubes. The shell is surrounded by two gas gaps which act as insulators during normal reactor operation and reduce heat loss in the radial direction. There is an outer steel shell, which is continuously cooled by circulating water. This shell has been provided with fins to improve its heat transfer capabilities. A system is provided to fill these gas gaps with molten metal in case of abnormal rise in coolant outlet temperature. Upper and lower plenums of the reactor have graphite blocks with machined channels to provide guided flow to the coolant between the coolant channels and the down corner tubes. The reactor has heat utilization vessels to provide interface to systems for high temperature heat application. A set of heat pipes transfer heat from the upper plenum of the reactor, to these vessels. If the temperature of coolant increases beyond its design value, a set of seven shut-off rods fall by gravity in the central seven coolant channels. Appropriate instrumentation like neutron detectors, fission/ion chambers, various sensors and auxiliary systems such as cover gas system, purification systems, active interventions etc., are part of the design.

This reactor needs a large number of developmental activities in the fields of engineering, materials, fuel, and instrumentation to be carried out. The conceptual design of the reactor has been completed and feasibility of major systems has been established. A feasible configuration of the reactor from reactor

"...In India, we have an atomic energy programme that is based on domestic research and development. Despite technological regimes, we are good in the nuclear power programme in the commercial domain. So we are now going from a phase of consolidation to one of acceleration..."

- Anil Kakodkar, Chairman, AEC

physics, heat removal and reactor control considerations has been worked out. Detailed design, computational modeling and analytical studies of systems are being carried out.

Accelerator Driven Systems

The Accelerator Driven Sub-critical (ADS) reactor system is an emerging technology, which could have a major impact on the nuclear energy scenario. It is a sub-critical reactor device for nuclear power, which would operate with high neutron economy and safety and can be used for power production, fuel breeding, and TRU waste incineration. In an ADS system, an intense proton beam of energy of the order of 1 Gev hits a heavy element target in which neutrons are produced by "spallation" reaction. These are called "external" neutrons in contrast to those resulting from fission in a conventional critical reactor. These external neutrons irradiate a sub-critical core in which they maintain the terminating fission chains. With a continuous feed of proton beam on to the target, steady fission rate in the sub-critical reactor can be maintained. More technical details of ADS system can be found in chapter 6.

Critical Facility at Trombay, for reactor physics experiments