Evolution of PHWR technology

A historical review

Sunil Kumar Sinha

Key to nuclear energy is the fission of Uranium atoms into two smaller atoms by energetic neutrons. These neutrons are themselves generated during the same fission process and therefore help in establishing a chain reaction. Fission of each Uranium atom is also accompanied by release of large amount (200 MeV ~ 3x10-11 Joule) of heat energy which is removed by circulating fluid (typically water)to produce steam to run turbine to generate electricity.

This simple looking process of generating electricity from nuclear fission of Uranium atom is quite complex. The neutrons released during fission have much higher energy (>1 MeV) whereas the efficient fission of uranium atoms happens by neutrons of much lower energy (0.0625eV). So, there is a requirement of reduction of neutron energy by a factor of 107. All the neutrons released during the fission process are not available to cause further fission activities as some are lost by leakage to the surrounding and/or in absorption in non-fission reactions. A careful balance of neutron population is required to be maintained for continuing chain reaction. Further, release of neutron in the fission process happens in a fraction of millisecond which is much smaller than processing and response time of control instruments. Still further, there is a need to replenish fuel on a regular interval which may vary from a day (Heavy water reactor design) to an year and a half (Light water reactor design).

A medium of light nuclei material such as graphite, heavy water, light water, Beryllium is used to slow down the high energy neutrons to a level to cause efficient fission. Going by the functionality of these light nuclei medium, it is named “moderator”. Quantity of fuel, moderator and neutron absorbing materials in a mix are chosen such that steady neutron population is maintained by the neutrons released immediately after fission (prompt) and those released a few seconds after fission (delayed) by the decaying fission products and as such the nuclear chain reaction takes place in a sustained and controlled manner. Control rods made out from neutron absorbing material such as Boron/Cadmium are used to accelerate, slow or shut down the nuclear reaction.

R & D activities carried out in BARC over the last few decades have contributed immensely in the indigenisation and improvisation of PHWR technology. The Indian PHWR design has evolved from initial 220 MWe RAPS with Canadian collaboration to completely indigenous 700 MWe PHWR through a series of improvements over the last five decades. India, today, completely owns this technology as it has mastered all the aspects of this technology – Design, Construction, Commissioning, Operation & Maintenance and Decommissioning

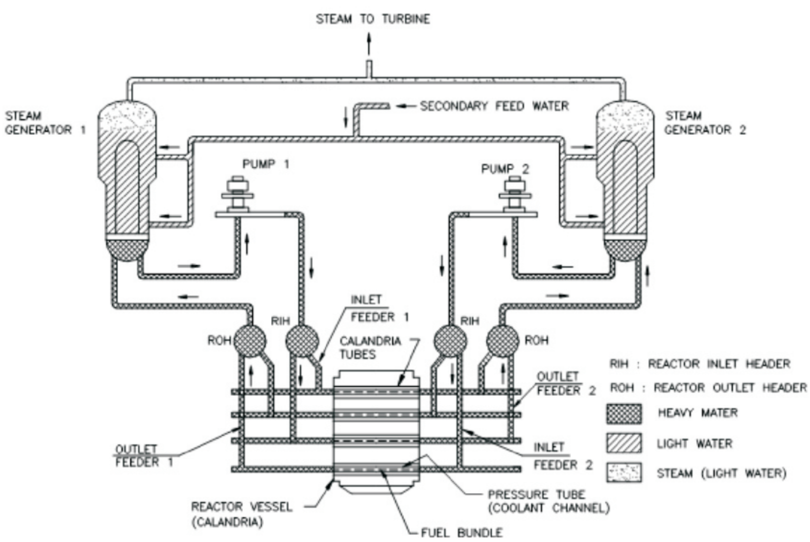
These complexities have been addressed in the design evolved over last 5-6 decades. Majority of reactor technologies developed to harness the nuclear energy to produce electricity are based on Pressurised Water Reactor, Boiling Water Reactor and Pressurised Heavy Water Reactor designs. The naturally available Uranium contains 0.7% of U-235 and rest as U-238. It is U-235 which undergoes fission in a nuclear reactor. The first two reactor designs use Uranium fuel enriched with U-235 (~4%) as fuel whereas the last one uses natural uranium as fuel.

India has limited domestic reserves of natural uranium and abundant reserves of thorium. These aspects have been considered while formulating three stages of Indian Nuclear Programme. The emphasis of the programme is on ensuring energy security by utilization of thorium at a later date.

Amongst the available reactor technologies, India selected Pressurised Heavy Water Reactor (PHWR) technology for Stage-I of its nuclear power programme from considerations of economics, technical viability and near term sustainability. The first PHWR unit (Rajasthan 1) constructed with the help of Atomic Energy of Canada Limited (AECL) at Rajasthan began commercial operation in 1973. When AECL assistance stopped during construction of Rajasthan 2, the Department of Atomic Energy(DAE) India, and eventually the Nuclear Power Corporation of India Ltd (NPCIL), completed the construction. Thence, with indigenous R&D activities spread over last five decades India has not only mastered the technology but also indigenised it with lots of improvisation. Today, India operates 19 PHWR units which is a mix of sixteen units of 220 MWe, two units of 540 MWe, and one unit of 700 MWe. Nearly five PHWR units, each of 700 MWe are in various stages of construction. This article will highlight the development journey of the last five decades in the some key areas of the complex technology.

General Description of PHWR

The PHWR is a heavy water cooled and heavy water moderated natural uranium based fuel reactor. It consists of a horizontal cylindrical vessel called “Calandria” which holds heavy water moderator at nearly ambient pressure and temperature. The calandria is pierced horizontally by large number of pressure tubes (306 in 220 MWe and 392 in 540/700 MWe PHWRs) which house twelve fuel bundles each. High temperature and high pressure heavy water coolant circulates through them to carry away the fission heat generated in the fuel bundles for producing steam to run the turbo-generator for power production. The schematic flow diagram of PHWR is shown in Fig. 1.



1. Flow diagram of PHWR

Indian 540 MWe PHWR design in the units 3&4 at Tarapur Atomic Power Station (TAPS) was an extension of the standardized design of 220 MWe PHWR. After successful operation of 540 MWe PHWR, India leapfrogged in the nuclear power with indigenous design, construction and commissioning of 700 MWe PHWR.

Evolution of PHWR designs in India

The Indian PHWR design has evolved through a series of improvements over the last five decades. Such improvements have been driven by, among others, evolution in technology as result of persistent in-house R&D activities, feedback from operating experience in India and abroad, including lessons learnt from incidents and their precursors, evolving regulatory requirements and cost considerations. The first two PHWRs units at Rajasthan Atomic Power Station were of 220 MWe Canadian Douglas Point reactor design. Work on these was taken up with Canadian co-operation. For the unit-1, most of the equipments were imported from Canada, while for the second unit a good amount of indigenisation was achieved. At the next station, Madras Atomic Power Station (MAPS), a number of changes in design were adopted mainly due to site conditions.

The imported technology content in these and subsequent plants was progressively reduced to 10–15%. Design of Narora Atomic Power Station (NAPS) units 1&2 – the third power station in the country had factored India's operating experience with PHWRs, including aspects such as ease of maintenance, in-service inspection requirements, improved constructability, increased availability and standardization. New design concepts and improvements in the design incorporated in NAPS units improved the safety of reactor, paved the way towards the standardisation of the design of 220 MWe capacity and served as stepping stones for the design of the larger version (540 MWe) PHWR.

Introduction of two independent fast acting reactor shutdown systems, a high pressure Emergency Core Cooling System (ECCS), and a double containment with suppression pool were the remarkable shift in the safety approach from the units at Rajasthan and Madras. Subsequent to NAPS, Kakrapar Atomic Power Station(KAPS) Unit 1&2, Kaiga Generating Station (KGS) Unit 1&2 and RAPS Units 3&4 saw further improvements leading to a standardized design and layout for 220 MWe PHWRs. Indian 540 MWe PHWR design in the units 3&4 at Tarapur Atomic Power Station (TAPS) was just an extension of the standardized design of 220 MWe PHWR. After successful operation of 540 MWe PHWR, India leapfrogged in the nuclear power with indigenous design, construction and commissioning of 700 MWe PHWR.

Indigenisation of the complex technology

Post AECL withdrawal of support during construction of RAPS unit-2, India launched extensive R&D activities in several key areas to indigenise the PHWR technology. These efforts bore fruits in terms of development of expertise in reactor physics design, reactor thermal hydraulics, component design and fabrication, special material developments, reactor controls & instrumentation, fuel fabrication, fuel handling and failure assessment and repair technology, and radiation monitoring.

PHWR being a Natural Uranium fuelled reactor, demand for fuel is continuous. Fresh fuel is loaded and used fuel is unloaded while reactor is in power. This technology is called on-power refuelling. It introduces complexity in fuel management. In-house reactor physics simulation to work-out scheme for refuelling with the central objectives of fuel management and maintaining steady neutron flux has been developed.

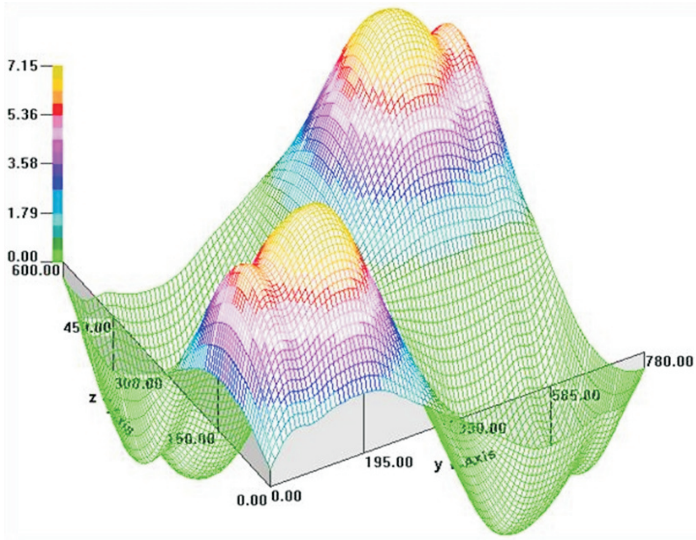
Reactor physics analysis capabilities for design and safety studies for Pressurised Heavy water Reactors (PHWRs) have leapfrogged over the past years. The major workhorse for these simulations is the design codes developed by BARC over the years. Development of new nuclear data (nuclear properties as a function of energy), methods for treating the heterogeneities and experimental validation have played key role in strengthening the design codes and taking informed decisions in respect of design improvements. Use of thorium for flux flattening in some of the PHWRs was one such mature decision which helped in development of several key technologies related to fuel fabrication, thorium in-reactor performance, irradiated fuel handling, generation of irradiation data in post-irradiation examination, technology for separation of U-233.

Flux profile of a transient analysed using In-house developed advanced computational tool simulating entire core in 3D space-time approach is shown in the Fig. 2.

As indicated above, the PHWR design requires online refuelling on daily basis. This is accomplished by opening the high pressure boundary using a specialised remotely operable machine called fuelling machine. This machine opens the high pressure and high temperature boundary of the channel, collects the burnt fuel, loads the new fuel and seals the pressure boundary. The design, development, qualification and testing of this specialised machine has been carried out in BARC. A dedicated high temperature engineering test loop was set-up for testing of this fuelling machine in simulated in-reactor conditions. Testing of fuelling machine is shown in Fig. 3.

Reactor thermal hydraulics deals with the aspects of core heat removal normal operating condition, shutdown condition and accident condition. It is one of the key areas of reactor design which requires meticulous attention. BARC has developed immense expertise in this field. Several experimental loops have been set-up over the years to study and validate the novel ideas of core thermal hydraulics.

Facility for Integral System Behavior Experiment (FISBE) is one such experimental loop in BARC which simulates full height of reactor primary heat transport system. This experimental loop has been used to investigate several concepts of core heat removal during incidence of leakage of coolant. Recently the loop has upgraded by installing a system for passively removing core heat during loss of all kinds of power. Such a system is installed in 700 MWe PHWR to assure cooling of reactor core in the event of loss of all kinds of power. The efficacy of such a system has been established experimentally in FISBE. In addition, the pressure drop experiments in the full scale fuel bundles of 220 MWe, 540 MWe and 700 MWe PHWRs were conducted in BARC providing key input for the thermal hydraulic design and safety analysis of PHWRs.



2. Higher harmonics of neutron flux in a typical PHWR core using space-time kinetics code

There are several instances during the journey of evolution of PHWR design where BARC contributions have been immensely impactful. In a first of a kind experiment, capability of thermo-siphoning to remove the heat from a reactor under shutdown condition when there is no electrical power of any kind is available, was demonstrated by BARC experts in Narora Unit. This capability proved bliss when fire broke out in turbine building of Narora unit. When Madras units suffered from moderator inlet manifold failure, BARC carried out detailed studies on rehabilitation schemes of both the units. Based on studies these units were first partially rehabilitated by changing the moderator inlet and outlet path (a first of kind improvisation in PHWR design technology) and were later completely rehabilitated by installing sparger tubes for moderator inlet.

Safety of PHWR design in the event of the most severe accident was further established by demonstrating experimentally that the vessel boundary will remain intact even if reactor core loses its coolant completely and heat removal is jeopardised.

Structural materials used for reactor core components have to perform in quite harsh environment characterised by high temperature, high pressure and high neutron radiation.

Choice of materials becomes limited due to demanding harsh environment. This choice is further restricted to Zirconium alloys for PHWRs where neutron economy is a controlling parameter. Because of strategic application of such materials, technology for

3. Qualification testing of fuelling machine

The PHWR design requires online refuelling on daily basis. For this, a specialised remotely operable fuelling machine (FM) had been designed and developed in BARC to cater to the needs of entire fleet of nuclear power plants of NPCIL

development has to be initiated indigenously. BARC has undertaken R&D activities and played key role in development of zirconium alloys specific to PHWR structural components and fuel cladding, development of manufacturing technology of components, mechanical and metallurgical characterisation of the materials at different stages of manufacturing of components, and development of quality assurance programme and related technology for assuring in-reactor performance for the anticipated design life. These technologies were transferred to Nuclear Fuel Complex, Hyderabad for mass production of components.

Instrumentation and Control (I&C) systems play important role in protection, control, supervision and monitoring of a Nuclear Power Plant (NPP). They together with plant operating personnel, forms the 'central nervous system' of an NPP. Historically, right from the days of India's first nuclear reactor APSARA, most of the I&C activities in Indian nuclear program have been supported by research and development undertaken at BARC in the areas of electronics, instrumentation, communication, computing and information security.

I&C system in a NPP, through its elements (e.g., equipment, sensors, transmitters, actuators, etc.), senses basic physical parameters, monitors performance, processes information, and makes automatic adjustments to plant operations as necessary. It also responds with appropriate action to process failures and off-normal events, thus ensuring plant and personnel safety. Since I&C systems need to meet not only the functional, performance and interface requirements but also the enhanced reliability, safety and security, a lot of importance is given to activities involving the design, review, testing, operation, maintenance and qualification of these systems.

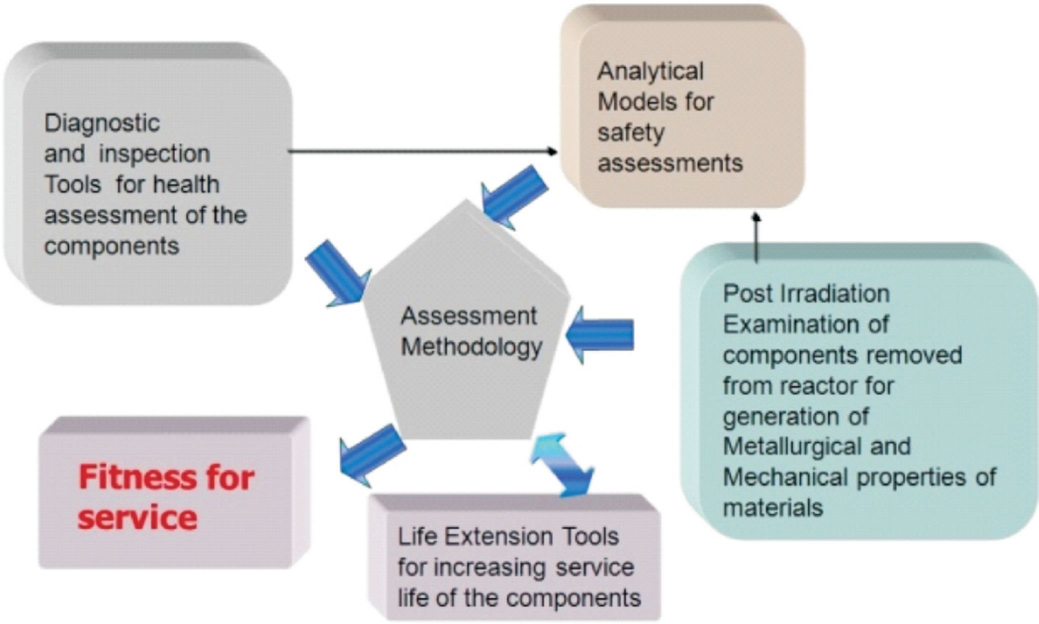
The I&C systems and equipment are classified depending on their relationship to plant safety such as safety critical, safety related etc. This categorization allows the systematic application of appropriate design and engineering techniques and, just as importantly, establishes its qualification roadmap.

Traditionally, hardwired analog systems were used in all safety significant applications. Nevertheless, the practice of design and implementation of computer-based systems (CBS) for carrying out functions important to safety has evolved and matured over past several years. Computer based system for monitoring purpose was introduced for the first time in the first indigenously built MAPS units where channel outlet temperatures were monitored and displayed in user friendly manner to the operator using computer based channel monitoring system. This led to reduction of large number of panels of analog electronics. CBS based Reactor Regulating System (RRS) for automatic control of reactor power, Disturbance Recording System (DRS) for recording an event and Event Sequence Recorder for recording sequence of events were developed and deployed in different NPPs came thereafter. With the confidence gained in CBS, the first of its kind safety system – Programmable Digital Comparator System (PDCS) – based on three channel architecture was designed, qualified and deployed in Kakrapar NPP.

All these developments led to simplification in design of control room layout and provided comfort to the control room operator in better visualization, monitoring and controlling the reactor operation.

Migration of analog based system to computer based system has passed through several stages of evolutions over the years. Independent Validation and Verification (IV&V), redundancy and channel independence, philosophy of single failure criteria etc. have been introduced to increase defense in depth following Indian and International safety guides. With the advent of computers with enhanced computing powers, CBSs were further upgraded with fault tolerant architecture based on Dual Processor Hot Standby configuration for RRS and process control system in KGS 1&2 and RAPS 3&4. These systems were designed to work in a networked configuration for communication of process and system information in real time. Further, implementation of Supervisory Control and Data Acquisition (ESCADA) System - for electrical systems introduced in RAPS-3&4 made visualization and monitoring of parameters much easier.

R&D activities concurrent with the evolution of technology in the field of electronics, instrumentation, communication, computation, networking and information security pursued in BARC played a vital role in indigenization PHWR technology which started from RAPS unit 2. India's 540 MWe PHWR which is more than double the capacity of previous 220 MWe PHWR is the true



4. Strategy for Fitness for service

display of capability of BARC R&D strength in designing and implementation of several field tested first of a kind systems. Flux mapping system, RRS following a distributed architecture, algorithms for control of spatial power distribution, and Liquid Zone Control System (LZCS) with fourteen zonal control compartments for power control are some of the examples. Smooth implementation of these new systems was possible only through rigorous testing and optimization in a full scale test setup of LZCS and a simulator, developed and installed in BARC.

Coolant channel assembly is one of the most important components of PHWR. They are equivalent to arteries in human body. Integrity of these arteries is vital for safe operation of reactor during its design life. Operating environment is quite harsh for the component. They affect the functionality of the component by degrading its geometry and the material of construction. Limited choice of material and performance feedback concurrent with operation requires continuous evolution in design as well as chemistry of alloy to maximise the anticipated design life.

The coolant channel assembly requires an evolutionary ageing management programme supported by extensive R&D. In the Indian PHWR context, the programme started way back in 1989 when there was failure of moderator Inlet manifolds (attachment inside the calandria vessel for allowing moderator to enter into the vessel) in MAPS. Failed pieces of manifold impacted calandria tubes near the bottom rows and damaged two calandria tubes. BARC played crucial role in retrieval of failed manifold pieces, removal of pressure tubes and calandria tubes at the failed calandria tube locations and rehabilitation of the two units by designing, installing and commissioning sparger tubes for moderator inlet. This was BARC first encounter with ageing management and life extension. Subsequently, there is no stopping. BARC did pioneering contributions in the life extensions of coolant channels of newly commissioned reactors at Narora and Kakrapar and also when these reactors became old.

Further, BARC holds credit for the development of several systems for inspection, diagnostic and life extension, analytical tools for safety assessment, examination of irradiated components removed from reactors for material surveillance and development of criteria for fitness for service of these components (Fig. 4).

Research and Development activities carried out in BARC over the last few decades have contributed immensely in the indigenisation and improvisation of PHWR technology. India, today, completely owns this technology as it has all mastered the aspects of this technology – Design, Construction, Commissioning, Operation & Maintenance and Decommissioning.

Dr. (Smt.) Umasankari K, Shri T. Srinivasan, Dr. Raghvendra Tewari and Shri U.W. Vaidya of BARC contributed to this article.