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On Nuclear Reactors

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Editorial
Nuclear Energy option & Climate Change

Exclusive interview
Dr. Anil Kakodkar
Former Chairman AEC

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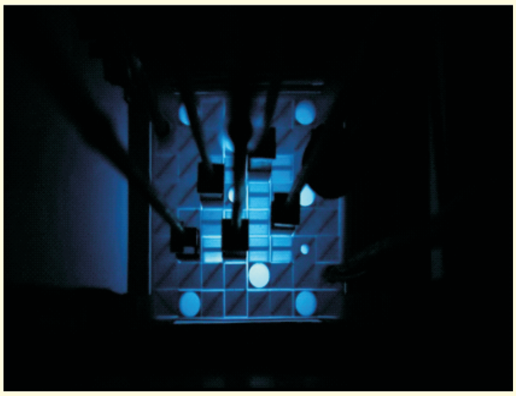
On Nuclear Reactors

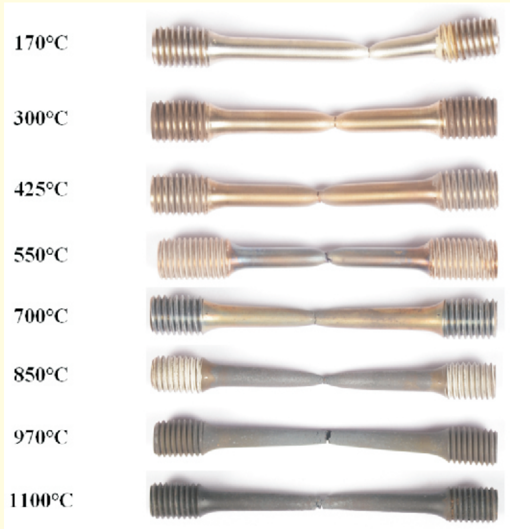
Bird's eye view of
Bhabha Atomic Research Centre
Mumbai



Cherenkov Radiation during
Power Operation of
Research Reactor Apsara-U

Artist's impression of
Advanced Heavy Water Reactor
Facility

On the Cover
Research Reactor Dhruva
@ BARC Photo



High temperature material data of Calandria material ~1100°C obtained from systematic tensile and creep tests programme

associate editors'
MESSAGE

Change is an inherent strength of any publication, and much has changed for the BARC Newsletter since the launch of its first issue as a four page booklet in August 1983. In its perpetual struggle for amelioration, from this issue onwards the BARC Newsletter reaches you in a new visual design and we are proud to instigate this transition. We do hope that this change is appealing to you.

This special issue includes several seminal articles which exemplify the contributions of BARC in design, development and operation of nuclear reactors. It includes four finest feature articles, respectively on technology evolution for Pressurized Heavy Water Reactors, development of Advanced Heavy Water Reactor, reactors for third stage of Indian nuclear programme, and history of research reactors in BARC, apart from the editorial note on nuclear power option for a low carbon energy hungry nation. In addition, it presents two full length research articles on radical innovations by research scholars of HBNI in the field of nuclear reactors, and several news articles featuring the latest developments in the subject matter. Above all, this issue features an in-depth interview with none other than Padma Vibhushan Dr. Anil Kakodkar.

Over the past three decades of BARC Newsletter, what has evolved is the continuous improvements on the standard and quality of this publication and its sincere efforts to reflect and mirror the achievements of BARC. We trust that this issue has also done justice to this legacy. We use this opportunity to express our sincere gratitude to all the scientists and engineers of BARC and HBNI who have contributed articles and to the reviewers who have spent their quality time to make the Newsletter look like what it is today.

(S.R. Shimjith) **(V.H. Patankar)** **(A.K. Nayak)**

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Editorial

By Dr. A.K. Nayak

Nuclear power option for a low-carbon climate resilient future India

Currently, India is one of the fastest growing economies of the world and the country is also one of the top energy consumers as well as producers in the world. India's total primary energy supply (TPES) is around 900 million tons of oil equivalent (Mtoe). Coal is the largest primary energy source in present scenario. Of the total 900 Mtoe TPES, 600 Mtoe (nearly two-thirds) is being provided by domestic production and remaining is imported[1].

Amongst the different fuels, the share of electricity consumption in India is only 17 % of the total final consumption, which is around 370 GW installed capacity. Even though, the country is the 3rd largest electricity producer in the world, however, our per capita electricity consumption is much below than the world average (India - 1182 kWh; Singapore - 9220 kWh; Malaysia - 4810 kWh; Thailand - 2870 kWh, China – 4600kWh,world average - 3150 kWh, OECD countries - 7990 kWh)2. If we have to provide a comfortable living standard to our growing population, we have to increase significantly the share of electricity in the total energy consumption.

India's population is continuously growing and it is expected to touch 1.7 billion by 2047[3]. The Population Reference Bureau (PRB)4 has also estimated Indian population to reach 1.67 billion by 2050 surpassing that of China. Similar estimates have been made by UN5 for India to have around 1.64 billion population by 2050. In future, India will see substantial increase in power demand on account of population rise, increased urbanization, GDP growth, increase in manufacturing, etc.

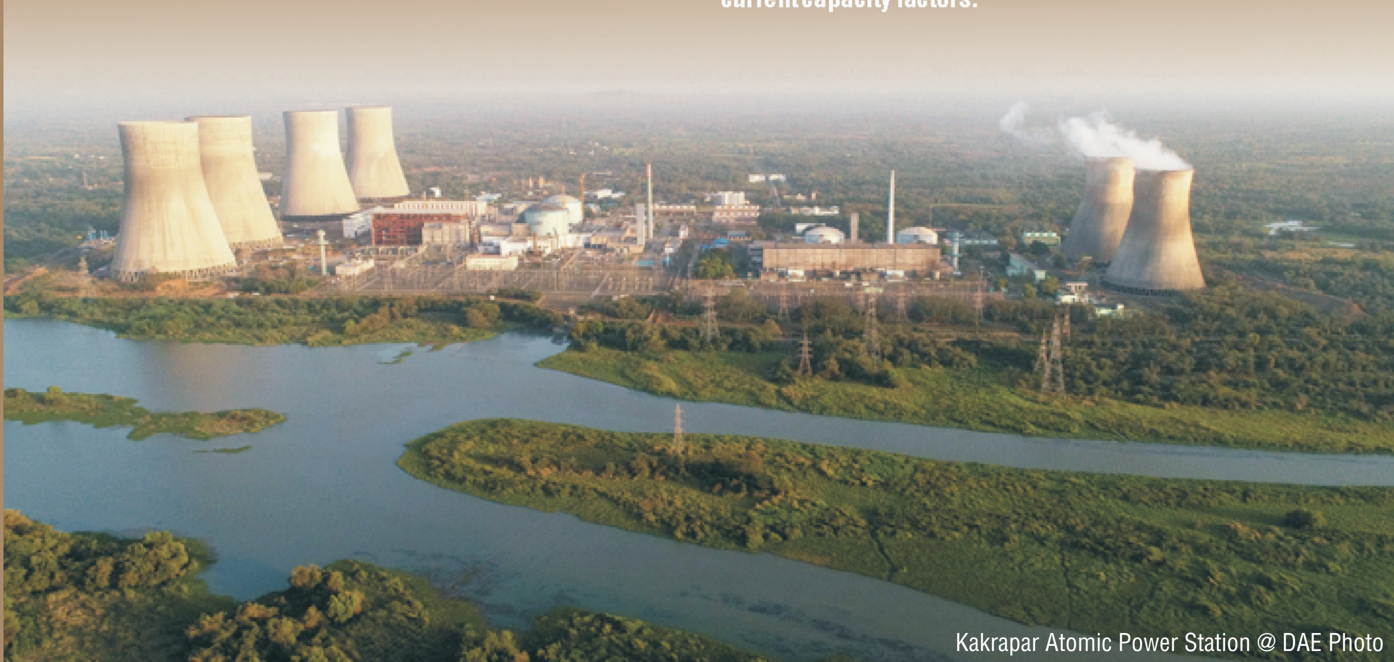
So how much electricity we require by 2050?

Based on a conservative estimate of 1.7 billion population by 2050, the electricity demand can be postulated by the following scenarios:

1) World Average Level (WAL) - meeting world average of ~ 3000 kWh/annum per capita consumption

2) Ambitious -achieving HDI close to 0.9 in developed countries where saturation effect is seen at around per capita consumption of 5000 kWh/annum[6]

These translate to raise the current capacity to 5100 TWh/annum (excluding transmission losses) for WAL target, and if we go for ambitious target, we need to grow to around 8500 TWh/annum (excluding transmission losses) in less than 3 decades. Or, in other words, the country needs to have at least 1200 GW (WAL target) or 2000 GW for an ambitious target by 2050 from the current installed capacity of 370 GW assuming current capacity factors.



Kakrapar Atomic Power Station @ DAE Photo

So what are our options? Electricity Option from Coal

Currently, coal is main driver of electricity production in India, and if it continues to be the main stay, there are other adverse effects in terms of Green House Gases emission (GHG). The potential of CO_2 emission from coal-based plants is ~500 folds more than nuclear, hydro and wind power generation[7]. A few decades ago, the energy policies were more focused for alternate energy sources because of concerns of fossil based fuels getting exhausted by the end of this century. Now the situation demands early induction of clean and green sources of energy much before fossil based resources are exhausted. To reduce the GHG emissions, Clean Coal Technologies are also being developed. The clean coal technology is primarily based on carbon capture and storage (CCS) with additional energy penalty. Moreover, if CCS is used, the costs for electricity production are likely to increase by 45-50%, due to an increase in fuel costs and capital expenditures[8].

To save the environment from the GHG, deep de-carbonization (reduction by one order of magnitude or more in GHG emission) of the electrical energy sector has been felt globally. Towards this, United Nations Framework Convention on Climate Change (UNFCC) was adopted in March 1993 in Rio Convention. Thereafter, Kyoto protocol (2005) and Copenhagen Accord (2009) targeted to limit the future increase in the global mean temperature to below 2°C[9-14]. The next major international commitment took place in 2015 in Paris, which replaced the Kyoto protocol. According to the Paris agreement with 197 countries' participation, the developing countries must be financially supported by the developed nations to meet emission goals[15,17]. In 2019, the British parliament became the first nation to officially declare climate emergency[18]. As of 5th Jan 2020, 25 countries including the UK, France and Argentina have made national declarations of climate emergencies. To achieve the 2050 target for climate stabilization set by the Paris Agreement, the United States would need to reduce CO_2 emissions by more than 97%; in other words, reduce the carbon intensity of its electricity mix from 500g CO_2/kWh to less than 15g CO_2/kWh. In a separate study, researchers at MIT estimated that emissions need to be further reduced to levels approaching 1g CO_2/kWh[19].

In the Indian front, the country submitted its Nationally Determined Contributions (NDCs) to reduce the emissions intensity of Gross Domestic Product (GDP) by 33–35% of 2005 levels by 2030. In addition, the NDCs also targeted 40% fossil-free power generation capacity by 2030 and the creation of an additional carbon sink of 2.5 to 3 billion tonnes of CO_2 equivalent (CO_2eq)[20]. However, in 2016, India's CO_2 emissions increased by about 4.7% to 2.5 Gt CO_2. To achieve the Paris agreement target, India needs a substantial reduction in carbon emissions. *It is now predicted that if India abandons further plans to build new coal-fired power plants, it could become a global climate leader and Climate Action Tracker (CAT) would rate it "1.5°C-compatible"*[21]. To achieve the target, IINDC has recommended utilizing the potential for all green energy sources to achieve the above targets, which include renewable energy sources like solar, wind, hydro power, and nuclear.

Energy Option from Renewable sources

Different agencies have assessed maximum potential of solar energy in India. MNRE[22] has estimated that solar potential of the country is around ~ 1640 TWh/annum assuming 3% of the waste land (~ 3% of 500000 Sq km = 15000 Sq km) area to be covered by Solar PV modules. NITI Aayog has estimated the potential to be around 2040 TWh/annum with storage technology available and smart grids in place[23]. Sukhatme[24] has made different postulates for utilization of 5%, 10% or 15 % of barren and uncultivated land in India (200000 Sq km), which can give 1095 TWh/annum - for 5% of land (10000 Sq km); 2190 TWh/ annum -- for 10% of land (20000 Sq km) and 3285 TWh/annum - for 15% of land (30000 Sq km) with 25 % annual plant load factor considering availability of solar energy for average 8 hrs per day and 9 months per year in Indian context. Assuming that, by 2050, storage technology is fully deployed, and there are no plant outages and other losses and around 10 % barren land has been utilized; the potential of solar energy may optimistically be considered to be around 2000 TWh/annum.

Potential for Wind energy

MNRE[22] has predicted maximum wind energy potential for the country as 978 TWh/annum with 100 meter MAST height above ground level.

Hydro power

As per assessment made by CEA[25], India is endowed with economically exploitable hydropower potential of the order of 150 GWe. However, at 60% load factor, it can contribute 788 TWh/annum.

Bio Energy

The bio energy potential has been estimated by MNRE to be 60 TWh/annum.

Total Energy potential by renewable alone is given in following table

Energy	Potential TWh/annum
Solar	2000
Wind	978
Hydro	788
Bio	60
Total	**3826**

It can be seen that, with only renewables alone with the assumptions that 10% barren land area is available for solar (as per Sukhatme[24]) with full deployment of storage technology and smart grids, and with 100 m Mast height for wind, there will be a deficit of 1274 TWh/annum to meet the WAL target of world average of 3000 kWh/annum. However, if we want to achieve HDI of 0.9, there is very large deficit of 4674 TWh/annum to meet the target of 8500 TWh/annum. This roughly translates into deficit of minimum 180 GWe for WAL target and more than 660 GWe for ambitious target assuming 0.8 plant load factor.

So the biggest question is what can meet this large energy deficit of India by 2050?

Obviously, this cannot be met without large scale deployment of nuclear energy.

Nuclear energy is well established as a compact green energy source recognized by IPCC (energy which does not contribute to direct GHG emissions)[26]. One kilogram uranium-235 contains two to three million times the energy equivalent of oil or coal. This implies that for high grade (G4 grade) coal with 6100 kCal/kg calorific value, around 2900 tons of coal per annum is required per megawatt of electricity produced for a plant load factor of 85 %. On the other hand, only 0.2 tons per MW per year natural uranium is required for the same energy production. If we use enriched uranium, the requirement comes down to mere 0.025 Tons per MW per year.

Currently, India operates with 19 PHWRs with an installed capacity of 5160 MW. In addition, new plants under construction are a 700 MW PHWR at Kakrapar, Gujarat and 2x700 MW PHWRs at Rawatbhata, Rajasthan. Financial sanction has been obtained for 2x700 MW PHWRs at Gorakhpur, Haryana. Administrative approval and financial sanction have been obtained for ten 700 MW PHWRs in fleet mode. To increase the capacity building, NPPs with LWR technologies are being built with foreign co-operation. These include 2x1000 MW PWRs at Kundankulam, Tamil Nadu; another 2x1000 MW PWRs are under construction at the same site and financial approval has been obtained for another 2x1000 MW PWR plants. This process has facilitated the import of fuel and fuel security for Indian NPPs from foreign sources through the international civil nuclear cooperation agreement. In addition, negotiations are being held with other vendors for import of light water reactors to increase the capacity building. Apart from this, the country is in the advanced stage of development of a mid-sized Indian PWR which can further enhance the capacity building. For second stage of the three stage nuclear power programme, the Prototype Fast Breeder Reactor is in advanced stage of commissioning. Once this technology matures, it is expected to provide 42000 GWe-year electricity in future. Technology development of molten salt reactors for third stage is also ongoing.

Since nuclear power growth using technologies from 2nd stage and 3rd stage will take some time, we need to explore other technologies including Modular Factory Assembled Nuclear Reactors (MFARs) which can help significant nuclear power growth in next 2 to 3 decades. MFARs are targeted to be built in factories and shipped to sites for installation. They are small, integral type reactors housing all the major components like SG, Pressurizer, pumps inside RPV. The RPV is placed inside a steel containment which is submerged under a water pool.

These reactors have advantage of small core size which requires less amount of decay heat to be removed as compared to large reactors. This enables use of several passive safety systems in its design to remove the small decay heat during any transients or accidents without leading to core melt during design extension conditions thus qualifying for "no impact in public domain".

Being small size, modularity in design and manufacturing can be established. Modularity will help in fabrication of the major components including the reactor pressure vessel, steam supply systems, and cooling systems in centralized manufacturing facilities and integrating them as a single unit; that means the nuclear steam supply system involving the reactor core, steam generator, pumps and pressurizer, are housed inside the pressure vessel making it a single product. In addition, modularized construction of steel containment saves a lot of civil construction at site. Reduction in civil construction and factory fabrication at site makes possible to connect the MFAR to grid in a small time period. The integrated reactor can be shipped directly to the site for installation and connection to the grid. If several such reactors are to be built, the economy of numbers will compete with the economy of scale. These technologies can be implemented for liquid metal cooled reactors in second stage and molten salt reactors for third stage.

Closure

Nuclear energy has to play significant role in the energy mix to meet the large energy deficit by 2050. Apart from current nuclear reactors, new technologies should be explored which have the ability for rapid multiplication and are extremely safe. Modular Factory Assembled Reactors (MFAR) is a promising option to meet the growing electricity demand of India.

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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

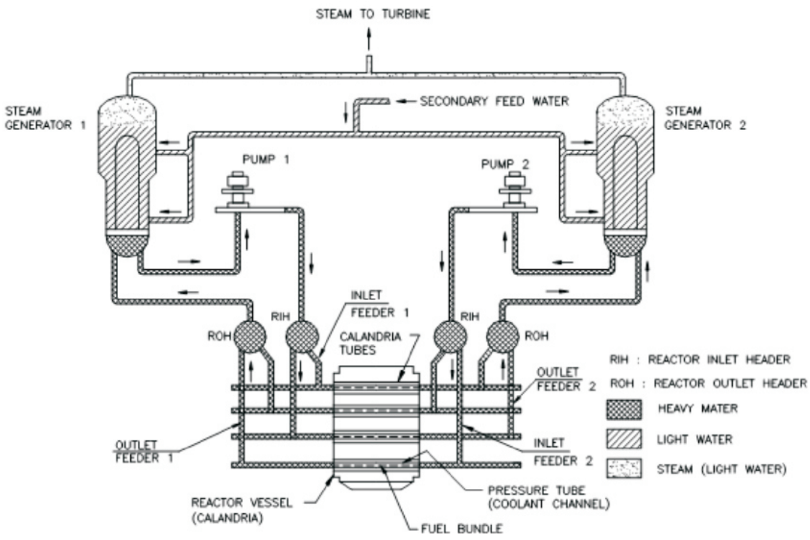
These complexities have been addressed in the design evolved over last 5-6 decades. Majority of reactor technologies developed to harness the nuclear energy to produce electricity are based on Pressurised Water Reactor, Boiling Water Reactor and Pressurised Heavy Water Reactor designs. The naturally 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.

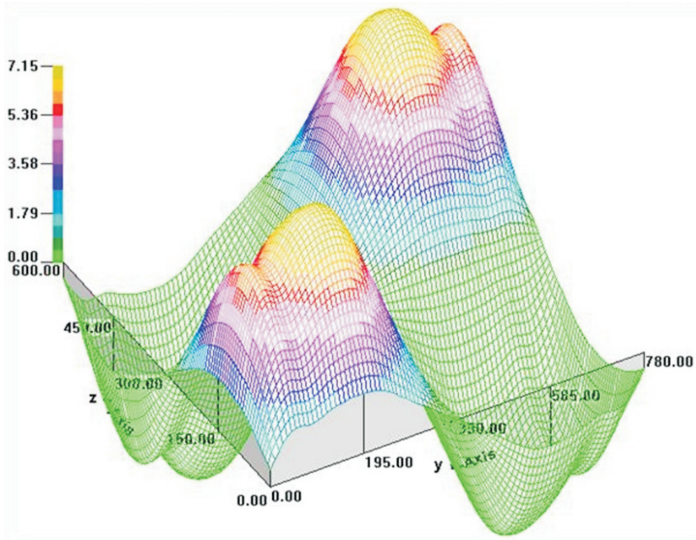
Reactor physics analysis capabilities for design and safety studies for Pressurised Heavy water Reactors (PHWRs) have leapfrogged over the past years. The major workhorse for these simulations is the design codes developed by BARC over the years. Development of new nuclear data (nuclear properties as 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.

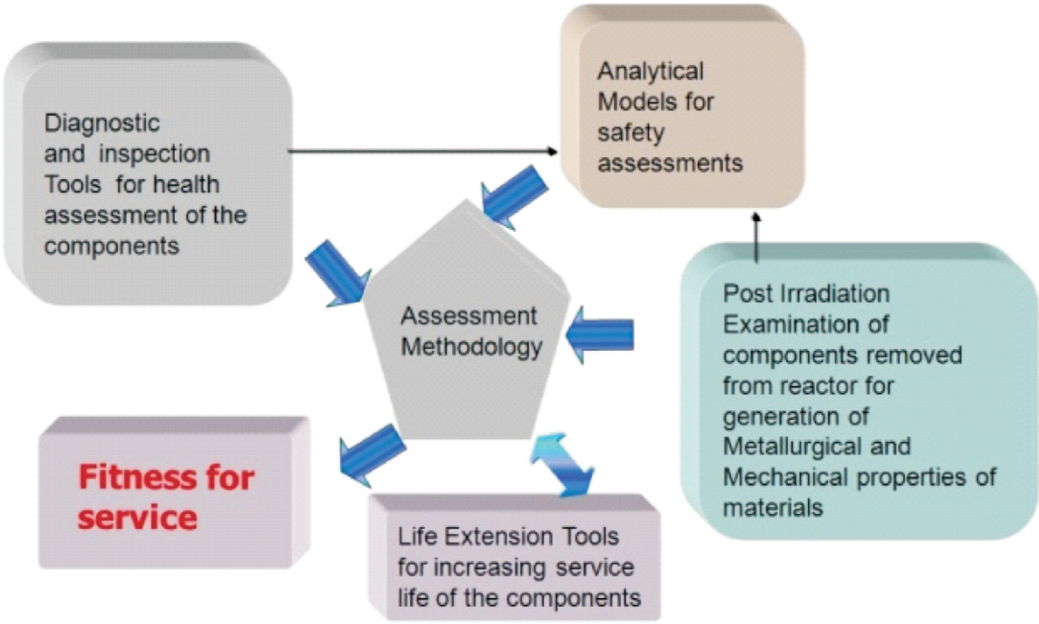
The I&C systems and equipment are classified depending on their relationship to plant safety such as safety critical, safety related etc. This categorization allows the systematic application of appropriate design and engineering techniques and, just as importantly, establishes its qualification roadmap.

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.

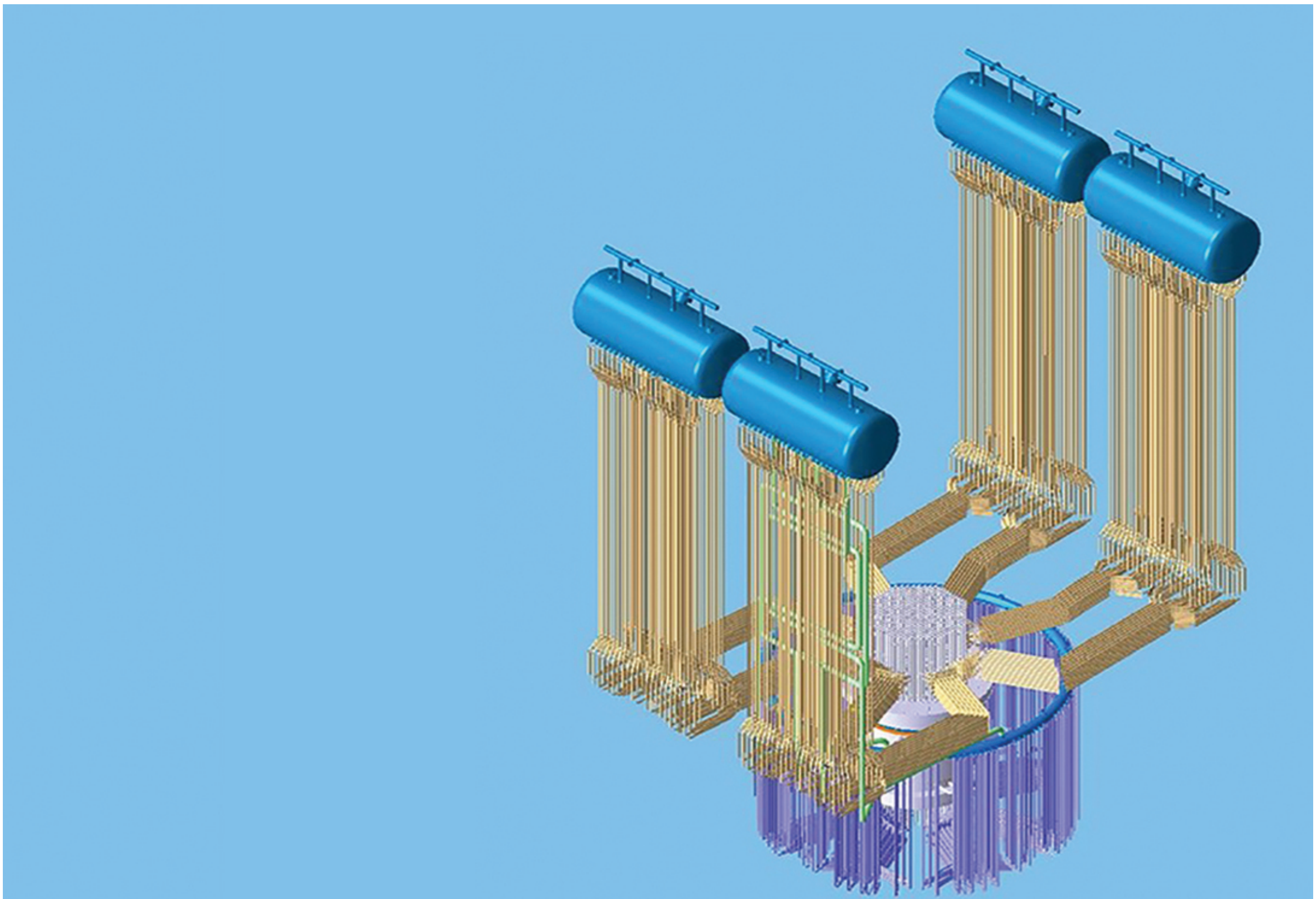
Coolant channel assembly is one of the most important components of PHWR. They are equivalent to arteries in human body. Integrity of these arteries is vital for safe operation of reactor during its design life. Operating environment is quite harsh for the component. They affect the functionality of 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.S. Srinivasan, Dr. Raghvendra Tewari and Shri U.W. Vaidya of BARC contributed to this article.



3D View of Main Heat Transport System

Advanced Heavy Water Reactor for Thorium Utilisation and Enhanced Safety

N. K. Maheshwari, M. T. Kamble, V. Shivakumar, Umasankari Kannan, A. K. Nayak and Avaneesh Sharma

The Advanced Heavy Water Reactor (AHWR) is designed and developed to demonstrate large-scale use of thorium for the generation of commercial nuclear power. AHWR is a 300 MWe, vertical pressure tube type, boiling light water cooled and heavy water moderated reactor. It incorporates several passive safety features and inherent safety characteristics including many First Of A Kind (FOAK) systems. Additionally, AHWR will produce desalinated water utilising process steam and waste heat. This article provides an overview of the AHWR design, fuel design, fuel cycle proposed, safety philosophy and experimental design validation undertaken.

Introduction

Long term sustainability of energy resources calls for the use of thorium as it is three to four times more abundant than uranium globally. In the context of Indian nuclear programme, thorium has a prominent place due to the unique resource position of having large thorium deposits, but limited uranium reserves.

To optimally utilise modest uranium but large thorium reserves in the country, the Department of Atomic Energy has adopted a three stage nuclear power programme. This is aimed towards achieving long term energy security and is based on a closed nuclear fuel cycle. The three stage nuclear power programme, which is being implemented sequentially, aims to multiply the domestically available fissile resources through the use of natural Uranium in Pressurised Heavy Water Reactors (PHWRs) in first stage, followed by use of Plutonium obtained from the spent fuel of PHWRs in Fast Breeder Reactors (FBRs), in the second stage. Large scale use of thorium is contemplated in the third stage, making use of fissile Uranium-233 (^{233}U) based breeder reactors, when adequate nuclear installed capacity in the country has been achieved. The third stage reactors are aimed to be self-sustaining reactors requiring only thorium as feed. Work has therefore been carried out on all aspects of thorium fuel cycle including mining, ore conversion, fuel fabrication, irradiation in reactors and fuel reprocessing [1].

After demonstrating the technologies at laboratory-scale and pilot-plant scale, currently, the focus is on development of all technologies related to development of thorium based reactor systems, so that, a mature technology is in place, well before beginning of large scale deployment of the thorium based reactors.

Advanced Heavy Water Reactor (AHWR) has been conceptualized as one of the options to provide impetus to the different thorium fuel cycle development programmes. It will demonstrate large scale use of thorium and associated technologies for generation of commercial nuclear power, at the same time, demonstrating advanced safety systems envisaged for next generation of nuclear reactors. Additionally, AHWR will produce desalinated water utilising process steam and waste heat. AHWR being a pressure tube type, heavy water moderated reactor, builds on use of the proven technologies developed in PHWR especially pertaining to pressure tube and low pressure moderator based design. At the same time, AHWR will demonstrate extensive use of passive systems and inherent safety characteristics for reactor operation and reactor safety under all operating conditions [2]. The implicit safety objective of AHWR is to limit impact of design basis as well as beyond design basis accidents to plant site, so as to have negligible impact in public domain. Its basic design and experimental development in areas required to establish feasibility of the basic design have been completed at BARC. Several major experimental facilities have been set-up and some others are under development to produce additional data.

Apart from above objectives, AHWR also addresses the concerns regarding long term sustainability and proliferation resistance. It will also have lower environmental impact due to lower long term radioactive waste generation including minor actinides.

Brief Description of Reactor

AHWR is a 300 MW_e, vertical pressure tube type, boiling light water cooled and heavy water moderated nuclear reactor [2]. AHWR uses thorium based fuel with slightly negative void coefficient of reactivity and boiling light water in natural circulation mode as coolant.

Simplified schematic of AHWR is shown in Figure 1. The reactor core is housed in calandria, a cylindrical stainless steel vessel containing heavy water, which acts as moderator and reflector. The calandria, located below ground level, contains 452 number of vertical coolant channels in which the boiling light water coolant picks up heat from fuel assemblies suspended inside the pressure tubes. The coolant circulation is driven by natural convection through individual tail pipes to steam drums, where steam is separated for running the turbine cycle. The four steam drums (only one shown for clarity), receive feed water at stipulated temperature to provide optimum sub-cooling at reactor inlet. Down-comers, four from each steam drum, bring the flow to a circular inlet header, which distributes the flow to each of the 452 coolant channels through individual feeders. The coolant channel consists of a pressure tube and end fittings at top and bottom ends.

In AHWR, emergency core cooling system (ECCS) is made highly efficient by employment of direct injection of ECCS water into the coolant channels and thereby on the fuel pins. In case of loss of coolant accident (LOCA), the emergency core cooling system (ECCS) with four independent circuits (only one is shown for clarity) is actuated in passive mode. ECCS operation consists of passive high pressure injection system using accumulators and passive low pressure injection system using Gravity Driven Water Pool (GDWP) as source of water, in a sequential manner. This can provide core cooling for at least 7 days by ECCS water flooding feeder and tail pipe vaults in V1 volume.

The Reactor Protection System comprises two independent fast acting shutdown systems. Shutdown System–1 (SDS–1) is based on mechanical shut-off rods with boron carbide based absorbers in 37 lattice positions, designed to provide sufficient negative reactivity worth even in case when two maximum worth rods not available. Shutdown System–2 (SDS–2) is based on a liquid poison injection into the moderator. In addition, a pressurized addition of poison, passively driven by steam pressure, takes place in the event of over pressure in the Main Heat Transport (MHT) system. In addition, for long-term sub-criticality control, there is a provision to add boron to the moderator.

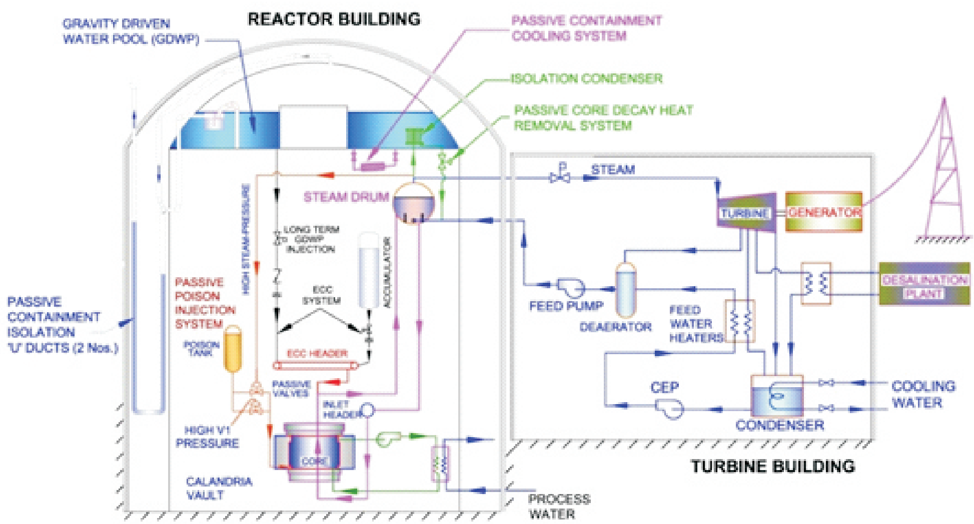


Figure 1: Schematic of AHWR

Main design features of AHWR	
Reactor thermal output	920 MW_{th}
Power plant output, gross	304 MW_e
Primary Coolant	Boiling light water
Moderator	Heavy water
Fuel material	(Th, ^{233}U)MOX and (Th, Pu)MOX
Number of fuel assemblies	452 Nos at pitch of 225 mm
Coolant Channel (Vertical pressure tube design)	PT: Zr-2.5%Nb-20% cold worked CT: Zr-4
Average discharge burn-up of fuel	38000 MWd/T
Active core height	3.5 m (Calandria - ID 6900 x 5000 ht)
Reactor operating pressure	70 bar
Core coolant inlet temperature	532.5 K (259.5^{0}C)
Core coolant outlet temperature	558 K (285^{0}C)
Average exit quality	19%
Non-electric application	Desalination - 2650 m^3/day
Design Life	100 years

Table 1: Important design data of AHWR

Reactor Core and Fuel Design

The core consists of total 513 lattice locations arranged in square pitch of 225 mm. There are 452 coolant channel assemblies, 8 absorber rods, 8 regulating rods, 8 shim rods and 37 shut off rods in the core.

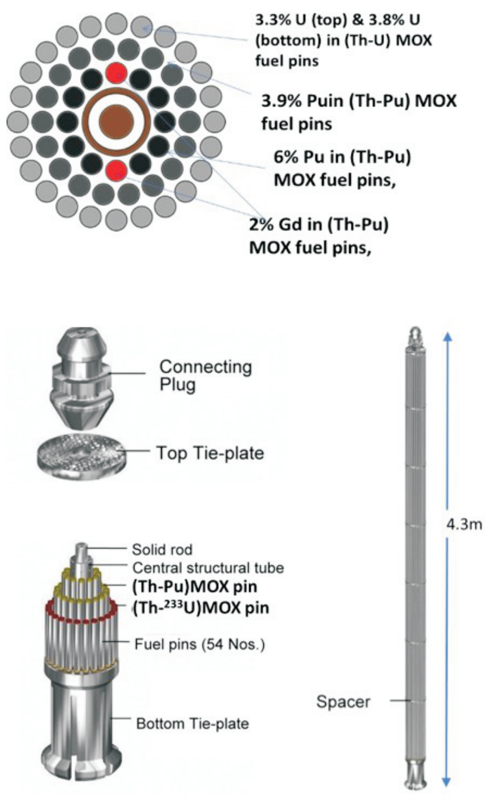


Figure 2: Details of fuel rod cluster of AHWR

The circular fuel cluster of AHWR (Figure 2) contains twenty-four (Th, ^{233}U)MOX pins in the outer ring and thirty (Th, Pu)MOX pins in the inner and middle ring, along with a displacer rod at the centre. The outer ring of (Th, ^{233}U)MOX pins have higher enrichment in bottom half of the cluster to maximise safety margins in operating conditions including normal operation.

Thorium Fuel Cycle Development Related Activities

Atomic Minerals Directorate for Exploration & Research (AMD) has identified 12.47 million tons of Monazite deposits (primary source of Thorium in India) which correspond to about 1 million tons of Thorium oxide. As part of operations of mining and separation of beach sand minerals, Indian Rare Earths Limited (IREL) has processed a few lakhs of tons of monazite ore. The process of producing nuclear grade Thorium oxide, called thoria powder from the monazite ore containing beach sands has been established. The fuel fabrication process for thoria based fuels by powder-pellet method is now well understood. Few tonnes of thoria fuel pellets have been fabricated at BARC and NFC for various irradiations in research and power reactors.

The large-scale utilisation of thorium requires the adoption of closed fuel cycle, wherein, there are challenges on both front-end and back-end. The highly stable thoria poses problems in dissolution in pure nitric acid for reprocessing the spent fuel. This problem is mitigated by small additions of fluoride, which however enhances the corrosion of stainless steel used as the material of construction for equipments. Another major concern with the thorium fuel cycle is the presence of ^{232}U along with ^{233}U. The daughter products of ^{232}U, ^{212}Bi and ^{208}Tl are emitters of hard gamma rays. This requires fuel fabrication and recycling of uranium to be carried out remotely in shielded hot-cells with a high level of automation.

The use of thorium in nuclear reactor is backed by experience of research reactors like PURNIMA, KAMINI, FBTR, CIRUS, Dhruva as well as commercial PHWRs (where it has been used for flux flattening). Reactor physics experiments in AHWR-Critical Facility have been initiated using thoria based test fuel pins. AHWR uses closed fuel cycle and will provide platform for development and demonstration of THOREX reprocessing on an industrial scale and the remote fuel fabrication of the highly radioactive ^{233}U based fuel in shielded hot-cells.

Back end of the fuel cycle is also well understood based on experience of Uranium Thorium Separation Facility (UTSF) and fuel reprocessing at IGCAR, Power Reactor Thoria Reprocessing Facility (PRTRF) etc.

AHWR will demonstrate use of thorium for generation of commercial nuclear power along with development of front and back end activities of thorium fuel cycle.

Safety Philosophy of AHWR

Following the Fukushima accident, more emphasis is laid globally on very low or nil radiological impact in the public domain and no requirement of evacuation in the spirit of “more good than harm”. Nuclear reactors like AHWR come closer to these requirements. With deployment of mostly passive and a few active safety systems, and many inherent safety characteristics, AHWR can manage Design Extension Conditions (DEC) including Anticipated Transients without Scram (ATWS) without any impact in public domain. The broader safety objectives of AHWR are reducing Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) to an insignificant level [4]. The design also has enhanced robustness to malevolent acts. The reactor is designed to provide at least 7 days of grace period during postulated accident events. However, it has also demonstrated its robustness for several beyond design basis accidents like Fukushima event [5].

Defense in Depth is followed as guiding principle in nuclear reactor design and safety, at all levels from operation to severe accident management. Innovation in safety requires adoption of higher degree of inherent safety characteristics or passive safety characteristics instead of depending on Engineered Safety Features (ESFs) at each level. Many First Of A Kind (FOAK) systems are developed for this purpose.

Inherent safety features rely on choice of design concept, laws of nature, materials or internal stored energy etc. Passive safety operate based on natural physical laws like gravity, buoyancy, etc. and may involve engineering mechanisms like flow, valves etc. It can use active components in very limited way to initiate subsequent passive operation. Both kinds of systems do not need operator intervention.

Inherent safety characteristics of AHWR:

- Negative fuel temperature coefficient of reactivity
- Negative power coefficient of reactivity
- Negative coolant void reactivity coefficient of reactivity
- Adequate shutdown margin even without two rods
- Low specific power to facilitate energy removal by natural circulation
- Fuel design characteristics like radial grading in fuel cluster and axial grading (bottom peaking) fuel to have better
- Conventional features like double containment, emergency planning zones etc.
- Thorium as robust nuclear fuel due to higher thermal conductivity, lower thermal expansion coefficient, higher specific heat, higher melting point, lower fission gas release characteristics and better dimensional stability at high burnups

Passive safety systems of AHWR

AHWR relies highly on passive safety systems and natural circulation systems play important role in achieving the safety objective. Natural circulation leads to robust response to deviation from normal conditions as well as to accidental conditions, like elimination of primary pump in MHT system eliminates PIEs due to pump unavailability or valve failures, at the same time providing better economics.

AHWR has large pool of water called Gravity Driven Water Pool (GDWP), which is located near the top of the containment. GDWP serves as a heat sink for some of the passive systems and also acts as suppression pool and a source of water for low-pressure emergency core cooling.

Under LOCA conditions, ECCS injection is followed by core submergence in reactor cavity providing additional effective cooling for extended period. In case of severe accidents including extended SBO, safety features like Calandria vault water and moderator etc. act as ultimate heat sink, and passive end shield and moderator cooling systems help in retarding the process of such accident. Severe accident management systems like core catcher and cooling systems, passive venting of containment, passive catalytic re-combiners ensure that radioactivity is contained.

Human error or malevolent actions by insider, which by nature are difficult to counter, have been addressed in this reactor design. In the very low probability event of failure of both the wired shutdown systems, poison injection through passive means is designed to cover high pressure ATWS scenarios. In such event, MHT system pressure rises, which causes to open passive valves and initiate the poison injection into the moderator. Major passive systems with their safety functions are listed in Table 2.

Experimental Validation Programme

A large scale R&D has been carried out for design validation of various features related to physics, engineering and safety. Some are listed as follows:

- Critical Facility of AHWR - Validation of AHWR reactor physics design.
- Safety assessment of reactor – Experimental Thermal Hydraulics Studies.
- Separate Effect Test Facilities - Stability, CHF, Condensation in presence of non-condensables, Carry Over/Carry Under, Poison Injection & Distribution, Core Catcher, etc.

About 25 major Test Facilities have been established in various Divisions and Groups of BARC, wherein, extensive experiments have been conducted to conclusively validate the design and safety aspects of AHWR. Figure 3 shows some of the installations viz. AHWR Critical Facility (AHWR_CF), Integral Test Loop (ITL) at Trombay and PARTH (Proving Advanced Reactor Thermal Hydraulics) test facility at Tarapur.

Passive Systems with their Safety Functions

Postulated event / safety function	Passive system
Normal operation	Natural circulation in MHT system
Shut Down / Station Black out	Isolation condenser for Decay Heat Removal System
LOCA	1. Passive ECCS injection a. High pressure injection b. Low pressure injection c. Direct injection of ECCS water on fuel 2. Vapour suppression in GDWP 3. Core submergence
Containment cooling	Passive Containment Cooling System (PCCS)
Systems cooling	Passive Moderator Cooling System (PMCS) Passive End Shield Cooling System (PESCS)
Severe accident	GDWP water cooling of core catcher Core submergence in reactor cavity
Response to malevolent action	Passive Poison injection system (PPIS)
Containment Isolation	Passive Containment Isolation System (PCIS)
Hydrogen management	Passive Autocatalytic Re-combiners (PARC)
Depressurisation of MHT	Passive Auto Depressurization System (PADS)
Containment Venting	Passive Containment Filtered Venting System (PCFVS)
Core damage state	Core Catcher

Table 2 Passive Systems with their Safety Functions



Figure 3: Installations having Critical Facility, Integral Test Loop and PARTH

Safety and Engineering aspects

Basic design including reactor physics, fuel design, shielding, fuel handling systems and reactor systems/structures/components has been completed. Safety analysis of AHWR was carried out for an exhaustive list of Postulated Initiating Events (PIE). Probabilistic Safety Analysis (PSA) Level 1, 2 and 3 PSA Analysis has been carried out for AHWR [6]. Design and Detailed Engineering of reactor and auxiliary systems/ structures/ components is being pursued. Plant Design and Plant lifecycle Management (PDPLM) is implemented for integrated plant design of AHWR which can be used for design, layout, construction management, training etc (Figure 4). Feasibility studies and design of steam turbine with associated steam and feed cycle has been carried out. Figure 5 shows bird's eye view of the AHWR plant.

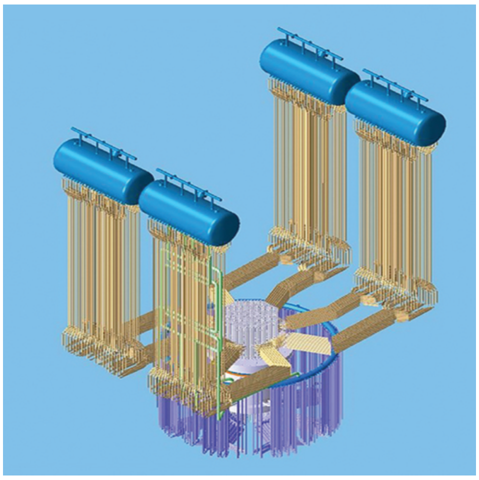


Figure 4: 3D View of Main Heat Transport System

Figure 5: 3D view of AHWR plant

Summary

AHWR will demonstrate industrial-scale use of thorium for the generation of commercial nuclear power. Technology for thorium based fuel fabrication and reprocessing has been demonstrated. The reactor incorporates several First Of A Kind (FOAK) passive safety features. With enhanced safety, AHWR is expected to have minimum impact in public domain.

The development of AHWR has been supported by a robust R&D infrastructure developed at BARC and many experimental facilities have been built to validate AHWR design. AHWR will be important step towards realization of long term energy security for India using indigenous thorium reserves.

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Shaping third stage of Indian nuclear power programme

With high temperature thorium reactors

I.V. Dulera
Members of task force for IMSBR development
Avaneesh Sharma

Our country has large thorium reserves. Therefore, in long term, our nuclear reactors are expected to predominantly use thorium as fuel. Since thorium is fertile, for its utilisation we need external fissile material in the beginning. The reactor concepts, which do not subsequently need fissile material, are expected to provide sustainability to our nuclear power programme. Additionally, nuclear energy assisted production of energy carrier for transport applications is important for the long term energy security of our country. In order to achieve these goals, BARC is developing nuclear reactors. This article summarises the R & D, in progress in BARC, for these reactors.

Indian Nuclear Power Programme

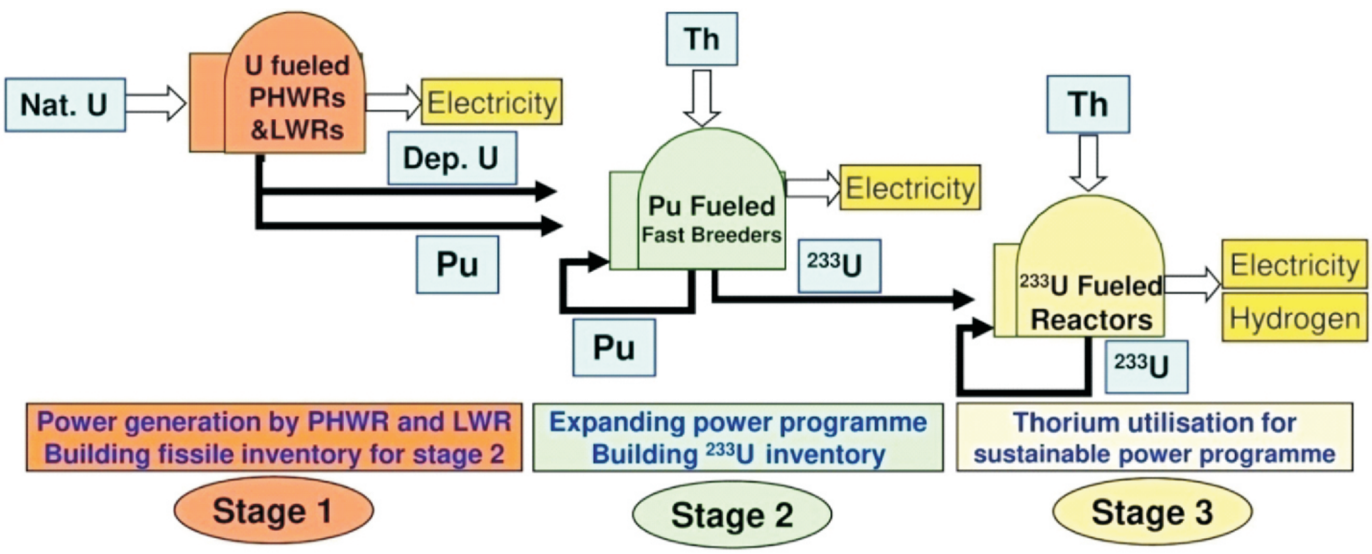
For a large country like India, an optimum mix of diverse energy sources will be required for satisfying the energy demands in future. Environmental concerns limit use of fossil fuel. Nuclear energy, which is a clean source of energy, will have to play a very important role in this objective. The importance of nuclear energy and in particular the role of thorium, as a long term sustainable energy resource for our country, was recognised right at the very inception of our atomic energy programme. Accordingly, Department of Atomic Energy (DAE) formulated and is following a three - stage nuclear power programme (Fig. 1) to efficiently utilise our nuclear resources consisting of modest uranium and abundant thorium. This programme is based on a closed fuel cycle, where the spent fuel of one stage is reprocessed to produce fuel for the next stage. This multiplies manifold the energy potential of the fuel and greatly reduces the quantity of waste.

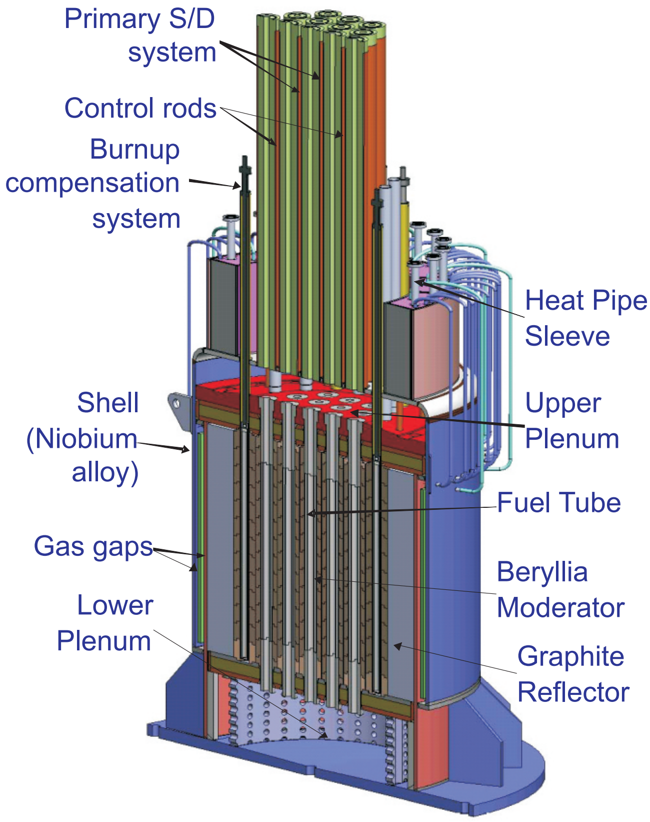
First stage uses natural uranium in Pressurised Heavy Water Reactors (PHWRs) and Light Water Reactors (LWRs, enriched uranium), followed by use of plutonium obtained from the spent fuel of PHWRs in Fast Breeder Reactors (FBRs) in second stage and ^{233}U fuelled systems in the third stage. The first stage, with 18 PHWRs and 4 LWRs in operation and many under construction and planning, has reached a state of commercial maturity. The first commercial FBR of second stage is in an advanced stage of commissioning. When adequate nuclear installed capacity has been built during the second stage, thorium based fuel will be used in FBRs to breed ^{233}U. The third stage of the programme, thus, envisages use of ^{233}U fuelled reactors with thorium as the fertile material.

Importance of third stage of our nuclear power programme

Due to vast thorium reserves in our country, the goal of DAE is to use thorium as the main stay of our long-term nuclear power programme. The reactor systems contemplated and being designed for the third stage will be self-sustaining in Th-^{233}U fuel cycle. It means, after attaining the required level of installed capacity in the third stage, it would be possible to maintain the achieved level of nuclear power programme with thorium alone, without additional demands on uranium or plutonium as fissile material. The ^{233}U required for this purpose would be obtained from operation of Th-^{239}Pu based fast reactors in the latter part of the second stage. The third stage reactors would provide long term energy security to our country. In preparation for the third stage of our nuclear power programme, development of

(1). Three stage Indian nuclear power programme





(2). Compact High Temperature Reactor (CHTR)

technologies pertaining to utilisation of thorium has been a part of ongoing activities in DAE. With the sustained efforts over the years, India has gained experience over the entire thorium fuel cycle-fabrication, irradiation and reprocessing on an engineering scale.

Why India needs High Temperature Reactor (HTR) and Molten Salt Breeder Reactor (MSBR)?

As we have limited reserves of oil and natural gas, our import bills on these is huge and in absence of alternate energy carrier, would continue to increase in future. Their use also has environmental concerns. For the long-term energy security of our country, it has become inevitable that we find an alternative energy carrier for transport applications. Hydrogen is an attractive alternative. While options for production of hydrogen from fossil fuel such as steam methane reforming can satisfy demands during interim periods, nuclear energy assisted hydrogen production, by splitting water, is a long term sustainable and environmentally benign option. Indian HTR programme has this objective.

For India, MSBR is attractive due to its potential to provide breeding for thorium based fuel in thermal or epithermal spectrum of neutron energy. This is mainly because, MSBR being a fluid fuel reactor using thorium based fuel salt, it is feasible to remove intermediate isotope ^{233}Pa, which then decays to ^{233}U, a fissile material, outside the reactor. In a solid fuel reactor, this advantage is lost, as fuel needs to be inside the reactor till it achieves intended burnup. In such a case ^{233}Pa absorbs neutron and ultimately ^{234}U is formed which is not fissile and hence adversely affects breeding ratio. MSBR is, therefore, capable of providing sustainable long term energy security, as is expected from the reactors forming the third stage of our nuclear power programme. Besides efficient conversion of thorium into ^{233}U, MSBR has other advantages also. Some of them are; continuous removal of Xenon and Krypton, resulting in better neutron economy; negative fuel salt reactivity coefficient; sub criticality in case of salt leakage; no violent interaction of salt with water or air; stability of salt under irradiation; online refuelling leading to low excess reactivity; passive decay heat removal; high boiling point of salt resulting in a low pressure system which reduces the risk of a large break and loss of coolant as a result of an accident, thereby enhancing the safety of the reactor; and no significant fuel fabrication. Moreover there is no scenario called 'fuel melt down' and in case of leakage in primary circuit, salt is collected in a non-critical configuration in safety vessel. Solidification of the salt, on leakage, reduces leak and probability of large scale radioactivity release.

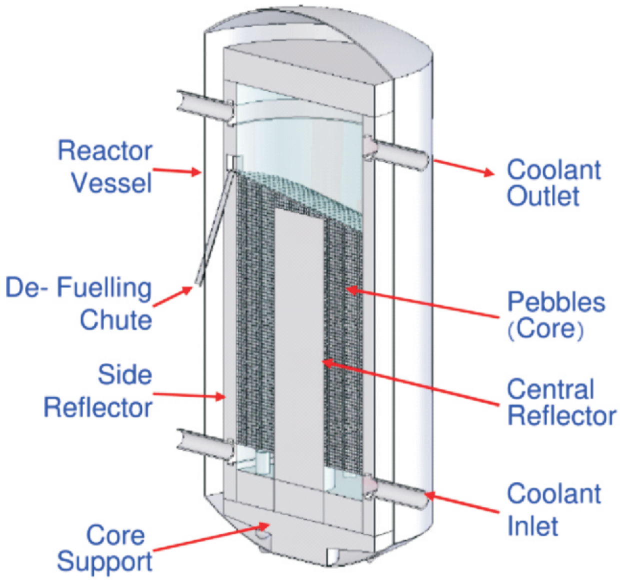
Worldwide status of HTR and MSR development

HTR and MSR both are part of six reactor technologies selected by the Generation IV Forum, an initiative involving 13 countries focused on next generation nuclear power technologies, for further R&D. Majority of the HTRs worldwide, which have been operated or are under design, are gas cooled reactors using high pressure Helium as the coolant. Currently small power (<50 MW) reactors are in operation in Japan and China. Large power HTRs are also under construction in China and under development in USA. Oak Ridge National Laboratory (ORNL) in the United States operated an experimental 7.34 MW reactor.

MSR known as the Molten-Salt Reactor Experiment (MSRE) from 1965 to 1969. This reactor demonstrated the concept of such reactors. Although work on development of MSRs is in progress in many countries, commercial deployments are yet to begin. Besides India, countries where MSR are under development include US, France, China, and Russia. MSRs can be potentially utilised to burn minor actinides which remain radioactive for long time and are of public concern.

Indian High Temperature Reactor Development Programme

BARC is developing a technology demonstrator Compact High Temperature Reactor (CHTR), and a 20 MW_{th} Innovative High Temperature Reactor (IHTR). During the initial developments for HTRs, coolant exit temperature for the reactors was proposed as 1000 °C so as to facilitate hydrogen production from water by Iodine-Sulphur process. As per the studies carried out in BARC for this process, it has been estimated that the high temperature (~ 850 °C) process heat requirement is only about 8% of the overall energy requirement. Balance heat is required at temperatures less than 450 °C. Considering severe challenges and qualification requirements for nuclear reactors at 1000 °C, it was decided to develop the reactors operating at relatively lower temperatures. For CHTR and IHTR, the coolant exit temperatures are 550 °C and 665 °C respectively. The high temperature heat at ~ 850 °C would be supplied through the electrical heaters. CHTR (Fig. 2) is a lead-bismuth eutectic (LBE) alloy cooled, ^{233}U-Th fuelled, beryllium oxide moderated 5 MW_{th} reactor with SS 316L as the structural material with around 5 years long refuelling period. Considering requirements for advanced reactors, the reactor has been provided with several inherent passive features and passive core heat removal systems such as natural circulation of LBE and use of high temperature heat pipes. Studies and developments for this version, called CHTR-B, are in progress.



(3). Innovative High Temperature Reactor

IHTR (Fig. 3) is based on pebble bed fuel configuration with molten fluoride salt as coolant. Earlier conceptual design was done for 600 MW_{th} reactor. In order to demonstrate the concept, it was decided to workout design of a 20 MW_{th} reactor. The reactor, with co-generation capabilities, will also produce electricity for plant load and providing electricity to heaters needed for high temperature segment of the hydrogen production process. Many of the developments carried out for high temperature version of CHTR has applications in IHTR and similar other high temperature applications.

Indian Molten Salt Breeder Reactor

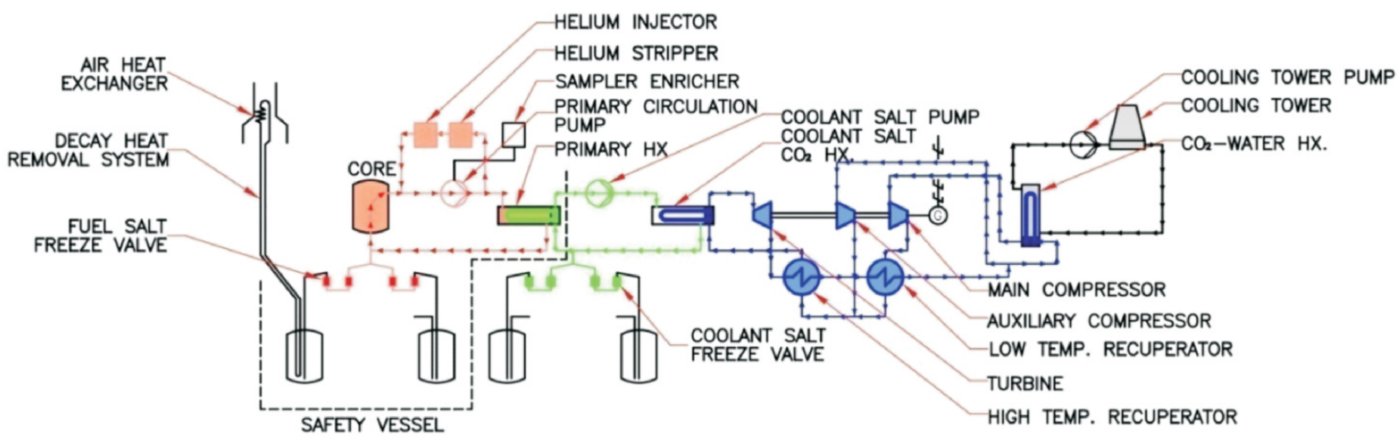
Attractiveness of MSBR in Indian context prompted initiation of R&D in BARC. Earlier, conceptual design of an 850 MW_e reactor was carried out with an aim to identify the R&D areas. Considering the large number of new technologies involved, a 5 MW_{th} demonstration IMSBR is being designed and the required technologies are being developed in BARC. This reactor (Fig. 4) uses LiF-CaF_2-ThF_4-UF_4 as fuel salt, which is circulated by pump through the reactor core containing graphite moderator blocks. The heat gained is transferred to the coolant salt (i.e. secondary side) by means of an intermediate heat exchanger. The coolant salt, LiF-NaF-KF, transfers the heat to the super critical CO_2 (SC CO_2) based Brayton cycle (SCBC) for power generation via a salt to SC CO_2 heat exchanger. SCBC based electricity generation system has been chosen as it operates at relatively lower pressure and results in higher efficiency. Ni-Mo-Cr-Ti based alloy has been indigenously developed as material of construction for all components wetted by molten salts. When the reactor is under shutdown, the fuel salt will be transferred to dump tanks from where the decay heat will be removed by passive means. The fuel salt circuit is contained within a nickel lined steel safety vessel. In an improbable case of leakage of fuel salt, it will be contained in this vessel in a subcritical configuration, while being cooled by passive means.

Technology challenges for HTRs and IMSBR

HTRs and MSBR are designed to operate at high temperatures, use different kind of fuel, coolants, and energy conversion systems. Therefore they pose a large number of technological challenges. These challenges are being addressed, mainly in BARC, through detailed R&D programmes. The specific challenges for these reactors have been briefly described in the following paragraphs:

For reactor physics analysis of long life compact CHTR core and pebble based moving fuels for IHTR, new computer codes have been developed. For understanding thermal hydraulics of the coolants, experimental and analytical studies have been carried out. LBE loops have been operated both at 550 and 1000 °C. For compatibility with LBE at high temperatures, Nb based alloy was developed with the help of NFC, Hyderabad. For HTR fuel, technology for TRISO (TRi-ISOtropic) coated particle based fuel was developed. Development of in-core ceramic material like beryllia, carbon-carbon composite etc. and graphite machining was carried out. For graphite, oxidation studies, large scale characterisation to develop design rules, and development of oxidation resistant coatings were carried out. For CHTR, passive heat removal by natural circulation of coolant under normal operation and passive heat transfer from primary to secondary

(4). 5MW_{th} IMSBR Flowsheet



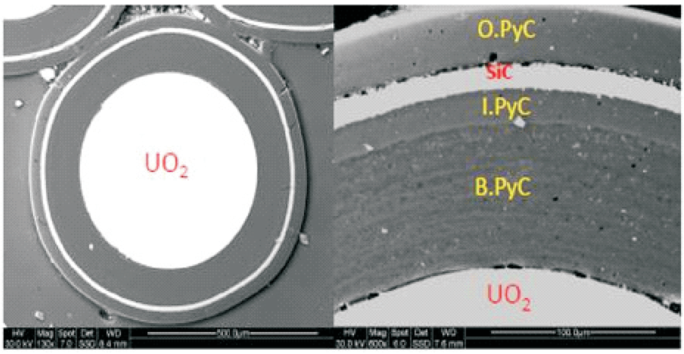
system by high temperature sodium heat pipes was experimentally demonstrated. Studies are in progress for decay heat removal by passive means. For IHTR, development of SCBC based power conversion system and interface with hydrogen production system are in progress. Development of heat exchangers, pump, instrumentation, and other components for the molten salt environment are also in progress.

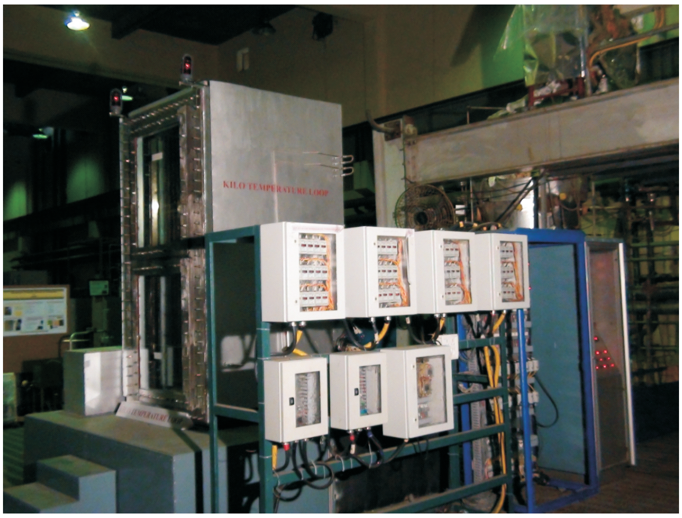
The reactor physics analysis of fuel circulating IMSBR needs closely coupled neutron transport and CFD codes with capability to account for online reprocessing. Existing computer codes are being modified and new codes are being developed to model the neutronic behaviour of such fluid fuel reactors. Developed codes have been validated against the experimental data from literature. A 3D kinetics code with fuel salt flow and multi-point heat transfer model has been developed. A computational tool is being developed, which can model refueling/reprocessing. To understand behaviour of fuel and coolant salts, chemistry studies such as thermophysical properties, solubility etc. are in progress. Purification facility for coolant salt has been set up, and setting up of similar facility for fuel salt is in progress. For handling of molten fluoride salts, inert gas glove boxes have been setup. Facilities for production of ThF_4 and UF_4 from their respective oxides have been setup. The reactor needs ^7Li based fluoride salts. A process is being developed for enrichment of lithium-7 at high throughput rates. An indigenous Ni-Mo-Cr-Ti alloy, which is compatible to molten fluoride salts under reactor conditions, has been developed in collaboration with MIDHANI. It's scaled up production and fabrication of engineering shapes is in progress. R&D is in progress for development of high density isotropic nuclear grade graphite. Design and technology development for pumps, valves, flow meter, off-gas system, dump tanks, helium injector and stripper, intermediate heat exchanger, salt-CO_2 heat exchanger etc. is in progress. Design and studies are also in progress for SCBC components. Facilities for studies on interaction of supercritical CO_2 with Ni-Mo-Cr-Ti alloy and high speed seals are being designed. Instrumentation and sensors are also under development for operation at high-temperature, high-radiation, molten salt environment. Studies are in progress for computational modelling of molten fluoride salts using molecular dynamics. This is expected to reduce experimental work, especially for the salt with fission products. Evolution of safety philosophy and safety studies, batch mode offline reprocessing studies without requiring cooling of fuel salt, reactor control, remote inspection, and qualification of new materials to meet codal design requirements are being initiated.

Current status and way forward

As mentioned above, for CHTR, besides designing the reactor, experimental thermal hydraulic studies, development of TRISO coated fuel on natural urania, BeO, graphite oxidation studies, development of high temperature sodium heat pipes have been carried out. Manufacturing of components for a lower power electrical heated facility of the reactor is in progress. Subsequently this facility will be set up to demonstrate integral behaviour of the reactor. The design of the demonstration IHTR is in progress.

The interface of this reactor with the hydrogen production plant and safety issues arising out of combined operation of IHTR with a hydrogen production plant is being studied. Development of pebble based fuel from TRISO coated fuel particles is being initiated. Due to use of fluoride salts, many developments such as structural material, graphite, salt related components, instrumentation, sensors etc. are similar as those for IMSBR, and are being developed together. This results in overall savings for technology development efforts. The detailed design of the 5 MW_{th} IMSBR is in progress. This will serve as test bed for salts,

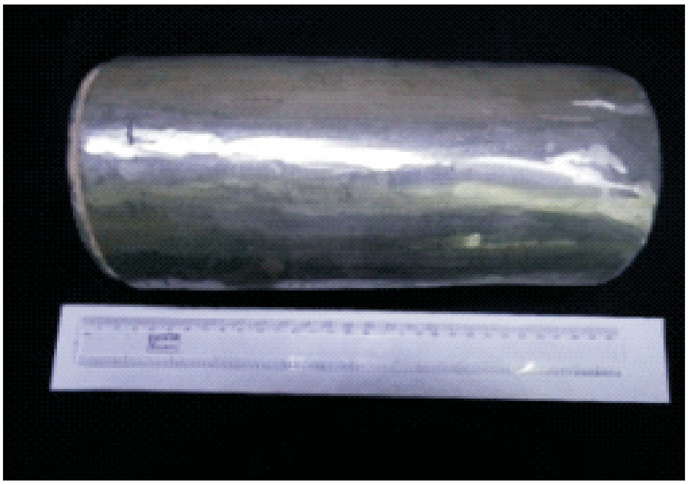




(5). SEM images of TRISO coated particle fuel (Top)

(6). LBE thermal hydraulic studies (Middle)

(7). Graphite oxidation studies (Bottom)



materials, reactors systems, instrumentation, reactor physics and chemistry studies. Operation of this demonstration reactor will help in finalizing the design of the large power IMSBR.

It is planned to demonstrate all the major equipments and systems of this reactor at full scale under actual operating conditions from temperature and salt chemistry point of view. For this purpose, a dedicated facility, Molten Salt Breeder Reactor Development Facility (MSBRDF) is being designed for setting up at Visakhapatnam. Pre-project activities for these facilities are already in progress.

(8). Glovebox for handling molten fluoride salts (Top)

(9). Indigenously developed Ni based alloy (Middle)

(10). UF_4 preparation facility (Bottom)

Task force members for IMSBR development and acknowledgements

Dr. A.K. Tyagi, Dr. R. Mishra, Dr. (Smt.) Nandita Maiti, Dr. V. S. Tripathi of Chemistry Group; Dr. R. Tewari, Dr. S.K. Satpati, Dr. K. Dasgupta of Materials Group; Dr. Anurag Gupta, Shri Rajesh Kumar, Dr. A. Borgohain, Shri A. Basak of Reactor Design & Development Group; Dr. D. Mandal, Dr. Sk. Musharaf Ali of Chemical Engineering Group; Dr. S.C. Parida, Dr. (Smt.) Aparna Banerjee of Radiochemistry and Isotope Group; Dr. R.K. Mishra, Shri V.P. Patel of Nuclear Reprocessing Group; Dr. B. Chakraborty of Physics Group.

For various technology developments for HTRs and IMSBR, large numbers of experts have contributed and continue to contribute. Work mentioned in the article is due to the untiring efforts of these experts and guidance from seniors. Authors thankfully acknowledge contributions from the experts in BARC as well as NFC, MIDHANI, and RRCAT.

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Research Reactors in BARC

History, Development & Utilization

Kunal Chakrabarty & C.G. Karhadkar

Research Reactors (RRs) are central to the development of nuclear science and technology programme for societal benefits. BARC has a long history of more than 65 years in designing, constructing and safely operating RRs of various type and size. Systematic ageing programme have been put in vogue to refurbish and extend the life of RR. The article describes in brief the history of RR at Mumbai, BARC, their progressive development and various aspects of utilization for societal benefits.

Research reactors (RRs) in BARC have been the forerunners for the nuclear energy programme of the country. They are primarily meant to provide neutron source for research and applications in healthcare, neutron imaging, neutron activation studies, neutron scattering experiments etc. Generally, any upcoming technologies are proven in a RR before their implementation in commercial power reactors. RRs are generally simpler and smaller than power reactors with power level varying from zero power (like Critical Facility for AHWR) to few hundreds of MW (like Dhruva), and generally operate at low temperature. They use far less amount of fuel than a power reactor, but their fuel may require Uranium with much higher enrichment like that in Apsara-U. They may have a very high power density in the core, requiring special design features. Being flexible, RRs are best suited for testing nuclear fuels of various reactor types, studying the safety margins for nuclear fuel, and developing accident tolerant and proliferation resistant fuel for future reactors.

Unlike power reactors having standardized design, different RRs have distinct designs and operating modes. A common design is a pool type reactor like Apsara, where the core is a cluster of fuel elements housed in a large pool of water. In a tank type reactor like Dhruva and Cirus, core is contained in a closed vessel as in the power reactors. In tank-in-pool type reactor, the core is enclosed in a tank which is in turn located in a pool of water. Most of the RR cores have channels to locate materials for irradiation experiments. In addition, beam tubes which penetrate the reactor vessel, pool and shielding provide neutron and gamma beams for experimental use in reactor hall or adjoining guide tube laboratory.

Around the world, a total of 818 research reactors have been built so far, out of these 443 have been decommissioned and 224 reactors are in operation. Russian Federation has the highest number of operational RRs (63), followed by USA with 42 reactors, China with17 and France 10.

The very first research reactor of Asia, named Apsara, was commissioned in BARC in the year 1956. It was a 1MW, swimming pool type reactor fuelled with enriched uranium–aluminium alloy clad with aluminium. The reactor core was housed in a stainless steel lined pool of 8.4 m long, 2.9 m wide and 8 m deep, filled with demineralized light water. The core, suspended from a movable trolley, could be parked at three positions to facilitate wide range of experiments at beam tubes, thermal column and a shielding corner in addition to the in-core irradiation. It produced an average neutron flux of 10^{12} n/cm^2/sec.

Apsara enabled the Indian scientists and engineers to understand the complexities and intricacies of operating a nuclear reactor safely. Simplicity of this reactor design had made it very popular among the researchers. Various experiments could be planned and carried out with relative ease, as the reactor core was easily accessible and movable. The thermal column and the shielding corner facilities in the reactor made it very versatile for carrying out experiments. Facility for irradiation of targets with fast neutron alone was also available in Apsara. In a span of around 50 years, the reactor had been instrumental in carrying out advanced studies in the field of neutron physics, fission physics, radio chemistry, biology, irradiation techniques and R & D work on reactor technology. Neutron activation analysis technique developed with Apsara found wide applications in chemistry, archaeology and forensic sciences. Various shielding experiments to verify the design adequacy of shield configurations used in reactors such as Dhruva, PHWRs, 500 MW$_e$ Prototype Fast Breeder Reactor etc had been carried out in the shielding corner of Apsara. The reactor was shut down for good in the year 2009.

Subsequent to the experience with Apsara, need for high power research reactors which cater to the additional requirements of radioisotope production, irradiation facilities etc. was felt. This led to the construction of a high flux and high power research reactor known as Canada India Reactor (CIR). This reactor, built with the help of AECL, Canada, was similar to Canadian NRX reactor, but with few changes based on the location and requirement. CIR was later renamed as Cirus by Dr. Bhabha. It was a vertical tank type 40 MW$_{th}$ reactor. The reactor was natural uranium fuelled, heavy water moderated, graphite reflected and light water cooled. It produced a flux of 6.5 x 10^{13} n/cm^2/sec.

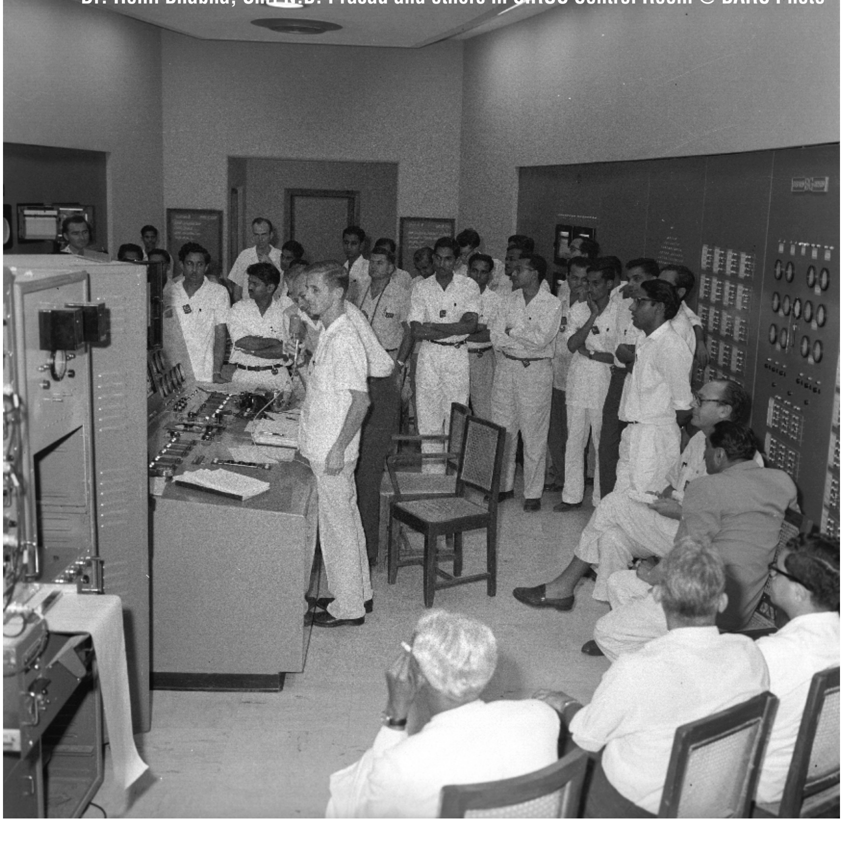
Cirus became the work horse of the nuclear energy program, as it provided larger irradiation volume at larger flux. Cirus reactor was solely catering to the country's radioisotopes requirements till Dhruva became operational in 1985. The reactor had a pneumatic carrier facility, where short term irradiations can be carried out. This facility was extensively used for activation analysis, for determining trace quantities of materials in a given sample. The reactor had also been used for silicon doping experiments, much needed for electronics industry. Cirus had a set of six self-serve units, in which on-power irradiation of 30 samples can be done simultaneously, for production of short-lived isotopes.

In order to utilize and develop Thorium fuel technology, irradiation of thorium was started in graphite reflector region of Cirus very early. The first charge of fuel for Kamini reactor was produced by irradiating thorium in Cirus. An in-pile Pressurized Water Loop (PWL) of 400 kW heat removal capacity operating at a pressure of 115 kg/cm^2 and temperature of 260°C was available at Cirus which was a valuable facility for test irradiation of power reactor fuel and materials. Utilizing this facility, development of MOX fuel for Tarapur BWR fuel program was taken up. This facility was also utilized for validating various design assumptions and analysis by carrying out test irradiations and later examining the fuel. Irradiation of various structural materials of the power reactors such as end shield and Zircaloy pressure tubes of PHWRs etc. were carried out at Cirus PWL. These experiments built the confidence for designing and operating power reactors.

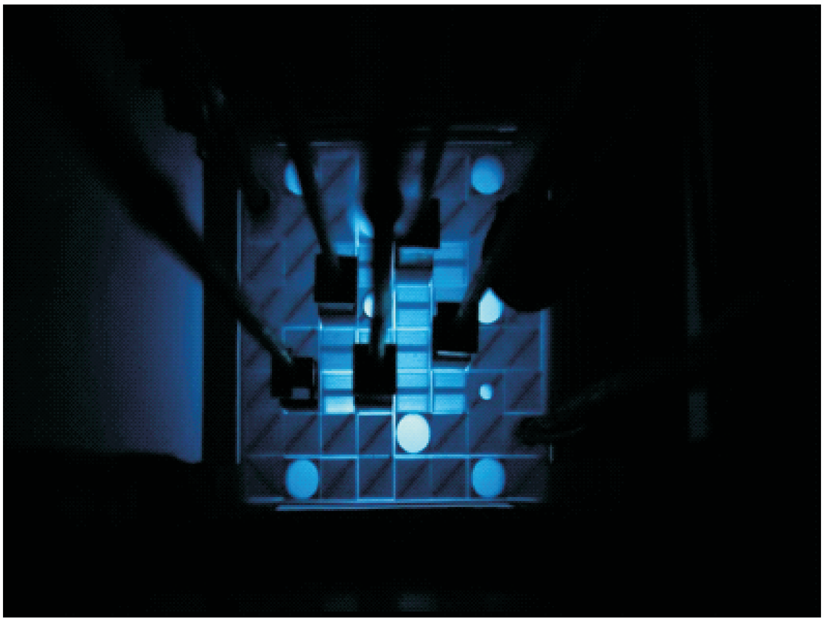
In those days Cirus and Apsara became centres of excellence in nuclear education. People who got early experience in operating these reactors, later grew to lead various nuclear projects and programs. After four decades of successful operation, detailed ageing studies were carried out in Cirus, which indicated possibility of substantial life enhancement by carrying out refurbishment of identified systems, structures and components. Refurbishment of the reactor was taken up during1997 to 2002. Along with, this major safety upgrades were also carried out to meet present safety standards. After operating the reactor for another 8 years, it was permanently shut down on 31st December, 2010 to honour the Civil Nuclear Deal.

During the early seventies, strong need was felt to build a research reactor with further higher neutron flux to meet the growing demand of radio-isotope production and advanced research in basic sciences and engineering. Accordingly, a high flux research reactor of 100 MW$_{th}$ capacity was designed,

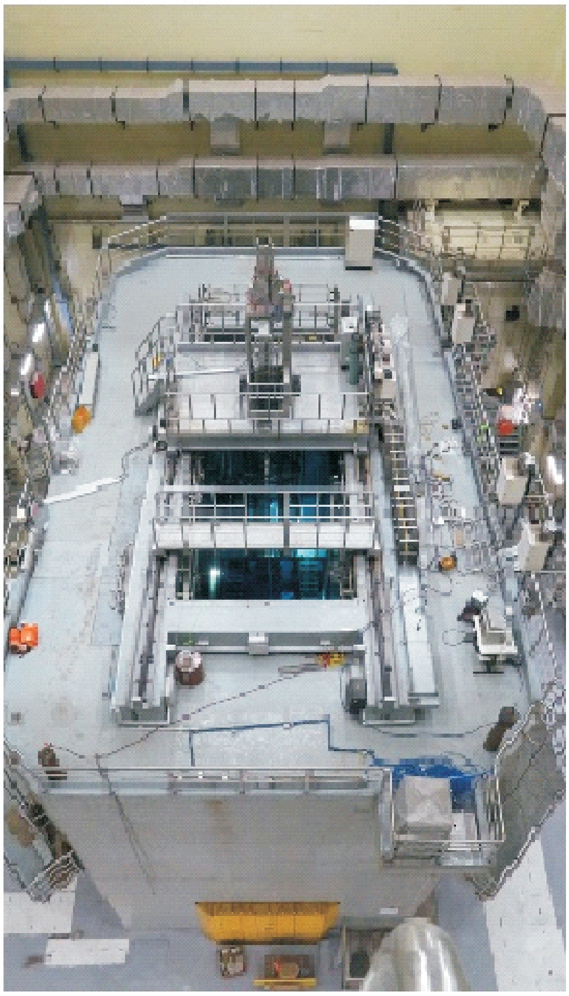
Dr. Homi Bhabha, Shri N.B. Prasad and others in CIRUS Control Room @ BARC Photo



Apsara-U Facility



Cherenkov Radiation during Power Operation



Reactor Pool Top

constructed and commissioned indigenously. Originally named the R-5, and subsequently renamed as Dhruva by the then President of India Dr. Gyani Zail Singh, this reactor first went critical on 8 August 1985.

Dhruva is a 100 MW_{th} reactor with metallic natural uranium as fuel, heavy water as moderator, coolant and reflector, giving a maximum thermal neutron flux of 1.8 x 10^{14} n/cm^2/ sec. Many of the reactor structure designs were a forerunner; to be adopted for the standardized Indian PHWR being designed at that time. Manufacturing of the reactor vessel which is over 7 meters in height, 3.72 meters in diameter and weighing about 30 tons was taken up in the central workshop of BARC. Many evolving technologies such as plasma arc cutting of thick (50 mm) stainless steel plates, precision welding and electron beam welding etc. were developed and successfully employed for the fabrication of reactor vessel. Fabrication of the 300 mm diameter beam hole re-entrant cans for the use in neutron beam research posed another big challenge. A cold rolling facility was set up at MIDHANI Hyderabad where the zircaloy-2 plates were cold rolled to the requisite uniform thickness. Technology for electron beam welding of zircaloy-2 plates in a glove box under argon atmosphere was developed in DRDL Hyderabad. Development of rolled joints for 300 mm diameter between SS and zircaloy-2 beam tubes was a major developmental activity. Designing of the re-fuelling machine for safe and reliable operation was another challenging work. The machine carrying fuel assemblies was to make a leak tight joint with the coolant channel and continue cooling of the fuel during transit. For this the fuelling machine, having lead shielding and weighing over 300 Tons is to be aligned with the channel with in an accuracy of ± 0.25 mm.

The design, construction, commissioning and operation of Dhruva have been a completely indigenous effort. In addition to the engineers and scientists of BARC, several governmental institutions and public sector and private industrial organizations in the country have participated in the above, meeting very stringent requirements. This high flux reactor which was designed, constructed and commissioned entirely indigenously reflects the country's resolve to achieve self-reliance in nuclear technology. For last 35 years Dhruva has been extensively utilised for engineering and beam tube research, testing of equipment and material and large scale production of isotopes.

Apsara-U is an upgraded version of the Apsara reactor, with 2 MW rated power. Here the reactor core is replaced with Low Enriched Uranium (LEU) in the form of U_3Si_2 dispersed in aluminium matrix as fuel to meet the international requirement. The core is surrounded by two layers of beryllium oxide reflectors. The reactor core is suspended from a movable trolley and can be parked at three reactor core positions inside the pool like the old Apsara reactor. The maximum thermal neutron flux is enhanced to 6.1×1013 n/cm^2/s in the core region and maximum thermal neutron flux in reflector region is increased to 4.4×1013 n/cm^2/s. Maximum fast neutron flux is 1.3×1013 n/cm^2/s. The higher neutron flux facilitates production of isotopes for applications in the field of medicine, industry and agriculture. The Apsara-U also provides enhanced facilities for beam tube research, neutron activation analysis, neutron radiography, neutron detector development & testing, biological irradiations, shielding experiments and training of scientists and engineers.

All the systems and components of Apsara-U are designed and manufactured to meet enhanced power level for better utilization of the reactor adhering to the latest safety codes and standards. The direction of primary coolant flow through the core has been made downward to avoid mixing of radioactive water directly with the reactor pool. The water from core outlet is sent through a delay tank to reduce the radiation field in the process equipment room to a reasonably low value, and the major short-lived activities are allowed to die down before the water is circulated back to the reactor pool. Additionally, a hot water layer is provided on the top of the pool water, which reduces the radiation field at the pool top to acceptably low values. Emergency power supply is provided for all the safety related equipment for the reactor.

Apsara-U reactor core is mounted on a 140 mm thick aluminium grid plate having 64 lattice positions arranged in 8 x 8 square array with a lattice pitch of 79.7 mm. The central 4 x 4 lattice positions of the core are loaded with fuel assemblies and are surrounded by two layers of BeO reflector assemblies. The core has two types of fuel assemblies, viz. Standard Fuel Assembly (SFA) with 17 fuel bearing plates, and Control Fuel Assembly (CFA) with 12 fuel bearing plates. Various types of reflector assemblies are designed to satisfy the requirements of positioning of various components in the reflector region such as fine control rod, irradiation positions, fission counters, thermocouple in addition to reflecting the neutrons towards the core standard BeO reflector assemblies.

The reactor power is controlled by means of two Control-cum-Shut-Off Rods (CFA-CSRs) and one Fine Control Rod (FCR). The shutdown of the reactor is also achieved by the CSRs. In addition, two Shut-Off Rods (SORs) are also provided to shut down the reactor. The CSRs and SORs are located inside control fuel assemblies. Twin fork type control element has been developed using Hafnium plates to cater to the control and shut down requirements.

In order to ensure the fuel safety, the coolant velocity has been so chosen that for the hottest standard and control fuel assembly and shut-off rod assembly, the fuel meat and clad temperatures do not exceed the prescribed limits. Accordingly, the primary coolant flow has been augmented with secondary flow to meet the high heat removal requirement. A natural circulation valve is developed to meet the requirements of core cooling in case forced circulation is not available due to failure of the primary coolant pump.

Research Reactors offer a diverse range of applications such as neutron beam research for material studies and non-destructive examination, neutron activation analysis to measure very small quantities of an element, radioisotope production for medical and industrial use, neutron irradiation of fuel and structural materials for advanced nuclear power plants, neutron transmutation doping of silicon, etc. Besides, RRs have contributed significantly in education and training of operators, maintenance staff, radiation protection and regulatory personnel, students and researchers.

Neutron Beam Research

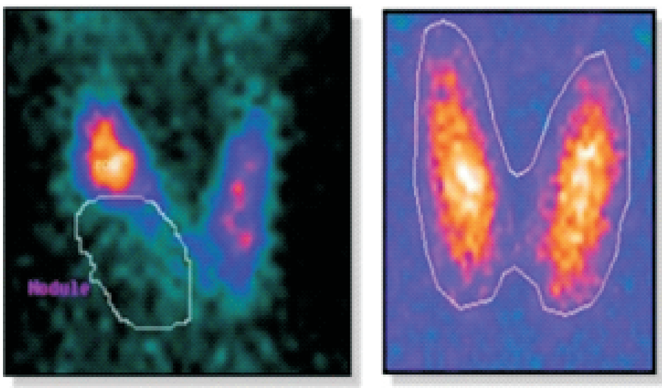
The internal structures of matter at microscopic and atomic levels are very important to understand as they determine macroscopic properties of a material including how they react. The short range strong interaction of neutron with matter and its inherent magnetic moment makes neutron scattering a unique probe to analyse solid and condensed fluid matter. An important advantage of neutron over other forms of radiation is that neutron, being neutral in charge, can penetrate the bulk of materials. The incident monochromatic neutrons are scattered without a change in their energy i.e. elastic scattering which informs about the arrangement of atoms in materials. When the neutrons undergo inelastic scattering i.e. a change in their energy during scattering, they can yield information about the dynamics of atoms. By performing neutron scattering, biologists understand proteins essential for the functioning of brain; how bones mineralise during development or how they repair or decay with age. Physicist can create more powerful magnet that could be of use in accelerators or levitated transport. Chemist improves batteries and fuel cell. Material scientist can improve steel for use in aircraft, nuclear reactor and many other challenging applications.

Radioisotope Production and Applications

A stable material can be made radioactive by bombarding it with neutrons in a reactor. The radioisotopes, thus produced, can be widely used for societal benefits especially in industry and medicine.

Radioisotopes are now considered indispensable in the diagnosis of a variety of diseases and also in therapy. In diagnosis, two types of techniques are employed, the first one being the in-vivo techniques, where the patient is administered a radio-pharmaceutical either orally or intravenously. The distribution of the injected radio-pharmaceuticals in different organs/metabolic pathways is studied from outside the body by using a suitable radiation detector such as gamma camera. Such techniques provide images of the organ function. Thus the procedure not only provides anatomical information but also the more important functional information about the organ.

Neutron Transmutation Doping (NTD)-Si, which is the process of creating non-radioactive dopant atom from the host Si atoms by thermal neutron irradiation and its subsequent radioactive decay, has been used extensively in manufacturing of



Thyroid Images using ^{99m}Tc
Normal scan (left) and an abnormal scan (right)with poor uptake in the left lobe, due to nodules

Buddha relic (original)

Neutron Tomography of Buddha relic: Wax remnants on surface due to bronze casting process were detected (last)

high power semiconductor devices. The quality of NTD-Si, both from the viewpoints of dopant concentration and homogeneity has been found superior to the quality of doped silicon produced by conventional methods.

To summarize, our research reactors have made tremendous contributions to almost all facets of science and technology. However, one of the bigger technological challenges is to refurbish them, not only to meet present day safety requirements and technology standards, but also to enhance their useful life. In this regard, a systematic ageing programme is required to be established at an early stage of reactor operation. Dhruva has already completed more than 35 years and it has been decided to refurbish the reactor for long term operation of at least 20-25 years. Main challenges involve safety evaluation and upgrades, remote handling techniques for inspection of inaccessible components etc. The expertise gained during refurbishment of Cirus would help in implementing this ambitious programme in a long way.

Sonneting Critical Heat Flux: Unexplored science in boiling flows

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Abstract

In water-cooled reactors, the operating heat flux is limited by phenomena known as critical heat flux (CHF). Even though CHF phenomena have been extensively investigated in the past, the flow physics at low pressure and low flow conditions (important during accidental conditions of LWRs and start-up of Natural Circulation Boiling Water Reactors (NCBWRs)) are not understood. To clarify the physics of CHF at such conditions, we have carried out a series of experiments. We found that at CHF, the classical bubbly flow changed to slug/churn flow due to an increase in bubble coalescence. With time, this flow pattern was found to change to an unusual reverse annular flow leading to an increase in the heater surface temperature, as the water did not come in contact with the heater surface. However, this new flow pattern could not sustain and was found to break down, allowing the liquid to touch the heater surface, resulting in quenching of the heater surface.The quenching heat removal rate was found to be significantly higher than the steady-state heat removal rate. Shortly after the quenching, the flow pattern was found to revert to bubbly flow again, and the phenomena repeated rhythmically with the wall temperatures continuously rising and falling in every cycle till the heater trip-setpoint was reached. We coined this interesting phenomenon as "Sonneting CHF". Interestingly, at CHF, unprecedented quenching scenarios were found to occur even though the heater surface temperatures were substantially higher.

Keywords: Sonneting critical heat flux, Subcooled flow boiling, Boiling crisis, Low pressure and low flow conditions

Introduction

Boiling is a common phenomenon in every household. Our ancestors started using the boiling process for cooking even before the invention of pottery via the quenching process, where stones were heated and dropped directly into water[1]. Owing to its ubiquitous nature, it comes as no surprise that every person has some degree of experience with the boiling process and steam production. In the 1700s, steam engines played a prominent role in the industrial revolution. However, it was not until the late 1800s, when the boiling process made its inroads into power industries where steam production started in conventional boilers. Most of the present generation's power industries rely upon steam to run turbines. In addition to steam production, some degree of the boiling process is generally used in many systems because of its good heat transfer characteristics. With a small temperature difference between the hot surface and cooling medium, excellent heat transfer can be achieved during boiling. It is not an understatement to say that boiling is the single most widely used process that runs the 21st century. The rate at which steam is produced is directly proportional to the heat input. Engineers aim to maximize the steam output for a given system.

In 1934, Nukiyama[2] was working on a project to maximize steam production. He conducted experiments on an electrically heated metallic wire immersed in a pool of water at 100°C. He observed that the wire burned even in the presence of water! This mysterious phenomenon is known as boiling crisis or critical heat flux (CHF).In day-to-day appliances, it is ensured that heat input is not so high as to cause damage. This is because beyond a certain heat input, the heat transfer can deteriorate siginificantly. Generally, the heat transfer from the surface to the cooling medium is measured in terms of heat transfer per unit surface area, also known as heat flux. The limited heat flux beyond which the heat transfer deteriorates is known as CHF. It is a practical limit on the maximum heat transferrable; it depends on various operating conditions, like flow, pressure, temperature, etc. Hence, CHF knowledge is essential to design systems that utilize the boiling process to its fullest potential for heat transfer and steam production. However, in the absence of CHF data, engineers design systems that operate at low heat fluxes to avoid system failure. Such a design is said to be conservative. Conservative designs are neither economical nor efficient. It is important to note that CHF is a design safety limit in nuclear reactors. A majority of the research is dedicated to ensuring adequate margins against CHF.

Classically, flow boiling CHF is broadly classified into two categories depending on the flow conditions: Departure from Nucleate Boiling (DNB) and liquid film dryout. Over the years, many experiments were performed at various operating conditions to understand boiling limits[3]. Usually, during DNB, the bubbles crowd near the heater surface and prevent the liquid from touching the surface, reducing

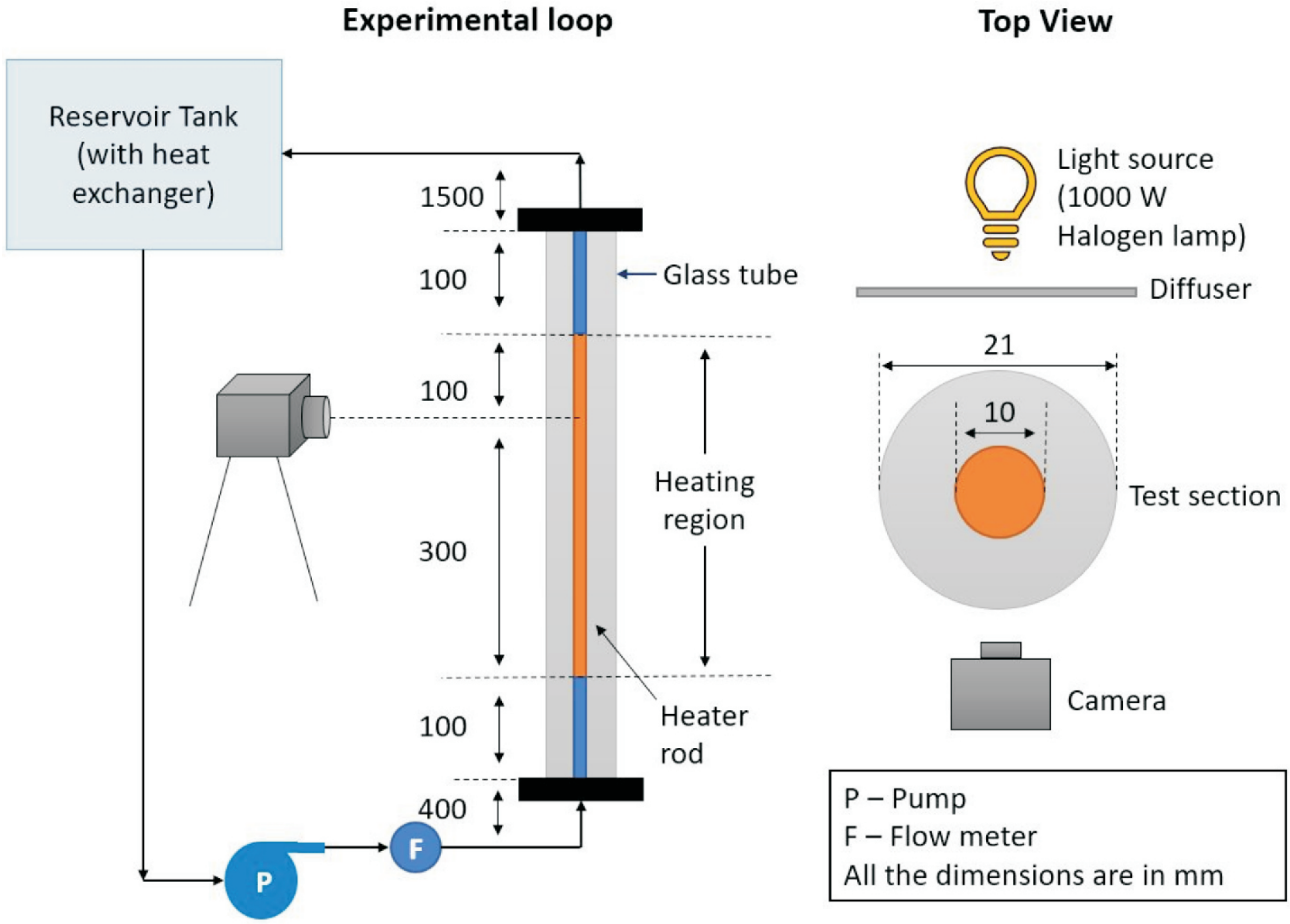

Fig. 1: Experimental Setup (Schematic)

heat transfer and resulting in a sudden increase in the heater surface temperature. The flow physics at CHF has perplexed researchers for many years; it is particularly true, especially at low mass flux conditions, when the operating pressures are low. In literature, many experiments were reported on CHF at low-pressure and low-flow conditions in round tubes[4,5], annular channels with internal heating[6-8], and in rectangular channels with one side and two sides heating[9]. To save the heaters in these experiments, researchers in their experiments mostly kept the heater trip-setpoint at about 50 to 100 K above the water's saturation temperature[6-8]. Hence, the flow physics at CHF, i.e., when the heater temperatures are relatively higher, has been least explored. To understand the flow physics at CHF, especially at low flow and pressure conditions, which are important during accidental conditions of water-cooled reactors and start-up conditions of natural circulation BWRs, we conducted a number of experiments. Details are highlighted below. The results are already reported in the Physics of Fluids journal[10] by the authors; only an abridged version is reported here.

Experimental Setup

Experiments were carried out in an annular test section, a glass tube with an electrically heated rod at the center, with an increased temperature trip-setpoint. The experimental setup consists of a forced convective loop comprising of a centrifugal pump, test section, and a reservoir tank facilitated with cooling coils to act as a heat exchanger (Figure 1). The heater surface, water inlet, and outlet temperatures were measured using 0.5 mm k-type thermocouples using the Yokogawa Data Acquisition system. The flow rate was measured at the inlet using a rotameter. Using Mikrotron Motion BLITZ Cube 4 high-speed camera, the boiling flow patterns were recorded. All the experiments were conducted at atmospheric pressure using demineralized water as a working fluid in the mass flux range of 150 kg/m^{2}s–200 kg/m^{2}s with an inlet temperature of 28°C, which are relavant for low pressure and mass flux conditions of LWRs.

Results and Discussions

In the experiments, the heat flux was increased in small steps until an abrupt rise in heater surface temperature was observed, and the corresponding heat flux is defined as CHF. In our experiments, a similar sudden rise in heater surface temperature was observed at CHF; however, surprisingly, the high heater surface temperature was not sustained and got quenched in a short span (Figure 2a and Figure 2b), leading to a sudden decrease of heater surface temperature. This process repeated for a few cycles until the rise in the heater surface temperature reached a very high value (close to 400°C), where the heater power was tripped as a measure of safety to prevent damage to the test section. Several types of flows (flow patterns) occurred in a span of few seconds. Due to rhythmic flow pattern

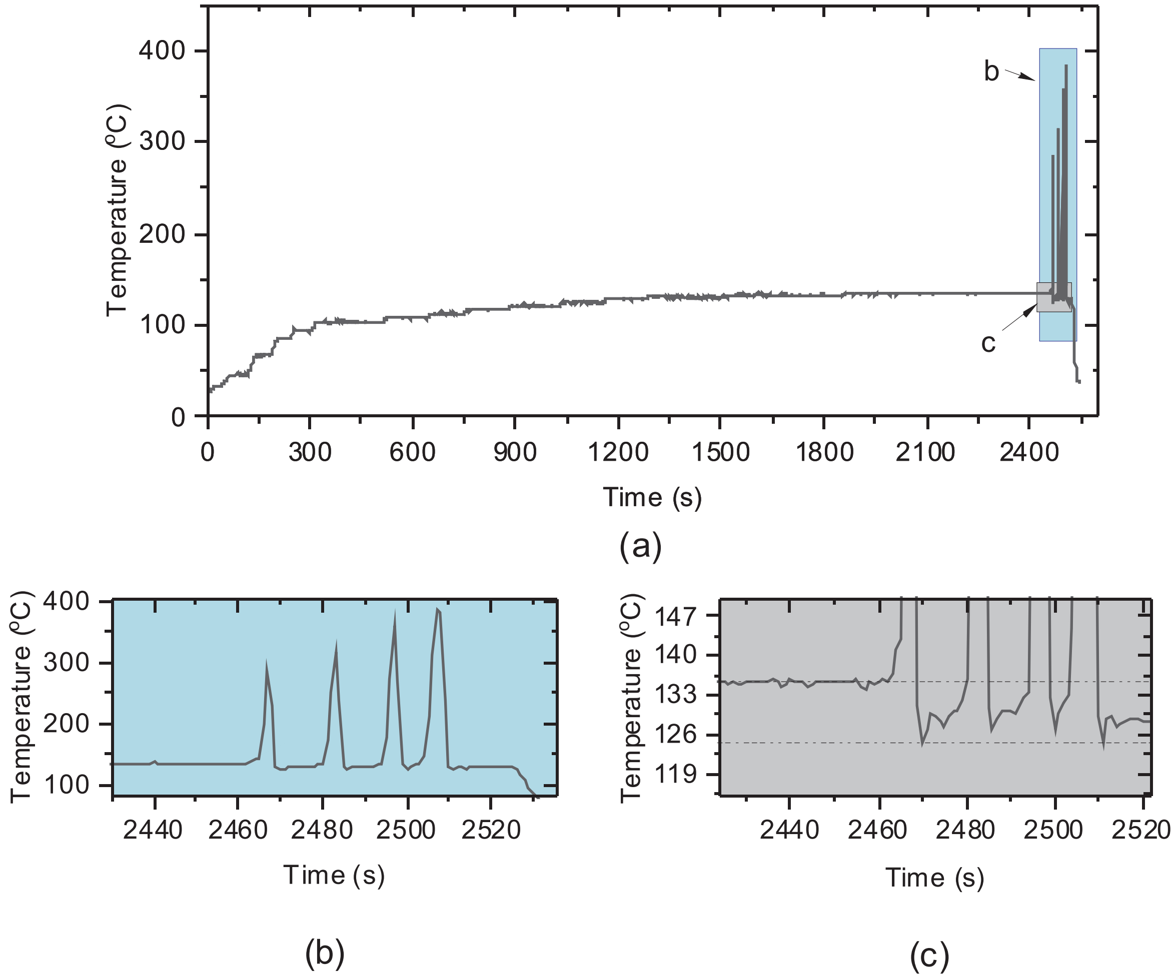

Fig. 2: (a) shows the CHF plot; (b) shows the zoomed-in portion of the same plot in (a) indicating the surface temperature peaks; (c)is a zoomed-in portion of the plot (a) showing the lowest temperature recordings at CHF.

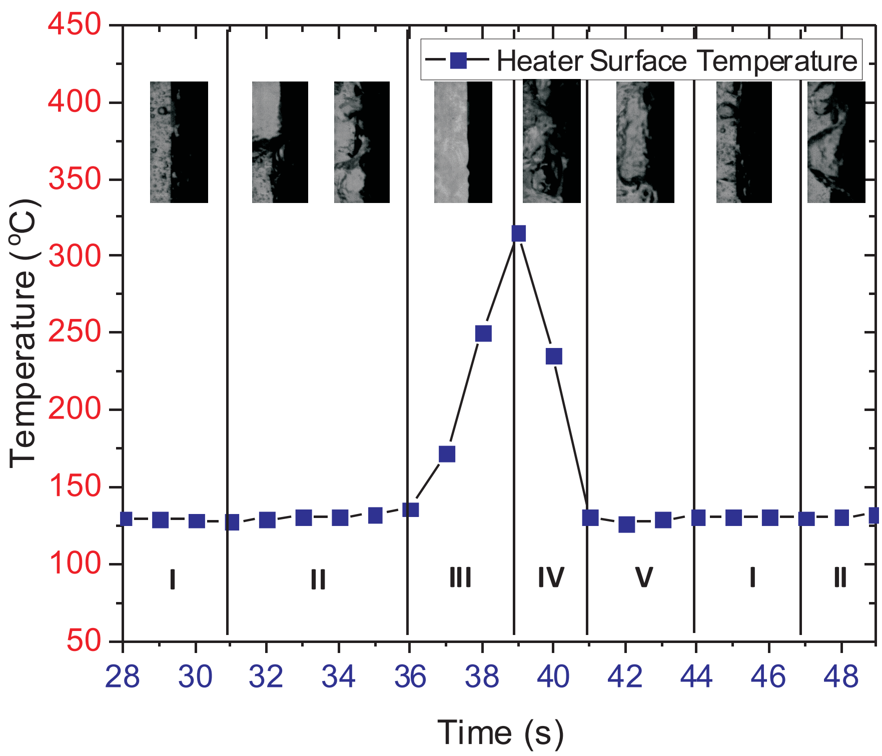

Fig. 3: Variation of heater surface temperature with time in one sonnet cycle

dance at CHF, we named this new CHF mechanism as 'Sonneting Critical Heat Flux'[10].

Sonneting CHF phenomenon is characterized by a unique cyclic flow behavior. Each sonnet cycle can be broadly classified into five zones (Figure 3). In Zone I and Zone II, the heater surface temperature was maintained despite the high heat flux. A sudden heater surface temperature rise was observed in Zone III due to a drastic decrease in heat transfer. An abrupt quenching followed this in Zone IV, with remarkable heat transfer. In Zone V, the heat transfer decreased from the peak value to almost the level of Zone I; nevertheless, in this region, the heater surface temperature was maintained.

Looking at the flow physics at CHF, in each sonnet cycle, initially, we observed bubbles around the heater surface moving upward alongside the liquid, like in classical bubbly flow (Zone I). Within a few seconds, the flow pattern changed to slug/churn flow as the bubbles coalesced to form large-sized bubbles (Zone II). Subsequently, it changed to an unusual reverse annular flow pattern (Zone-III) in which the vapor core moved upward with high velocity while the liquid film moved downward. As the water was not able to wet the heater surface, heater surface temperature increased. However, with time, the liquid film penetration (entrainment) into the vapor core increased due to interfacial shear as both the liquid and vapor are moving in opposite directions.

In addition to this, the coalescence of falling liquid ripples helped in the formation of big disturbance waves. The combined effect resulted in a chaotic flow pattern (Zone-IV). It led to a sudden quenching of the heater surface. The heat transfer was surprisingly very large, which resulted in a reduction in heater surface temperature to a smaller value compared to the value at pre-CHF conditions, e.g., during the fourth sonnet cycle. In fact, the heater surface temperature dropped from 385°C to 125°C, whereas the heater surface temperature at pre-CHF conditions was 135°C. Notably, during the quenching process, the total heat removed was 1.5 times higher than the supplied heat.

This unprecedented quenching process helped in bringing down the heater temperature. After the quenching, the flow pattern changed to slug/churn (Zone-V), following which bubbly flow was restored (Zone-I). This whole process repeated for a few cycles. With each sonnet cycle, the amplitude of heater surface temperature continuously increased and then fell to a lower temperature, which remained almost the same at about 125°C (Figure 2c). The periodicity continuously reduced, as seen in Figure 2. The entire transient of four sonnet cycles occurred in about 70 s. After which the power supply was tripped to save the test section against damage.

Furthermore, it is important to note that for the investigated range of mass flux conditions, the CHF value predicted by the look-up table is about 15% less than the experimental CHF[11].

Conclusion

CHF is generally characterized by an abrupt increase in heater surface temperature. However, the phenomenon is not clearly understood at low pressure and low flow conditions. Considering its importance in accidental conditions of water-cooled reactors and start-up conditions of natural circulation BWRs, we performed a series of experiments to understand the flow physics at CHF.

At CHF, we observed the heater surface temperatures to increase and decrease in a rhythmic fashion. We coined this unique CHF mechanism as 'Sonneting CHF'.

A unique cyclic flow behavior characterizes sonneting CHF phenomenon. In each sonnet cycle, several flow pattern transitions were observed in a span of few seconds. The entire transient at CHF, which involved four sonnet cycles, took place in about 70 s.

Notably, during an unusual reverse annular flow pattern, heater surface temperature was observed to increase. And it was followed by a sudden quenching of the heater surface even when the temperatures were close to 400°C.

During the quenching process, the heat removal rate was about 1.5 times higher than the steady-state heat removal rates, which resulted in a reduction in heater surface temperature to a smaller value compared to the value at pre-CHF conditions.

The CHF look-up table under predicted the experimental CHF value by about 15%.

Sonneting CHF phenomenon has brought new insights to the boiling systems, especially at low flow conditions. It would enable further insights into innovative boiling systems. Moreover, it is important to note that the journey of understanding the limits of boiling is far from over and further efforts are required to unravel its mysteries

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Development of improved fatigue design procedure for nuclear/non-nuclear materials subjected to simple/complex cyclic loads

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Abstract

The piping/ vessel components of Indian Nuclear Power Plants (NPPs) are subjected to complex multiaxial cyclic conditions. These components are designed for envisaged cyclic conditions using standard codes. The current design procedures under complex multiaxial cyclic loading conditions do not adequately account for fatigue damage. Therefore, such design procedures result in inaccurate fatigue life assessments. In this regard, extensive fatigue tests have been conducted to determine the extent of fatigue damage under complex multiaxial conditions vis-à-vis simple uniaxial cycling. A new procedure has been developed which predicts fatigue crack initiation life and crack orientation plane reasonably well. The developed procedure has been validated for various materials used for nuclear and non-nuclear applications.

Keywords: Fatigue crack initiation life, multiaxial, cyclic plasticity, critical plane models, crack initiation plane

Introduction

Almost all structural components used for nuclear and non-nuclear applications are subjected to cyclic loads during their design life. The fatigue failures may occur in material even if it is subjected to load amplitudes lesser than the yield strength of material. Therefore, the conventional fatigue design procedure differs from that under static (or non-cyclic) load conditions.

A very large factor of safety, typically 20 on number of cycles or 2 on stress amplitude, whichever is conservative, is adopted in fatigue design codes unlike 1.5 on yield strength or 3 on ultimate tensile strength, for the design against static loads. Despite this large factor of safety for cyclic loading, various fatigue related failures have been cited by International Atomic Energy Agency (IAEA) for various components of Light Water Reactors (LWRs) ([1],[2]). This indicates that either the conventional design procedure does not adequately quantify the fatigue damage or even a large safety factor of 20 on number of cycles is inadequate.

This mismatch in realistic fatigue damage and that as quantified using design code, is primarily due to the use of material fatigue life curve determined under pure axial cyclic conditions for benign air environment. However, the real component of NPP is subjected to multiaxial state of cyclic stresses/ strains under comparatively harsh coolant environment. These major key points of multiaxial cyclic stress state and synergistic damage under corrosion-fatigue are not accounted in the present design explicitly.

To overcome this shortcoming, it is required to understand the fatigue damage under complex multiaxial stress state vis-à-vis pure axial conditions and quantify the variable safety factors (in-spite of fixed factor of 20) for various operating coolant conditions w.r.t. air at room temperature. For this purpose, the present study investigated the first key reason of stress multiaxiality. Extensive test investigations have been performed on primary piping material of Pressurized Heavy Water Reactor (PHWR) under simple uniaxial and complex multiaxial cyclic conditions. The actual fatigue life under such conditions has been determined and compared with that predicted using present fatigue design procedure. The popular critical plane models, as available in literature, have also been explored. The predicted fatigue life using existing code procedure/ popular critical plane models is found to be higher than that observed in many multiaxial fatigue tests.

A new simple-to-use critical plane model has been developed which results in accurate fatigue life assessments for PHWR piping material. The validity of the developed model has been confirmed w.r.t. 17 different ferrous/ non-ferrous alloys used for various engineering applications and subjected to wide variety of simple/ complex multiaxial cyclic conditions.

Uniaxial and multiaxial fatigue tests: Determination of fatigue life

Extensive uniaxial and multiaxial fatigue test data have been generated on Primary Heat Transport (PHT) piping material (low carbon steel) of PHWR ([3], [4]). The duration of each fatigue test typically varies from few days to several weeks depending on the applied loading conditions. Typical multiaxial test set up

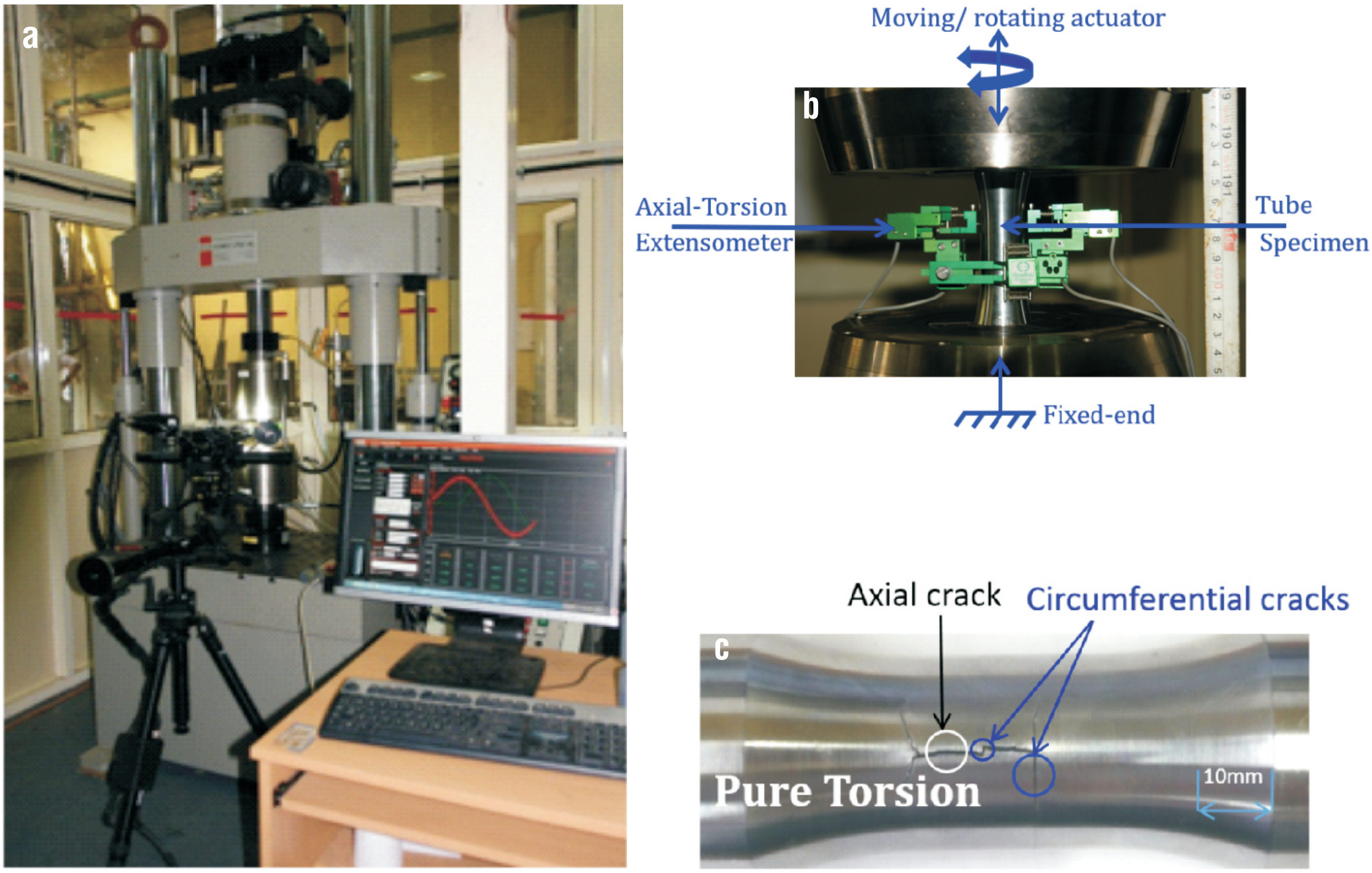


Fig. 1: (a) Multiaxial fatigue testing machine, (b) Tube sample mounted with biaxial extensometer and (c) Cracked tube after fatigue test

showing a tube (representative material of PHWR piping component) along with necessary instrumentation, has been shown in Fig. 1.

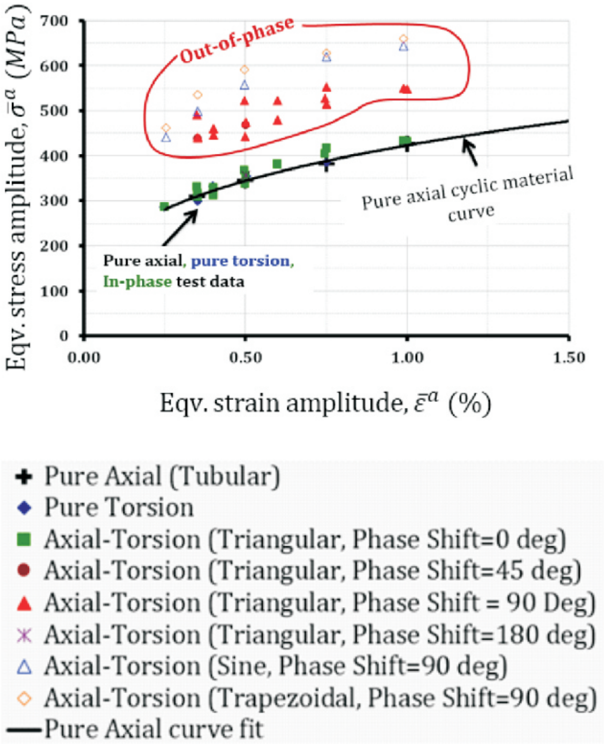
A wide variety of multiaxial loading conditions simulating different service transients, have been investigated. These multiaxial loadings can be in-phase or out-of-phase with each other. Test observations have brought out that carbon steel material shows significantly higher fatigue damage under out-of-phase loading conditions than corresponding in-phase scenario. Therefore, the observed fatigue life under out-of-phase conditions is significantly shorter than the corresponding in-phase condition. Fig. 2 indicates the higher fatigue damage in terms of higher equivalent stress amplitude (material response) for out-of-phase multiaxial conditions than in-phase condition for a given equivalent strain amplitude (measure of controlled variables) . The equivalent stress amplitude under in-phase condition has been observed comparable to corresponding pure axial and pure torsion conditions (Fig. 2).

Standard code design procedure for fatigue life assessment

The section III of American Society of Mechanical Engineers (ASME),Boiler and Pressure Vessel (B&PV) code is referred for the design of pressure vessels/ piping components[5] of nuclear facilities. This code is based on maximum shear stress/ maximum distortion energy failure criteria. These criteria have been adopted for evaluation of fatigue damage along with stress analysis based on linear elastic material considerations. Though, there exists two different design procedures for in-phase and out-of-phase loading scenarios in code, yet, these procedures do not account for additional damage taking place under out-of-phase multiaxial conditions, as observed in test studies. Due to this reason, the code procedure results in over-estimation of fatigue life for out-of-phase conditions. The comparisons between fatigue damage (in terms of alternating stress intensity amplitude, Salt) and test fatigue life (Ni) for pure axial, pure torsion, in-phase axial-torsion and out-of-phase axial-torsion are shown in Fig. 3.This figure shows that best fit curve as determined under pure axial conditions on PHT piping material of PHWR is close to ASME median fit curve for carbon steel material. A factor of two on median fatigue life is a well-established band to account for material intrinsic variability. This band has been considered for assessing the goodness of different fatigue life assessment procedures. Fig. 3 shows that fatigue damage assessed(in terms of Salt) using ASME procedure results in over-estimation of fatigue life mostly for out-of-phase conditions. However, pure axial, pure torsion and in-phase loading cases fall within the material intrinsic data scatter band.

A new critical plane methodology: Developed for improved fatigue assessments

A new methodology [6] has been developed for accurate assessment of fatigue life of nuclear structural components. In this methodology, elastic-plastic finite element analysis is first performed for simulation of multiaxial stress-strain behavior. The accurate



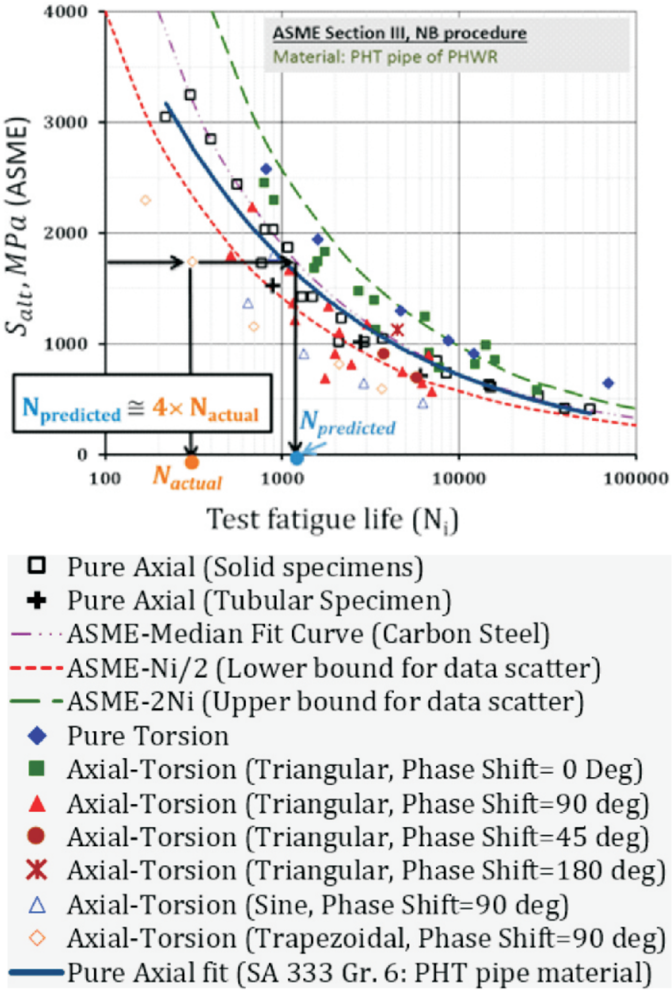
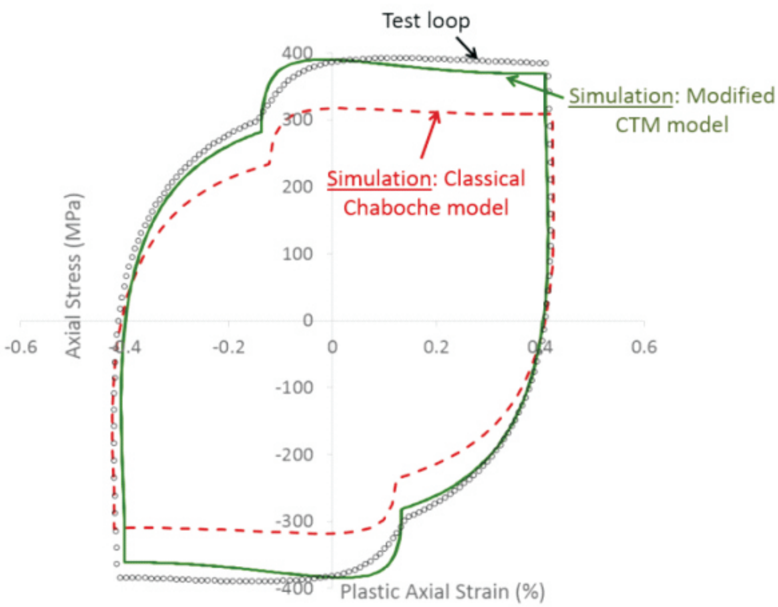


Fig. 2 (above): Cyclic stress-strain material diagram showing pure axial, pure torsion, in-phase axial-torsion and out-of-phase axial-torsion test data points.

Fig. 3 (right): Alternating stress intensity amplitude (Salt, code measure of fatigue damage) versus test fatigue life curve for PHT piping of PHWR.

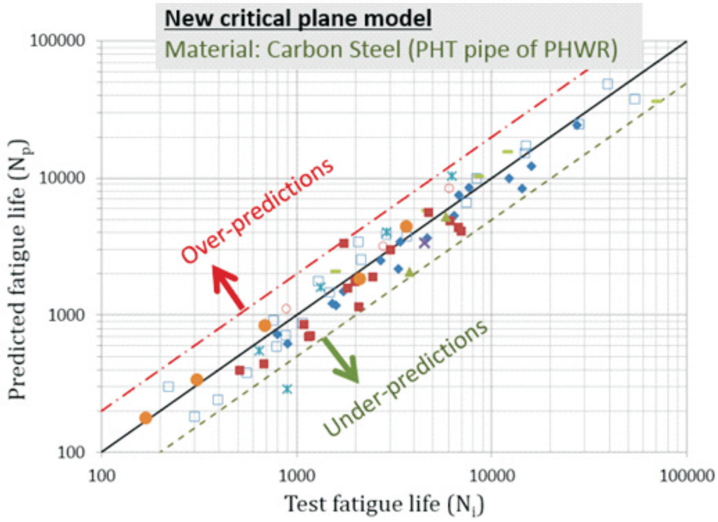
Fig. 4 (below): Typical comparison between test and simulated axial hysteresis loop (axial stress-versus-plastic axial strain).



information of simulated stress-strain response is used as input for fatigue life assessment model. Therefore, the information of higher hardening under out-of-phase conditions is used for fatigue life assessments. Due to this reason, critical plane models are expected to produce improved assessments. Although, there exists several critical plane models ([7]-[10]) in the open literature, however, these models have been validated for limited class/ grades of materials. Also, nearly all these models ([8]-[10]) are associated with the subjectivity in calculations of resultant shear component on an oblique material plane. The newly developed model has eliminated this subjectivity and is simple-to-use[6]. Further, the new model uses a material parameter (k) in fatigue damage to quantify the relative extent of shear (or normal) strain energy w.r.t. total strain energy. Therefore, model with such material parameter is expected to produce reasonably accurate fatigue life assessments for materials failing in both shear mode (highly ductile materials) and normal energy mode (relatively high strength and less ductile materials).

a. Modeling of cyclic stress-strain response

The first step towards fatigue life assessment is to simulate cyclic stress-strain behavior of material in the form of cyclic hysteresis loops accurately. For this purpose, commercial Finite Element (FE) softwares are available with various classical cyclic plasticity material models. These models perform reasonably well for uniaxial/ in-phase loading scenarios. However, the stress response is significantly under-estimated for out-of-



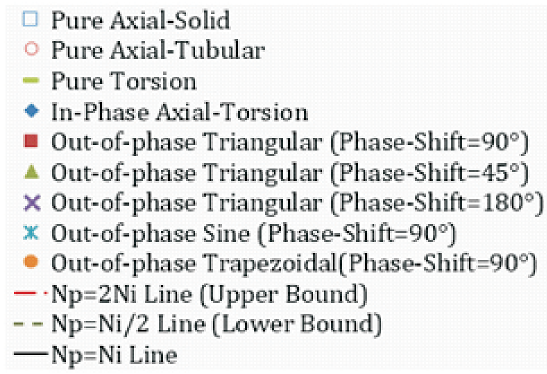
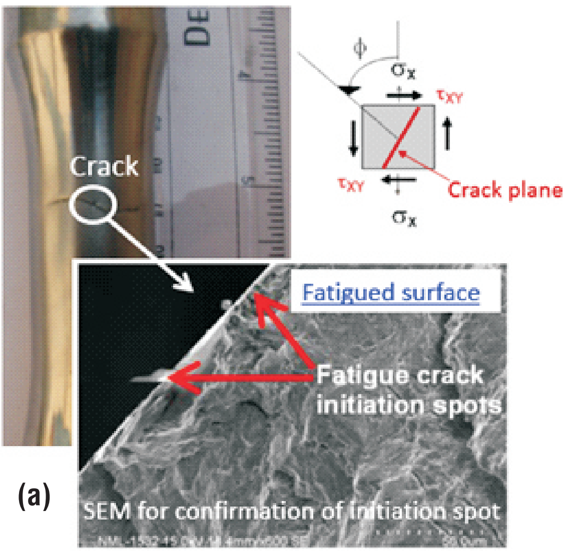


Fig. 5: Comparison of predicted (using new critical plane model) and test fatigue life for simple uniaxial and complex multiaxial load cycling for the tests carried out on PHT piping material of PHWR.



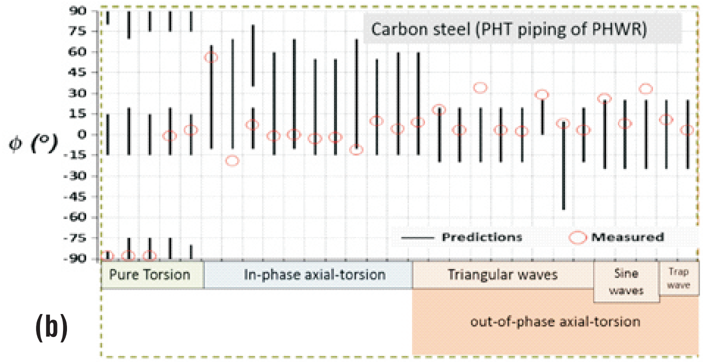


Fig. 6. (a) Fatigue tested specimen showing crack on outer surface of tube and SEM image to confirm crack initiation location, (b) comparison for predicted cracking plane range with measured crack angles

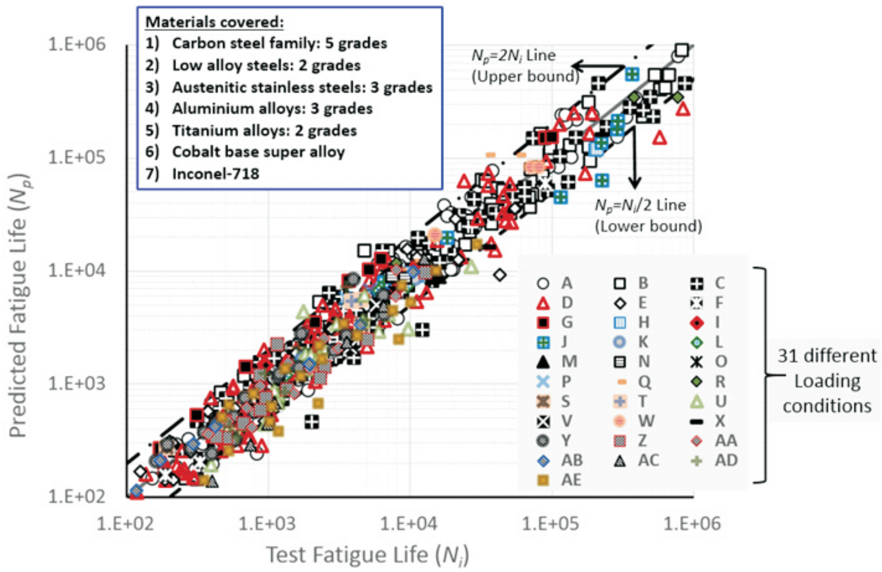
phase conditions [11]. Some of the advanced cyclic plasticity material models are available in literature ([12]-[14]). However, these models have been hardly included on commercial FE platforms. In this view, an in-house FE code has been developed for accurate simulation of material response under simple and complex cyclic conditions. This code is based on incremental plasticity theory at continuum length scale and includes von-Mises yield criterion, Prandtl-Reuss flow rule, linear / non-linear kinematic hardening rules and isotropic hardening rule. The code has been benchmarked for various loading conditions and material considerations. This FE code includes classical cyclic plasticity material model of Chaboche and few advanced material models, such as Chaboche-Tanaka-Meggiolaro (CTM) and in-house developed modified-CTM [11]. The modified-CTM material model results in accurate simulation for uniaxial, in-phase and out-of-phase loading conditions. A typical comparison between test and simulated hysteresis loops is shown in Fig. 4 for out-of-phase condition using classical Chaboche (commercial FE software) and modified-CTM (in-house FE code) models.

b. Fatigue life assessments: Predicted-versus-test fatigue life

The simulated hysteresis response is input to newly developed critical plane model under various uniaxial and multiaxial conditions. Fig. 5 shows the comparison between predicted and test fatigue life for various tests conducted under pure axial, pure torsion, in-phase axial-torsion and out-of-phase axial-torsion conditions with various phase shift angles such as 45°, 90° and 180° with different loading waveforms (triangular, sine, trapezoidal). The comparison shows that predicted and test fatigue life under all such loading scenarios are in close agreement with each other and the test data are mostly contained in material intrinsic acceptable data scatter band of two.

c. Orientation of crack initiation plane: predictions-versus-measurements

Since the critical plane theory is associated with the material plane experiencing maximum fatigue damage, therefore, the crack plane orientations can also be predicted using critical plane model. The actual cracking angles have been measured using post image analyses of fatigued tube specimens [3]. The location of crack initiation spot has been confirmed using Scanning Electron Microscopy (SEM) as shown in Fig. 6 (a). The material planes experiencing fatigue damage higher than 90% of the maximum damage value are the probable cracking plane orientations. Fig. 6 (b) shows that new critical plane methodology also results



Legends:
pure axial: A , pure torsion: B
pure axial with positive/negative mean strain: I/J, torsion cycling with mean shear strain or tensile/ compressive axial mean: K,Q/R, completely reversible in-phase axial-torsion cycling: C/S, in-phase axial-torsion with mean axial/ shear strain: L-P, out of phase axial-torsion: D-H/T-W and asynchronous (different loading frequencies) axial-torsion: X-AE

Fig. 7. Comparisons between predicted fatigue life (using new critical plane model) and test fatigue life for 10 numbers of ferrous alloys and 7 numbers of non-ferrous alloys

in accurate predictions of cracking planes under nearly all loading scenarios.

Extensive validation of developed model for various ferrous/ non-ferrous alloys

Although a large number of critical plane models are available in literature, yet scarcely a model exists which has been validated w.r.t. wide variety of engineering materials and large sets of multiaxial test data. With this viewpoint, a large set of uniaxial/ multiaxial tests data on 17 different grades of ferrous/ non-ferrous alloys has been collected from literature.

The ferrous alloys, typically used as PHWR, LWR piping/ vessel materials, cover different grades of mild/ carbon steel, low alloy steels, austenitic stainless steel and non-ferrous alloys include various grades of aluminum alloys, titanium and its alloys, cobalt base super-alloy and nickel alloy.

The yield strength of these materials/ alloys varies from 191.5 MPa to 1160 MPa and ultimate tensile strength ranges from 229 MPa to 1420 MPa. Thirty one different uniaxial and complex multiaxial loading conditions have been considered for this validation exercise. The test fatigue life typically ranges between ~100 cycles and ~10^6 cycles covering low to high cycle fatigue regimes. Fig. 7 shows reasonably accurate comparisons between predicted fatigue life using new critical plane methodology and test fatigue life for various ferrous and non-ferrous alloys. This validation for fatigue life assessments on 17 numbers of widely different engineering materials subjected to 31 loading conditions with more than 800 fatigue life comparisons, further strengthens the applicability of the newly developed critical plane model ([6],[15]). Hence, this model can be used for realistic fatigue life assessment of wide range of metals (both ferrous and non-ferrous) under simple as well as complex multiaxial loading cases.

Conclusions

The uniaxial/ multiaxial test studies carried out on Primary Heat Transport (PHT) piping material of Indian PHWR and fatigue life assessments using current design code procedures vis-à-vis new methodology are summarized below,

- PHT piping material showed higher material hardening under out-of-phase axial-torsion conditions than corresponding in-phase and uniaxial cases. This higher hardening resulted in reduction of fatigue life for out-of-phase multiaxial conditions.
- The current ASME section III, NB procedure for fatigue life assessments results in over-prediction of fatigue life mostly for out-of-phase conditions.
- A new simple-to-use critical plane based fatigue life assessments model has been developed. This model predicts both fatigue crack initiation life and crack orientations accurately.
- The developed model has been extensively validated for various materials and loading conditions. This model may be used for improved fatigue assessment of mechanical components.

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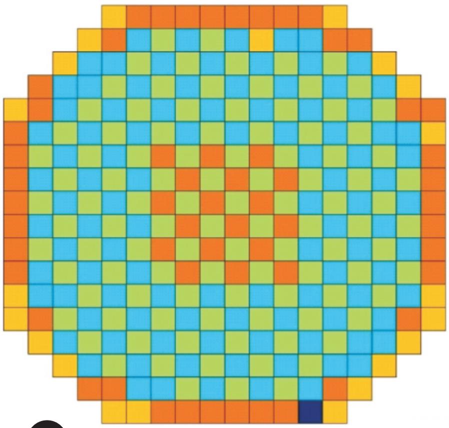
Core loading pattern optimization for TAPS Units # 1&2 for maximizing the energy output

Fuel management of boiling water reactors TAPS #1 and 2 at Tarapur is very challenging. These units have been successfully operated for the past 50 years with an intense collaboration between NPCIL and BARC. It is worth mentioning that BARC is the sole agency responsible for the optimization of core loading patterns and fuel management of these units.

The operational history is followed and the energy output of every operating cycle is maximised. The refueling frequency is about 18 months. A third of the core is replaced at each refueling cycle. The core loading pattern for each operating cycle is evolved after a detailed optimization study with respect to several operational parameters. The complete fuel management entails design of the new reload pattern for the next operating cycle, core follow-up for the current operating cycle, and monthly updates for control rod sequences. The core loading optimization is more challenging if leakers i.e. failed fuel assemblies are identified and have to be removed. The optimization is done based on the Haling principle. The underlying principle is based on the fact that the core power distribution is maintained constant over the cycle. The Haling power distribution is the flattest power distribution possible over the entire cycle. In BWR, power distribution is bottom peaked due to reduced water density at top.



1 **Core loading pattern for TAPS BWR unit-1,cycle-27**

At Beginning-of-Cycle (BOC), the power peak at the reactor bottom part can be controlled by control rods. The control rods positions are changed over the operating cycle to obtain a flat power distribution and hence maximum energy output for the cycle.

TAPS BWR Unit-2 Cycle-26 was made critical on 24/10/2018 and is capable of delivering energy 280,000 MWD (~6.0 GWD/ST). This cycle was operated upto June 2019 with the optimised parameters and it required control rod sequence change. Control rod patterns at mid cycle for sequence change were worked out and the shut down margin was estimated.

TAPS Unit#1 was operated till January 8, 2020 with the operating cycle-26 and it has subsequently undergone refueling shut down. The reload pattern for cycle-27 of the Unit # 1 was worked out using the End-Of-Cycle (EOC) exposure of cycle-26. Haling calculation, described above was done for cycle-27 to achieve maximum power throughout the cycle. One of the fuel assemblies had failed and was identified at EOC. The average core exposure of Unit-1 cycle-26 at EOC was ~15 GWD/ST (Burnup in TAPS BWR is estimated as Gigawatt-day /Short ton where 1 shot ton = 90 kg). The average discharge burnup at the unit-1 cycle-26 at EOC was ~20 GWD/ST with highest discharge burnup at ~26 GWD/ST. the core loading pattern worked out for cycle 27 for Unit#1 is shown in Figure 1. The core average exposure at Unit #1 cycle-27 at BOC is ~8 GWD/ST. The estimated value of cycle energy is 6.0 GWD/ST.

The TAPS BWR units have successfully operated with the inputs from BARC and are a fine example to continuous coordinated effort between the operators and reactor physics experts for safe day-to-day operation of the aging boiling water reactors.

This article was contributed by ***K.P. Singh, Rashmi Rai, Arun Kumar Singh*** and ***Umasankari Kannan***

Reactor Physics Design Division

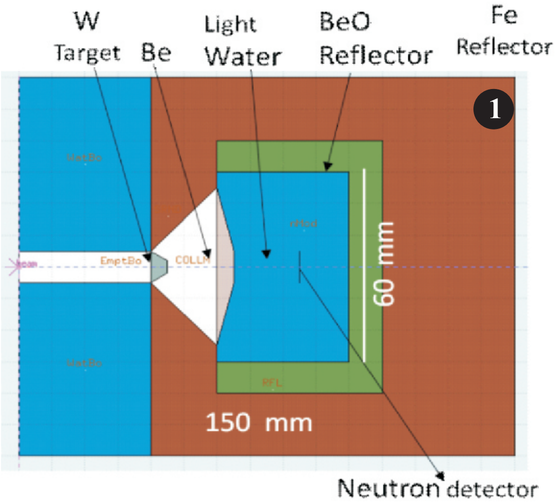
Production of Molybdenum-99 from linear accelerators

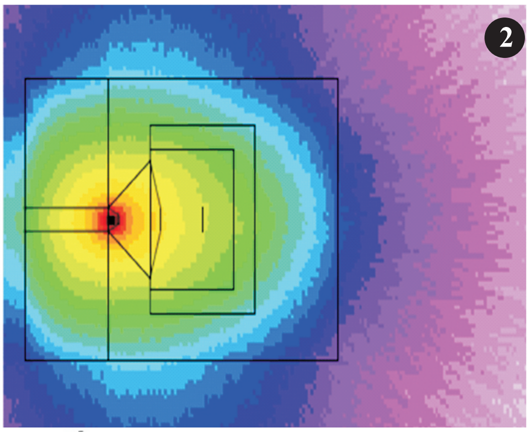
Tecnitium-99m (Tc-99m) is the decay product of Molybdenum-99 (Mo-99) and is widely used in nuclear medicine for diagnostic purposes. Mo-99 is primarily produced in research reactors using neutron induced reactions i.e. capture or fission. Alternate routes for producing Mo-99 are being explored worldwide. The neutrons from electron based linear accelerators (e-LINAC) can be another source to produce radioisotopes. A feasibility study for the production of Mo-99 with the electron LINAC facility at the Electron Beam centre (EBC), Kharghar was taken up. The aim will be to design a facility for producing low specific activity Mo-99 which is a precursor to Tc-99m which is extensively used as a radiotracer for diagnosis.

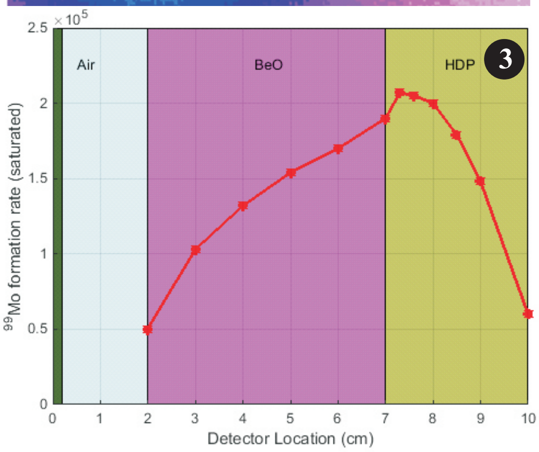
The basic principle of this alternate route is briefly outlined here. The beam of electrons emerging from electron accelerator is made to strike a suitable target like Tungsten or Tantallum. The Bremsstrahlung gammas thus produced are then allowed to fall on a Be target. Photoneutrons are then produced from Be target. These neutrons when captured by Mo-98 produce Mo-99. This reaction is basically (e-γ-η) cascade reaction. Radio-isotope tracers are characterized by their specific activity which is the amount of radiations emitted per second per unit mass of the substance. The maximum activity that can be induced in the sample during irradiation is termed as saturation activity and is an important parameter for such irradiations.

As a first step, it was planned to measure the effective cross section of the capture reaction of Mo-98 in the e-LINAC set-up. Theoretical studies were done to design an experimental set-up for the same. Simulations of photoneutron source from 10 MeV electron beam striking a Tungsten (W) or Tantalum(Ta) target were performed. A set-up was designed and modeled using Monte Carlo particle transport code, FLUKA is shown in Figure 1. The neutron flux can be enhanced by optimizing the moderating medium surrounding the target. The design studies included optimization of the moderator with Be, Water, Carbon and Polyethylene. With the use of Be as target assembly surrounded with water, the saturation activity was found to be maximum. The effective cross section for production of Mo-99 was calculated to be 0.23 barns where the thermal neutron flux was nearly 1.0E+7 n/cm^2/s. The neutron flux profile obtained is shown in Figure 2. The specific activity for water moderated configuration was estimated to be 39 µci/g which can be enhanced to 250 µci/g by a suitable conical geometry of the target a shown in Figure1.

In order to perform the measurements for the effective cross section of 98Mo(n,γ) at EBC, Kharghar, the experimental set-up was re-optimized with Be target and High Density Polyethylene (HDP) moderator. Pre experimental analysis for measuring the 98Mo(n,γ) 99Mo cross section using 10 MeV LINAC was performed with Monte Carlo particle transport code, FLUKA and in-house code PATMOC.







1. Experimental set-up designed using e-LINAC for Mo-99 production

2. Neutron flux profile in experimental set-up with Be as target and light water as moderator

3. Saturated rate of formation of Mo-99 at different locations of the set-up

The experiment was performed on 7th October 2020 at EBC Kharghar. The electron accelerator was operated for 4 hours at 0.24 mA current. Natural Molybdenum metal foil along with other thermal and resonance flux monitors were placed on the emergent surface of the HDP block in the setup. The activation foils used were thermal flux monitors i.e. Gold, Copper, Manganese, Indium and Cobalt and resonance (epithermal) flux monitors namely, Scandium, Silver, Lutetium, Tantalum and Tungsten. Estimate of the effective cross section from the analysis of experimental data from different activation foils and methods range from 0.28 barns to 0.4 barns, which is reasonably higher than the 0.13 barns in a reactor environment. The spatial variation of saturation reaction rate for formation of 99Mo estimated from FLUKA for the experimental set-up is shown in Figure 3.

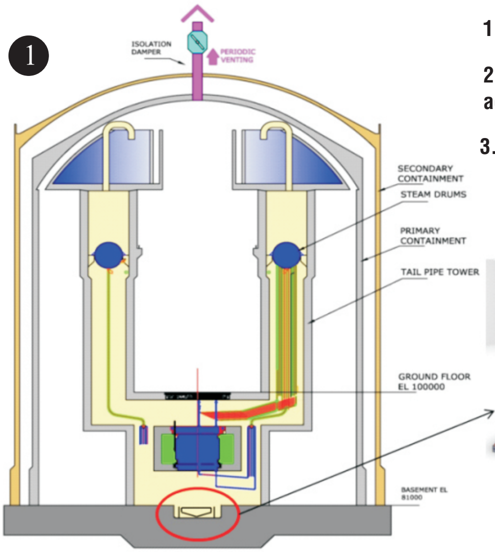
This article was contributed by ***Umasankari Kannan, Kapil Dev, Rajiv Kumar, Amod Kishore Mallick, Sudipta Samanta, Deep Bhandari***

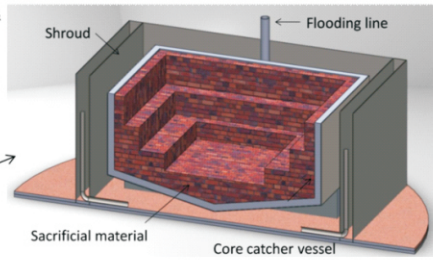
Reactor Physics Design Division

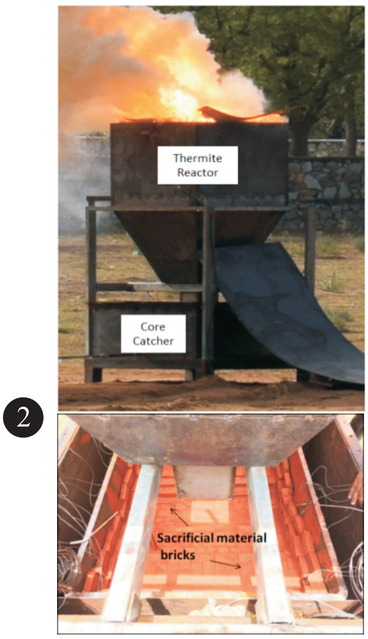
Technology Development of Core Catcher for Indian Advanced Nuclear Reactors

Post Fukushima, to manage low probability core melt accidents, dedicated core catchers are being developed worldwide. Since these technologies are proprietary, for Indian AHWR and light water cooled reactors, technology for a unique core catcher has been developed to contain and cool the core melt for extended period and reduce the radioactivity release to public domain substantially. This core catcher can be deployed in other advanced light water reactors also. Cooling of more than 100 tons of this mixture of nuclear fuels, structural material, control rod materials, having temperature more than 2800 °C and generating decay heat continuously, to very low temperatures for prolonged period is a technologically challenging and scientifically complex task.

To achieve this, a unique core catcher has been developed by BARC which consists of inverted pyramid shaped thick steel vessel as shown in Figure 1. Special sacrificial material was developed in house, which absorbs the heat of this corium by melting and mixing in it. When corium mixes with this sacrificial material, it becomes lighter and moves to the top and the metallic components sink at bottom which is termed as density inversion. This has two advantages: (i) there is no metal at top thereby eliminating chance of hydrogen generation by metal water reaction and (ii) the light weight ceramic melt forms a stable crust enveloping the heat generating high temperature melt like a “capsule” so that when water is added to



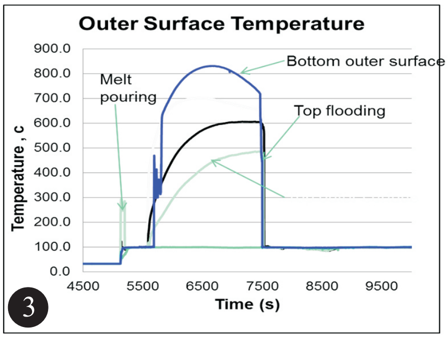




1.AHWR Core catcher

2.Design validation at prototypic condition using actual sacrificial material

3.Vessel surface temperatures



the top of melt to cool it, the stable crust prevents water seeping into the bottom of core catcher and avoids metal water interaction. The core catcher contains and cools corium for prolonged period. The core catcher has design life same as that of reactors.

The core catcher geometry and flooding strategy was optimized by conducting several experiments in scaled facilities [1-3]. In addition, integral experiments were conducted at prototypic condition and using actual sacrificial material in which about 550 kg simulant melt at more than 2500 °C [4] was poured in the core catcher and was cooled with water as per the actual flooding strategy (Figure 2). It was observed that, the inner vessel temperature remains always below 900 °C and the outer vessel temperature never exceeded water saturation temperature when water is present outside (Figure 3). A stable ceramic crust was obtained at the top. The experiment was repeated with 300 kg melt which showed good repeatability by obtaining similar results. The decay heat removal capability was also demonstrated in integral experiment which demonstrated stable crust formation and decay heat removal for prolonged period.

Together, all these tests demonstrated the efficacy of core catcher for cooling and stabilization of molten corium for prolonged period in case of severe accidents.

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This article was contributed by
P. P. Kulkarni, A. K. Nayak, S. K. Sinha
Reactor Engineering Division

Integrated Test Station for C&I Systems of AHWR

Control and instrumentation (C&I) systems play crucial role in safe, secure and reliable operation of Nuclear Power Plants (NPPs). Owing to their pivotal role, high demands are always placed on functional, performance and qualification requirements of such systems. This calls for critical testing and validation of each and every component of the C&I systems prior to their deployment in NPPs. Traditionally, these systems and components are tested in stand-alone mode, with limited simulated interface with other relevant systems. To stretch the testing beyond this, it was envisaged to design a facility that supports integrated testing of all the important C&I systems along with a plant simulator.

In this context, a facility named AHWR Integrated Test station (ITS) has been designed, developed & commissioned at Reactor Control Division, BARC. It is a configurable test bed for integrated testing

C&I systems of ITS

and validation of functional, performance and qualification needs of proposed C&I systems and architectures of NPPs. This first of its kind facility is configured with all the major control and protection systems and other components of the C&I architecture for Advanced Heavy Water Reactor (AHWR)[1]. It comprises of prototypes of C&I systems built using the in-house developed Trombay Programmable Logic Controller (TPLC), network switches, data servers, display workstations, test panels and a real time engineering simulator. These interconnected components form the test bed.

Prototypes of safety critical systems including Reactor Protection System with Test and Monitoring System, Containment Isolation System, and safety related systems including Reactor Regulating System, Reactor Process Control System, Core Monitoring System, Alarm Annunciation System etc are part of the ITS. Operator workstations for this facility, developed using an indigenous Linux based SCADA, are provided in a centralized control room. The safety related system prototypes are interfaced with the real time engineering simulator facilitating close loop integrated testing.

By virtue of the mathematical models of all the major systems and components including reactor kinetics, thermal hydraulics and process systems, the engineering simulator [2] allows creating realistic scenarios of normal and off-normal operating conditions of the plant, thus making it possible to test the C&I systems in a realistic, virtual reactor environment.

C&I Security is an important aspect which needs to be considered right from the conceptual design stage of C&I architecture. An indigenous trusted computing module named ANU-NISHTA has been deployed in servers and workstations to overcome the security issues posed by commercial Operating Systems used therein. Likewise, an in-house developed secure Network Time Protocol time server module has been used in this facility for time synchronization of C&I systems.

Engineering simulator of AHWR

Various components of the C&I architecture of AHWR are tested and validated in this facility. The ITS is being extensively used in verification and validation (V&V) of plant control algorithms and safety interlocks, optimization of control gains of reactor and process control systems, optimization of detector locations in the reactor, design of control room interfaces etc. ITS is also used as a platform to carry out dynamic safety analysis of AHWR.

Further Reading:

[1] "Preliminary Safety Analysis Report of AHWR", Report no. AHWR/PSAR/USI-**01591**

[2] "Engineering Simulator for Advanced Heavy Water Reactor", BARC Newsletter, pp. 1-5, **May-June** 2018

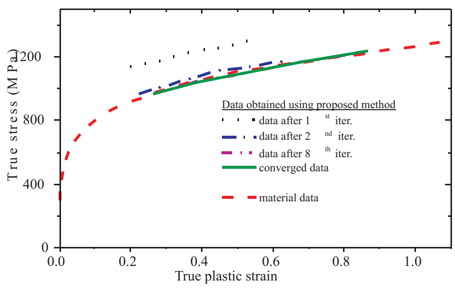
This article was contributed by
S.R. Shimjith, D.A. Roy and Uday Vaidya

Reactor Control Division

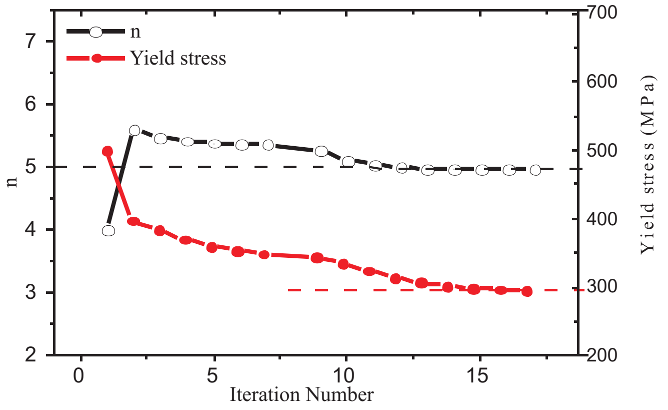
Evaluation of material stress-strain curve using load-indentation data from ball-indentation tests

For evaluation of mechanical properties such as yield stress, ultimate tensile strength and strain hardening exponent of service exposed materials, ball-indentation technique is handy as it can be used in the field using a suitable portable device. However, the expressions used in literature to evaluate true stress and strain contain empirical constants which are not valid for multiaxial nature of the indentation process. Moreover, these constants are dependent upon the type of material, microstructure etc. which are difficult to estimate a-priori. In this work, these limitations are removed through use of a novel method where multiaxial nature of stress and strain are incorporated in the formulation itself.

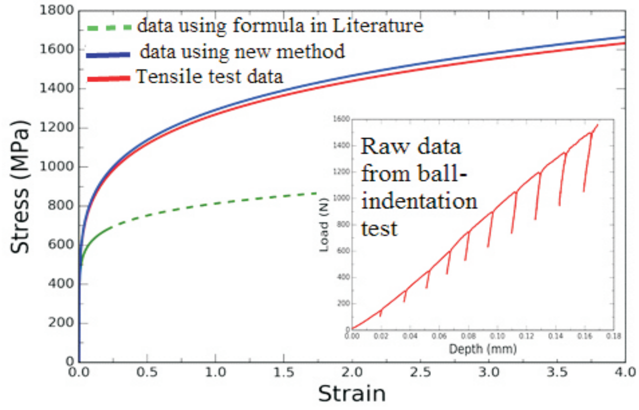
A new set of stress and strain multiaxial parameters have been evaluated from 3D finite element analysis and these have been expressed as functions of load, yield stress and strain hardening exponent (n). To evaluate the true stress-strain curve, the raw data in terms of load-indentation curve from experiment is taken. Using an initial guess value for yield stress and 'n', the multiaxial stress and strain parameters are evaluated. These data are used further to evaluate stress-strain curve. This algorithm is invoked in an iterative manner by updating the multiaxial parameter and the stress-strain curve is updated after each iteration as shown in Fig. 1. The stress-strain curve, yield stress as well as the hardening exponent 'n' converge quickly as shown in Fig. 1 and 2 respectively.



1. Stress-strain curve obtained from raw data



2. Convergence characteristics of new method



3. Comparison of stress-strain data for SA516Gr.70 steel among tensile, new, old methods

This method is valid for a wide range of strain data unlike the method in literature where strain is limited to 20%. This method doesn't use any empirical constant and the expressions for true stress and true strain are valid for multiaxial state of deformation as observed in ball-indentation tests. Moreover, no material-specific constants are required in the algorithm.

The algorithm has been applied to evaluate stress-strain curve of SA516Gr.70 steel. The load vs. depth of indentation data as obtained from experiment is shown in Fig. 3. The stress-strain curve as obtained using the method in literature and the current technique is shown in Fig. 3. The results of new method are vastly superior when compared to method currently used in literature and it is also close to tensile test data. Hence, it can be concluded that this new technique is more accurate and versatile as the multiaxial effect of state of stress in the indentation region of ball is taken care of inherently in the formulation.

This article was contributed by ***M.K. Samal, A. Syed & J. Chattopadhyay,*** Reactor Safety Division

Nuclear system for PHWR Start-Up after En-Mass Coolant Channel Replacement

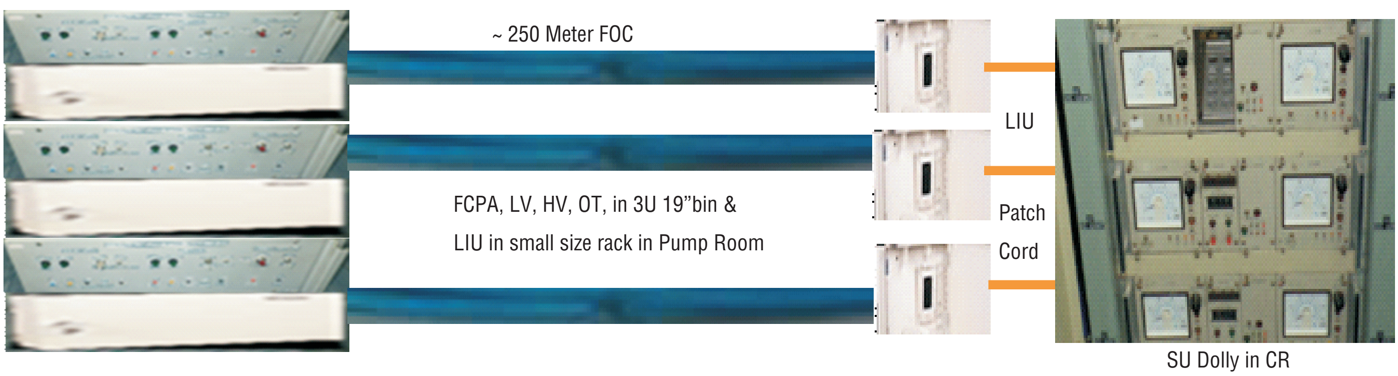
Indian PHWRs conventionally use ^{10}B lined Proportional Counters (BLPC) based in-core start-up channels during Initial Fuel Loading (IFL) and First Approach to Criticality (FAC).

In fresh core, in-core low gamma dose rate allows to measure neutron flux during IFL & FAC using BLPC with Pre-Amp located close to detector & pulse processing channel at around 50-meter distance, interfaced to Start-Up (SU) dolly in Control Room (CR) at around 200-meter distance. SU dolly houses triplicate Scalar / Timer, analog electronics, trip module, display & meters.

During start-up after EMCCR (En-Mass Coolant Channel Replacement), in-core gamma dose rate is high which precludes use of BLPC due to its poor gamma tolerance & count rate limitation of the detector & instrumentation system. Hence it requires specialized nuclear system based on Fission Counter (FC: IEC 60515 & ORNL reports) having higher gamma tolerance & higher count rate capability.

Nuclear System comprising Fast Current Pulse Amplifier (FCPA) with Optical Transmitter (OT) in 19", 3U bin & Optical Receiver (OR) was developed as shown in Figure 1. Triplicate bins & Light-guide Interface Unit (LIU) are located in Reactor Building (RB). FCPA amplifies FC signal, discriminates against noise and gamma pulses & drives OT which launches optical pulses on three sets of Armoured Fibre Optic Cable (FOC) that interface the pulse outputs from bins to SU dolly in CR through LIU and Optical Receiver (OR).

The system noise performance is enhanced by adequate power line & EMI-EMC filters, low noise LV, HV supply. The optical pulse transmission scheme makes the system immune to field noise,



1. Triplicate Startup channel of KAPS-1 used after EMCCR

maintains pulse characteristics & results in accurate high count rate capability up to 100 Kcps.

The system, first of its own kind in Indian NPPs, using optical pulse transmission scheme was used for KAPS-1 start up after EMCCR. The system with about 5% accuracy, linearity and acceptable statistics obtained though settable counting time performed satisfactorily for low neutron flux in high gamma background under challenging reactor noisy environment.

The system recorded counts during all phases of reactor start-up & up to criticality, matching the theoretically estimated counts within 10%. The system was also employed for execution of subsequent physics experiments.

The system can be used for FAC & start up after long shut down for any Nuclear Power Plant.

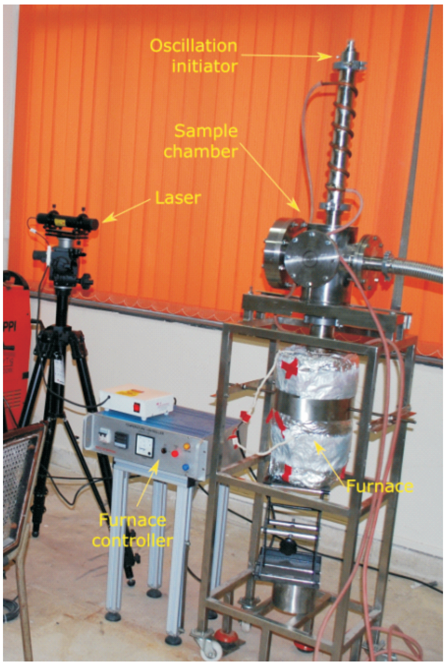
This article was contributed by ***Dr. P.V. Bhatnagar***, Electronics Division

Indigenous instrument development for molten fluoride salts

Molten fluoride based salts are to be utilised in both the primary and the secondary circuits of Indian Molten Salt Breeder Reactor (IMSBR) and Innovative High Temperature Reactor (IHTR). Instruments for measurement of molten salt density, impurities and its purification are not available commercially. High temperature viscometers need to be imported and the molten salt sample is susceptible to contamination from the cover gas. Further, special precautions need to be taken while handling salts containing uranium and thorium. In view of this, these instruments have been developed in-house.

Densitometer and viscometer for molten salts

The densitometer is based on Archimedes principle with compensation for thermal expansion. The viscometer is based on the oscillating cup technique in which the sample is sealed inside a capsule and the viscosity is determined by the change in damping characteristics of the filled capsule vis-à-vis the empty capsule. As the sample is sealed inside the capsule, it is not exposed to the cover gas and can be used for low level active salts.



Oscillating cup viscometer

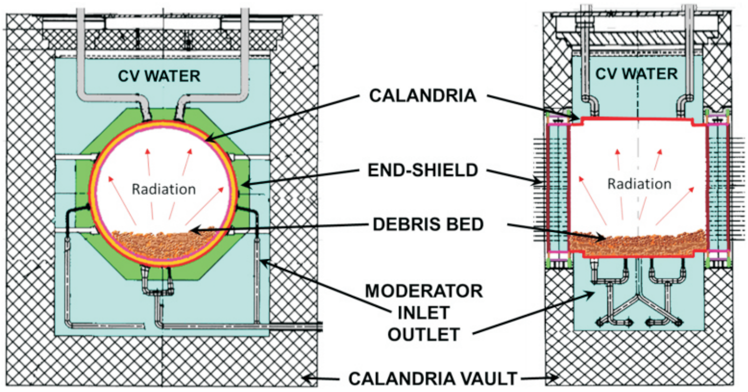
Impurity measurement and purification systems for molten salts

The electrochemical measurement technique has been developed for online measurement of impurities at high temperatures which can be utilised in a high radiation environment. In order to remove impurities from the salts, an electrolysis based purification method has been developed. These facilities were utilised to find out impurities and to purify FLiNaK, the secondary side coolant salt for both IMSBR and IHTR.

This article was contributed by ***A. Basak***, High Temperature Reactor Section

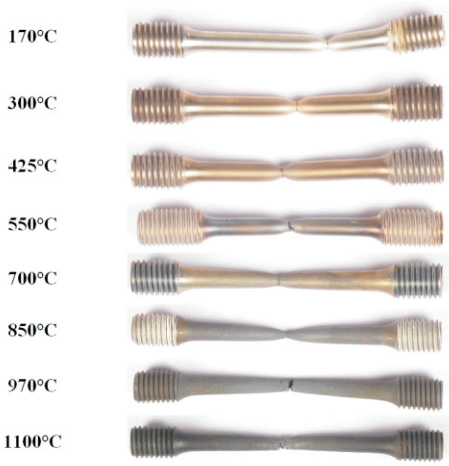
Time Estimation for In-Calandria Retention of Corium under Total Loss of Heat Sink

Severe core damage accidents in Indian PHWRs are rarest of the rare scenario that involve cascading failure of multiple systems provided under defence-in-depth design philosophy. Severe accident analysis requires postulation of accident scenario that involves multi-physics aspects such as reactor physics, thermal-hydraulics, transport behaviour, structural degradation etc. Calandria vessel of PHWR acts as a physical barrier in limiting the progression of such accident. It is submerged in a large pool of water in Calandria vault and is expected to contain and cool the core debris/ corium for reasonable duration, delaying/ preventing the adverse event of molten corium-concrete interaction. Thus, the in-Calandria retention of core debris/ corium for a larger time frame is of great importance to operator and regulator for planning of accident management action.



1. Calandria located inside vault under core-collapse conditions

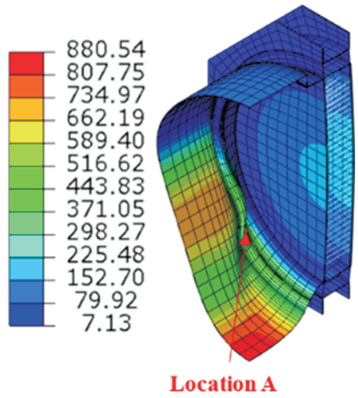
In view of above, structural integrity of Calandria has been assessed for in-vessel retention of core debris/ molten corium for a postulated accident scenario of unmitigated total loss of heat sink (no active means are available to remove the decay/ stored heat) for standard 220 MWe and 540 MWe Indian PHWRs. In absence of any management action, the fuel channels would degrade gradually and collapse in the form of debris on to the Calandria bottom (Fig. 1). Here onwards the Calandria is responsible for cooling and retention of core debris/ corium which has been analysed in detail using sequential thermal-hydraulic and structural analyses. Thermal-hydraulic analysis considered all modes of heat transfer–conduction, convection and radiation; two phase heat transfer; decay heat; heat from metal–water reaction; melting and ablation; and was conducted using the ASTEC code with suitable PHWR specific adaptations. Structural response was evaluated using finite element analysis where geometric and material non-linearities; temperature dependent material properties and deformation models were considered. Here the domain consisting of complete Calandria-end shield assembly was modelled to account for the boundary conditions and flexibility provided by structural elements such as annular plates and diaphragms. High temperature material data of Calandria material up to



2. Failed tensile specimens of 304L at different temperatures

1100°C were obtained from systematic tensile and creep tests programme (Fig. 2). Structural failure time of Calandria was assessed considering various likely failure modes that are plastic instability, excessive inelastic strains and creep-stress rupture criteria.

The study revealed that in 540 MWe PHWR, Calandria remained fully submerged till ~ 36 h after core collapse. The debris bed didn't undergo melting till 74 h and Calandria failure is observed roughly 3 days after the core collapse (Fig. 3). In standard 220 MWe PHWR, Calandria failure is observed at about 83 h after the core collapse. It demonstrated that the Calandria, aided by inherent design



3. Displacement (mm) in 540 MWe PHWR Calandria at 74.1 hrs from core collapse

features of PHWR viz., several heat sinks and low power density, delays the accident progression extensively and provide a large time frame available to the operators for appropriate SAMG actions. The details of study are available in BARC external reports (BARC/2017/E/005, BARC/2019/E/008, BARC/2019/E/011, BARC/2020/E/015 and journal papers (NED, Vol.367, Article ID – 110791, NED, Vol.368, Article ID-110801).

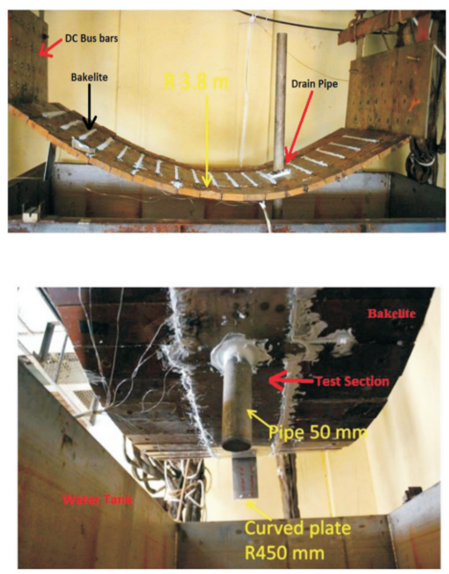
This article was contributed by ***Keshav Mohta, Onkar S. Gokhale, Suneel K. Gupta, Deb Mukhopadhyay, J. Chattopadhyay***

Reactor Safety Division

Demonstration of Critical Heat Flux Limits in calandria vessel of PHWRs under severe accident condition

Pressurized Heavy Water Reactors (PHWRs) are workhorses of Indian Nuclear Power Program with or 19 PHWRs currently in operation having installed capacity of 5160 MWe. In –general, the safety record of these reactors has been excellent over the years. However, the recent Fukushima accident has compelled the nuclear engineers to have a re-look into the safety systems of these reactors. In PHWRs, a very low probability accident can be postulated involving extreme events accompanied with failure of multiple safety systems which can lead to the collapse of coolant channels which may ultimately relocate to the lower portion of the calandria vessel (CV) forming a particulate bed. Due to the continuous generation of decay heat in the debris, it may melt and form a molten pool (also called as corium) at the bottom of the CV. If not cooled, this will ultimately fail the vessel, release into containment leading to basemat melt through and generation of large amount of gases including hydrogen which pose a threat to containment failure.

Fortunately, the Calandria Vessel is surrounded by tons of vault water, which can act as a core catcher in this scenario. This is termed as in-vessel retention of molten corium. Under these conditions, it is necessary to ensure that the heat flux imposed on the inner wall of calandria due to the melt is less than the maximum heat that can be removed by outer wall (also called critical heat flux (CHF)) at the bottom of the calandria vessel wall. The calandria vessel being 8 meter in diameter with 6 meter in length, it is very difficult to predict



1 Experimental Facility with prototypic curvature

the CHF occurring on the external surface with downward boiling with any heat transfer correlations. Hence, conducting experiments is a necessity. To determine the CHF, experiments were performed on a 25° curved section with actual curvature [1] as that of calandria vessel (Figure 1). The test section was connected to DC electrical supply. The upper side of the test section was insulated so that all the heat flux was directed downward as in actual case. The test section was submerged in vault water. During the experiment, electrical power to the test section was increased in small steps. Temperatures at various locations on test section were monitored. At a particular heat flux, there was a sudden jump in the test section temperature at bottommost point which marks the CHF (Figure 2). The CHF is highly dependent on the vault (bulk) water temperature. Experiments were conducted with different vault water temperatures. It was observed that, at around 75°C vault water, CHF occurs at 565 kW/m². As vault water temperature increases the CHF decreases to 425 kW/m² at saturated conditions (Figure 3). Our earlier experiments on in-vessel retention have determined that, the maximum heat flux imparted on the inner wall of the calandria vessel is around 200 kW/m² [2].This ensures that sufficient thermal margin is available during severe accidents.

There was a concern that, in actual calandria vessel, moderator drain pipes are located at the bottom. This may pose obstruction to sliding of bubbles and can have effect on CHF. The effect of moderator drainpipe on CHF was also investigated by placing drain pipes at the bottom and conducting experiments in similar ways. As the drainpipes are away from the center (6° and 13° location), no effect on CHF was observed. CHF was obtained at bottom of the calandria vessel as in earlier case. These experiments established that, sufficient thermal margin is available for in-vessel retention of corium in calandria vessel during severe accident conditions.

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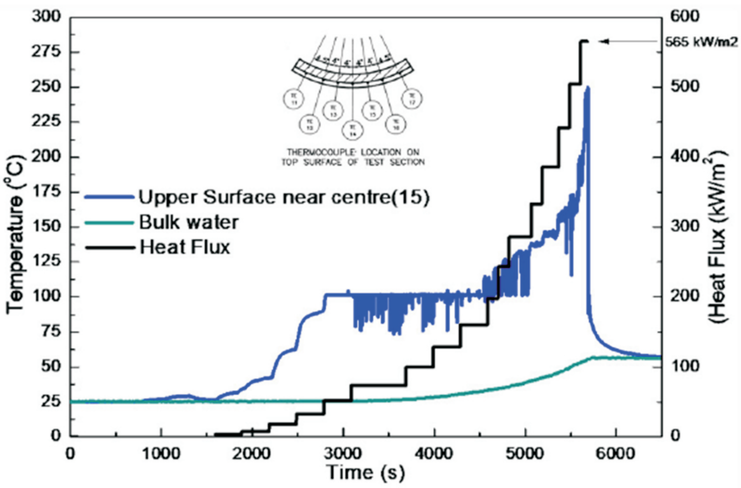

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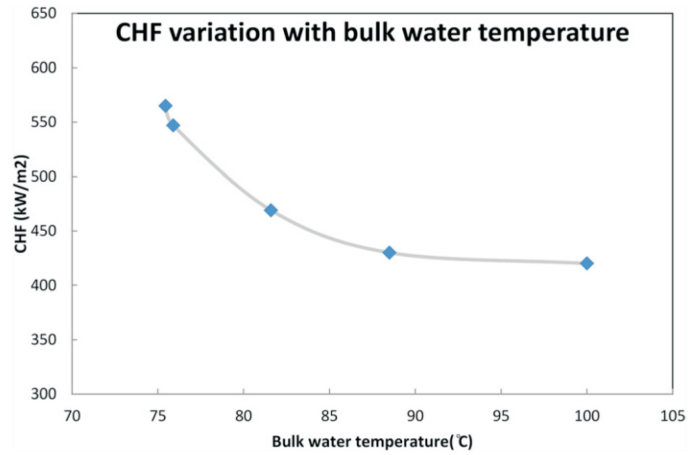

2. Sudden Temperature rise indicating CHF

3. CHF variation with bulk water temperature

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This article was contributed by ***P. P. Kulkarni, P. K. Verma, A. K. Nayak, S. K. Sinha***

Reactor Engineering Division

Full Scale Demonstration of Thermal Margin of AHWR and Development of AHWR-Critical Power Correlation

Theoretically, there is no limit to the power which can be generated in a nuclear reactor core. Practical limit however exists due to the ability to carry away the heat generated and the related phenomenon is termed as the Critical Heat Flux (CHF). The margin available against the CHF is known as thermal margin. Thermal margin is thus an important parameter which limits the reactor power. In AHWR, CHF is one of the important design parameters. Adequate thermal margin must be maintained under normal operation and anticipated transients and, this needs to be demonstrated experimentally in view of the uncertainties in the theoretical predictions of CHF. The fuel cluster power corresponding to CHF

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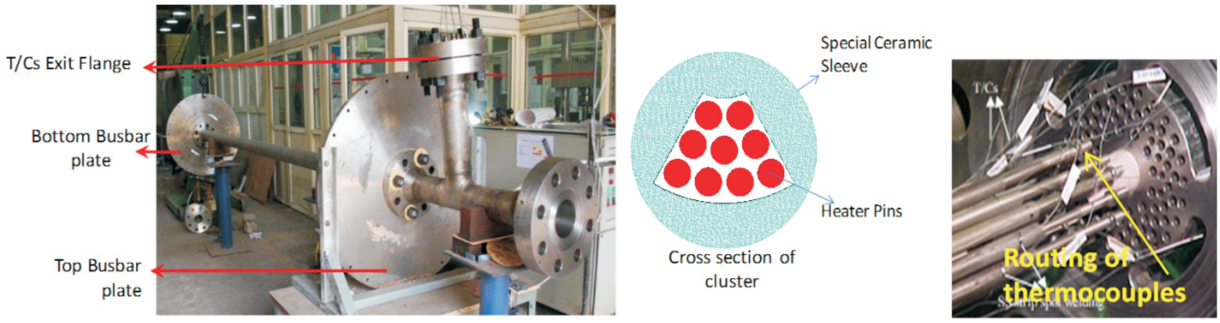

condition is termed as the critical power. The ratio of Critical power to operating power serves as the figure of merit for thermal margin. This is known as Critical Power Ratio (CPR).

CHF phenomena in rod bundles are highly dependent on geometry. Thus, the existing models/correlations of CHF are not applicable to AHWR. Moreover, the CHF correlations for the rod bundle are proprietary in nature. The Jansen-Levy model of CHF was adopted at the design stage of AHWR which was found to be highly conservative when compared with the other approaches like the CHF look-up table and experimental data. Thus, a full scale demonstration of CHF in the rod bundle is necessary for the design validation of AHWR as an important regulatory input. Simulation of nuclear heating using electrical simulator under high pressure and high temperature, mimicking reactor conditions, and detection of CHF occurrence are the challenging tasks for CHF experiments in a rod bundle.

In view of the above, CHF experiments were conducted with simulated fuel rod cluster in 3 MW Boiling Water Loop (BWL) at one to one pressure, temperature and flow conditions of AHWR. Fig. 1 shows assembled fuel rod simulator of AHWR and its sectional view. Fig. 2 depicts the temperature transients during CHF experiments indicating temperature excursion at CHF. The experimental data has led to development of a dedicated AHWR Critical Power Correlation (AHWR-CPC). The form of this correlation is shown below.

$$q''_{CHF} = Ap^{*2} + Bp^{*} + C$$

Where, $p^{*} = p/70$; $A = f_1(G,x)$;

$B = f_2(G,x)$; $C = f_3(G,x)$

p is the pressure in bar, G is the mass flux in kg/m^{2}s, q'' is the heat flux in kW/m^2.

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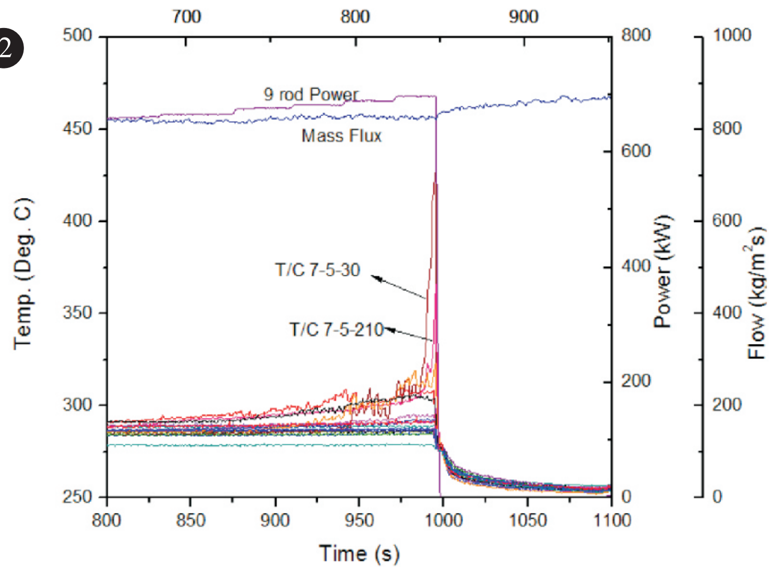

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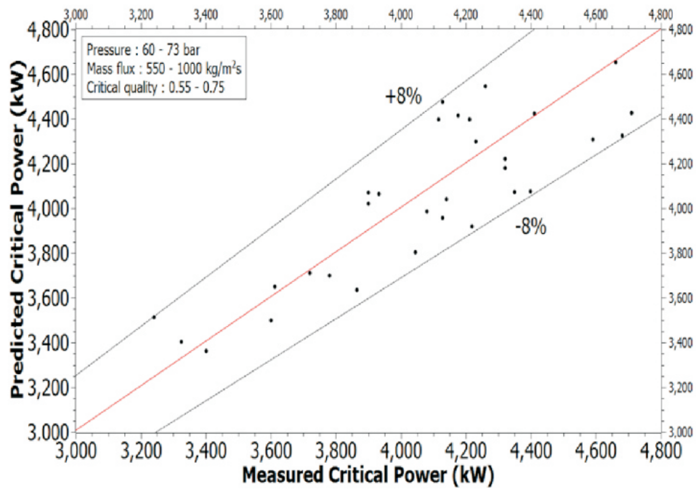


1. Assembled Fuel Rod Cluster Simulator of AHWR with 1/6th symmetry sector and mounting of the thermocouples on the fuel pins

2. Typical critical power experiment of AHWR showing temperature rise at CHF

3. Thermal margin evaluation using AHWR-CPC

The AHWR-CPC correlation was found to predict the CHF within ±8%. The exact nature of the correlation is not revealed due to its proprietary nature. A statistical analysis of correlation data indicates minimum required thermal margin of 1.15 for AHWR-CPC. The experiments established that adequate thermal margin exists in the AHWR design. This is the first time that such rod bundle CHF experiments have been conducted in India under reactor conditions.

This article was contributed by ***D.K. Chandraker, A. Dasgupta, Alok Vishnoi, A.K. Nayak and S. K. Sinha***

Reactor Engineering Division

Thorium fueled small PHWRs present near ready export opportunity

BARC Q&A

Dr. ANIL KAKODKAR
Former Chairman, AEC

Dr. A.K. Nayak, Associate Editor of current issue of BARC Newsletter interacts with Dr. Anil Kakodkar on some of the burning issues on Indian energy security in the context of growing environmental concerns due to climate changes

Dr. A.K. Nayak: ***The growing population and ambitious GDP growth rate in India would require tremendous demand on energy for electricity, transport, industry and domestic sectors. How much do you foresee the demand by 2050 from all these sectors?***

Dr. Anil Kakodkar: As per India Energy Outlook 2021, recently brought out by IEA, India will need to add a power system of the size of the European Union, to what it has now, over the next twenty years to meet the growth in electricity demand. India currently produces around 1600 TWh electricity through its utilities and non-utilities. Power system of European Union is ~ 2800 TWh. Extrapolating these numbers, one can expect India's power system in 2050 a little more than 6000 TWh. This is still short of the electricity consumption required to support a Human Development Index (HDI) comparable to developed countries which is around 8000 TWh.

Dr. A.K. Nayak: ***The world is talking about Clean and Green energy, deep de-carbonization, etc. Do you feel that it is possible to have a carbon free energy supply by 2050 or by the end of this century? If so how?***

Dr. Anil Kakodkar: Four of the top ten CO_2 emitting countries in the world have announced 2050 as the target year for reaching zero emission. China has announced 2060 as the target year for the same purpose. In USA, State of California has set the target year as 2045. Now that USA is back in the Paris process and there is a major traction towards clean energy technologies, it is only a matter of time that USA would be adopting a zero-emission target year. With six of the top ten emitter countries as well as the European Union becoming zero-emission countries by mid-century as recommended in the IPCC Special Report on Global Warming of 1.5°C,there is bound to be a strong push for India, the third largest CO_2 emitter country after China and USA, to define target year for reaching zero emission.

On the other hand, while the developed countries have reached a level of energy consumption far beyond what is necessary to sustain highest level of HDI and there is no need for them for any further increase in per capita electricity generation, India happens to be a developing country that needs to augment its per capita electricity generation around 4-5 folds to reach a HDI comparable to developed countries. As a matter of fact, India's need for additional energy to reach the desired HDI is the highest in the world.

> To my mind the goal of reaching carbon free energy supply by 2050 would not be affordable without a large contribution of nuclear energy

While reaching a carbon free energy supply by 2050 may be a very tall order, we should target reaching that goal as early as possible. While there is a major thrust to renewable energy in that context, to my mind the goal would not be affordable without a large contribution of nuclear energy.

Dr. A.K. Nayak: ***What is the role of nuclear power in the energy mix, and what types of technology would play major role in next 2-3 decades?***

Dr. Anil Kakodkar: Nuclear energy has a major role in India's electricity generation mix to support, in a sustainable way, the large growth in demand that is expected with assured energy security. In the recent times with climate crisis looming large, deep cuts in carbon emission in the current decade reaching to zero emission by middle of this century, have become of paramount importance. That has made rapid deployment of nuclear power, which is a base load clean energy, an urgent necessity. Studies have shown that without a sizeable role of nuclear, reaching zero emission would be unaffordable. Also, there could be significant vulnerability in terms of potential blackouts under extreme weather conditions which could become more frequent. Looking at the available time period one would need to utilize readily available technologies. Clean energy technologies like solar, wind, bio-energy, hydro and nuclear would need to play a major role.

Dr. A.K. Nayak: ***First stage of Indian nuclear power programme has already matured; second stage is still evolving, and technology for third stage has just been initiated. India targets to harness substantial power in second stage and then significantly larger in third stage of its nuclear power programme. What is your vision to accelerate and strengthen the power programme by 2050 and then by 2100?***

Dr. Anil Kakodkar: Three stage nuclear power programme is designed to unleash vast energy potential in our thorium resources. This is necessary for our long-term energy security. This involves development of a number of reactor and nuclear fuel cycle technologies on which work is in progress. We must accelerate this process. While the rationale for this development continues to remain strong since India's needs for additional energy to reach a quality of life comparable to developed countries are the largest; decarbonization of energy supply and use by middle of this century (which is just thirty years away) has become an urgent necessity to protect the earth and our existence on it.

So, there is a need to accelerate nuclear power deployment which is the only near zero carbon - base load electricity production option that can deliver at the necessary scale.

To do this in three decades with realization of deep emission cuts in next ten years as recommended by IPCC would need reliance on readily deployable proven technologies. This is where rapid expansion of PHWR and LWR technology becomes important.

Dr. A.K. Nayak: ***Worldwide, most of the nuclear reactors today require large investments, face delay in supply-chain, incur large civil construction costs, and added safety barriers Post-Fukushima have aggravated the economics of nuclear industry. How could we reduce the capital cost of nuclear plants?***

Dr. Anil Kakodkar: Long construction period and consequential increase in construction outlays

> To realise deep emission cuts in next 10 years, rapid expansion of PHWR and LWR technology becomes important

due to added interest burden as well as and financial risks associated with this and other aspects is one major factor that challenges enthusiasm about nuclear power even when other merits of nuclear power are increasingly well recognized. Making the reactor design simple, safer and standardized with assured continuity for repetitive manufacture and construction in a competitive environment is the key to rapid deployment. I believe that fleet mode construction approach for our 700 MWe PHWRs is an important step in this direction. We must pay attention to other related aspects to facilitate their rapid deployment.

Dr. A.K. Nayak: ***SMRs are being talked about as the "Game Changer", what is your perception?***

Dr. Anil Kakodkar: Small Modular Reactors (SMRs) have been talked about for a long time with shifting drivers. Some of these are safety (small is beautiful), reduced financial risk through lower capital cost for a smaller unit and modular site construction to realize the required capacity, increasing work content in the factory as against at the construction site (modular construction/fully factory assembled units that can be transported to site), barge mounted plants that can be moved to a coastal site and connected to grid and such other logics. Such plants are also a good fit for countries which can accommodate only marginal capacity addition either due to smaller grid sizes as would be the case in many small emerging economy countries or to meet specific needs in advanced countries. However, at the moment their specific capital cost would be higher as compared to large plants unless benefit of mass production can be leveraged. That is a matter of market development requiring a massive policy push. In the Indian context, we need to understand that manufacturing costs of high-tech products in India is way lower as compared to developed countries and unless one is talking about SMR activity with factory assembled units manufactured in India, they are unlikely to be competitive. Having said that, if these units in addition are demonstrated to be so safe that off-site impact in an accident is a non-issue and exclusion radius requirements can be done away with (AHWR development experience can be leveraged here), then there exists an opportunity to rapidly ramp up nuclear capacity by making use of sites of retiring coal plants provided the design can be made site independent and reactors mass manufactured and fully assembled in Indian factories and transported to site. This would require a new approach involving credible Government commitment and policy support, mobilizing a consortium involving public private partnerships and co-working between NPCIL and NTPC.

AHWR development experience can be leveraged for ramping up of factory assembled SMRs in place of retiring coal plants in India

Dr. A.K. Nayak: ***What do you feel about the future of fission technology? Do you think ADSS and fusion technologies would drive the power generation by the end of this century?***

Dr. Anil Kakodkar: To my mind both ADSS as well as fusion technologies are important technologies for securing long term energy needs on earth. However, the first commercial unit in my assessment may be around fifty years away.

We should welcome these developments particularly in the context of abundant clean energy potential these technologies can unleash and transmutation of long-life nuclear waste products.

Dr. A.K. Nayak: ***Nuclear technology is yet to reach the transport sector including civil aviation; what is your vision towards that?***

Dr. Anil Kakodkar: I believe nuclear technology can play an effective role essentially for shipping including transportation under the ice cap and space transportation beyond low earth orbit.

Dr. A.K. Nayak: ***The Russian KLT-40 has created new avenue for floating nuclear reactors, what do you feel about relevance of this technology for India, considering its large population density and limited sites for nuclear expansion?***

Dr. Anil Kakodkar: I think, these units are too small and expensive to make a difference to our capacity addition needs when deployed as single or twin units. However, there may be special needs in specific cases.

Dr. A.K. Nayak: ***AHWR is your brain child. Substantial work has happened towards technology development, proof of concept of this reactor. What are your views for faster deployment of this reactor?***

Dr. Anil Kakodkar: AHWR was conceived with the twin objective of enlarging our experience on thorium utilization for power generation and evolving a safe design leveraging current knowledge and technologies that does not lead to public trauma as was being experienced following a severe accident like Chernobyl then and Fukushima later.

I do believe that we should set up one or two units early and offer the units for export market. This will also take us a bit closer to our third stage which is yet to evolve besides making a significant contribution to mitigating climate change.

Dr. A.K. Nayak: ***To expand the nuclear power from few GW today to several hundreds of GW tomorrow, non-Government funding would play major role. How could this be achieved?***

Artist's impression of AHWR Facility @ BARC Photo

> AHWR serves the twin objective of enlarging our experience on thorium utilization for power generation and in evolving a safe design

Dr. Anil Kakodkar: This is the main challenge. Beyond what Govt. funding can achieve, we need to create conditions to be able to leverage internal resource generation of our energy PSUs (on the basis of clean energy transition for them) as well as market resources on the basis of BOT (Build, Operate and Transfer) model.

Dr. A.K. Nayak: ***Large business potential from radio isotopes are being talked about. What are the perspectives for India?***

Dr. Anil Kakodkar: With vibrant Pharma industry in the country and domestic capability in building large research reactors, we should really be world leaders in radio-isotopes and radio-pharmaceuticals. In past we missed opportunities when there was disruption in production in other countries. We should now leapfrog and at least become a net exporter to start with and then move on to become a global leader.

Dr. A.K. Nayak: ***You have been talking about exporting small sized PHWRs to new entrants using thorium fuel; what are the bottlenecks today?***

Dr. Anil Kakodkar: HALEU-Thorium fuel in PHWR enable some unique advantages. Long burn-up, safer and much reduced spent fuel storage needs, proliferation resistance and superior safety performance.

Qualification of high burn up fuel can be done in test reactors abroad in a couple of years. This should open export opportunity for our well proven and cost competitive 220 MWe PHWRs which is really the need of emerging economy markets.

Export of thorium fueled small PHWRs represents a near ready opportunity for India to make a significant contribution to mitigating climate change besides boosting our exports. As mentioned earlier AHWRs could expand this opportunity further.

Dr. A.K. Nayak: ***You are an ambassador of solar energy. Do you feel that the nuclear has to compete with solar for cost and safety?***

Dr. Anil Kakodkar: Cost competitiveness is always a key factor. Having said that, nuclear and solar energy are complimentary to each other. The energy basket must be diverse, also there is merit in promoting both decentralized as well as centralized generation. I thus see a lot of synergy and not conflict.

> With a vibrant pharma industry and domestic capability in building large research reactors, we should really be world leaders in radio-isotopes and radio-pharmaceuticals

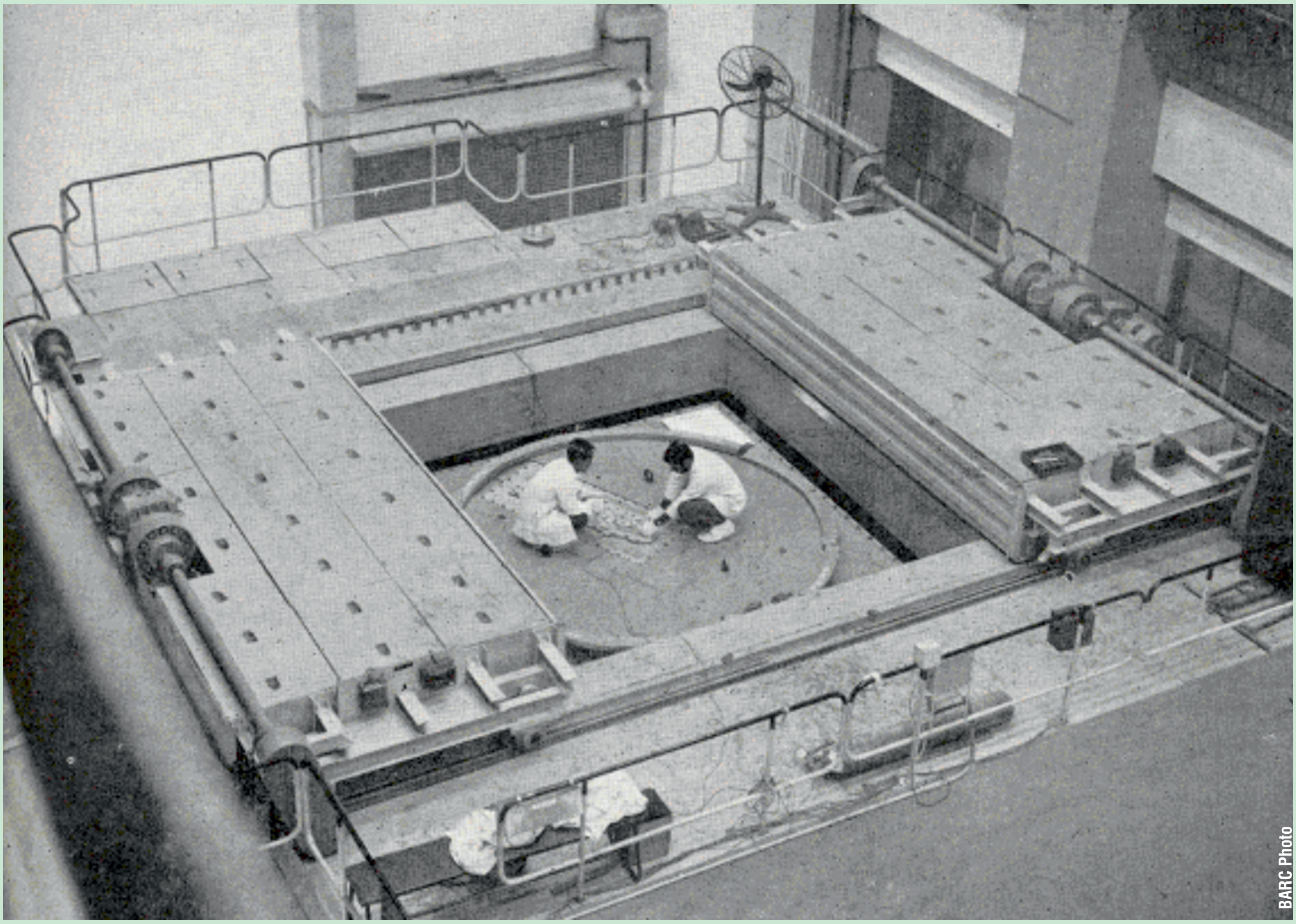
Dr. A.K. Nayak: ***Public acceptance of nuclear power has been a growing concern over decades. What can change the mindset of common people, and how?***

Dr. Anil Kakodkar: Sustained familiarity, being a practical beneficiary and consistently good performance along with meaningful outreach would be the key to greater acceptance of nuclear power. I also believe that taking up holistic development of all people and villages within 5 KM radius under a special planning authority framework utilizing CSR funds and synergizing renewable (wind & solar) energy deployment in the area including in exclusion zone would change NIMBY (not in my backyard) syndrome to Nuclear4Climate syndrome.

The editorial board expresses its deep gratitude to Dr. Anil Kakodkar, Former Chairman, AEC for sparing his valuable time to be the Special Guest for the new year issue of BARC Newsletter-2021, and replying to some of the critical issues on energy front the country faces at present.

● **Dr. Anil Kakodkar** served as the chairman of the Atomic Energy Commission of India and the Secretary to the Government of India during 2000-2009, and was the Director of the BARC, Trombay during 1996–2000. He was awarded the Padma Vibhushan, India's second highest civilian honour, in the year 2009. Apart from playing a major role in India's nuclear tests in 1974 and 1998 asserting India's sovereignty, Dr. Kakodkar had done pioneering work towards India's self-reliance on thorium as a fuel for nuclear energy. Technology development for AHWR and High Temperature Reactors are some of his major contributions for thorium utilization. He led the indigenous development of India's Pressurised Heavy Water Reactor Technology, and designed and built the research reactor Dhruva at Trombay.

● **Dr. A. K. Nayak** joined Reactor Design and Development Group in BARC in 1990.In his career spanning across multiple decades, he made immense contributions on design aspects of Advanced Heavy Water Reactor and on development of innovative passive safety systems of the reactor; he is also popularly known as "severe accident expert". His developmental work on small and modular reactors for addressing growing carbon emissions culminated into an innovative water-cooled 'Passive Safe Integral Reactor'.He is currently Outstanding Scientist and Head, Thermal Hydraulics Section, BARC and Professor, Homi Bhabha National Institute.



Zerlina Reactor

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