India’s three-stage nuclear power programme is chalked out based on the domestic resource position of uranium and thorium. The first stage started with setting up of the Pressurised Heavy Water Reactors (PHWR) based on natural uranium and pressure tube technology. In the second phase the fissile material base will be multiplied in Fast Breeder Reactors using the plutonium obtained from the PHWRs. The third stage is focused on reactors designed to utilise the large thorium reserves based on thorium-233U fuel cycle. The Advanced Heavy Water Reactor (AHWR) has been designed to fulfill the need for the timely development of thorium-based technologies for the entire thorium fuel cycle. This chapter highlights the recent activities carried out in the design and development of the AHWR, such as the core & process system design, nuclear data, fuel design, fuel handling systems, safety analyses, analytical studies and experimental validation.
Nuclear power employing a closed fuel cycle is the only long-term, sustainable option for meeting a major part of the Indian energy demand. Indian resources of thorium are larger than those of uranium. Thorium, therefore, is widely viewed as the ‘fuel of the future’. Thorium-based nuclear fuel cycle possesses several well-known characteristics as indicated below.

**Advantages of Thorium Fuel Cycle**
- Thorium can sustain a thermal breeding cycle using external fissile materials like uranium-235, plutonium or an accelerator driven neutron source.
- The cycle produces virtually no plutonium.
- The waste products contain low amounts of long-lived alpha-emitters

The Indian **Advanced Heavy Water Reactor (AHWR)** is designed and developed to achieve large-scale use of thorium for the generation of commercial nuclear power. This reactor will produce most of its power from thorium, with no external input of uranium 233, in the equilibrium cycle.

AHWR is a 300 MWe, vertical, pressure-tube type, boiling light water cooled, and heavy water moderated reactor. The reactor incorporates a number of passive safety features and is associated with a fuel cycle having reduced environmental impact. At the same time, the reactor possesses several features, which are likely to reduce its capital and operating costs.

**Important Safety Features of AHWR**
- Slightly negative void coefficient of reactivity.
- Passive safety systems working on natural laws.
- Large heat sink in the form of Gravity Driven Water Pool with an inventory of 6000 m³ of water, located near the top of the Reactor Building.
- Removal of heat from core by natural circulation.
- Emergency Core Cooling System injection directly inside the fuel.
- Two independent shutdown systems.

AHWR employs natural circulation for cooling the reactor core under operating and shutdown conditions. All event scenarios initiating from non-availability of main pumps are, therefore, excluded. The Main Heat Transport (MHT) System transports heat from fuel pins to steam drum using boiling light water as the coolant. The MHT system consists of a common circular inlet header from which feeders branch out to the coolant channels in the core. The outlets from the coolant channels are connected to tail pipes carrying steam-water mixture from the individual coolant channels to four steam drums. Steam is separated from the steam-water mixture in steam drums, and is supplied to the turbine. The condensate is heated in moderator heat exchangers and heaters and is returned to steam drums by feed pumps. Four downcomers connect each steam drum to the inlet header.

Emergency Core Cooling System (ECCS) is designed to remove the core heat by passive means in case of a postulated Loss of Coolant Accident (LOCA). In the event of a rupture in the primary coolant pressure boundary, the cooling is initially achieved by a large flow of water from the accumulators. Later, cooling of the core is achieved by the injection of cold water from a Gravity Driven Water Pool (GDWP) located near the top of the reactor building.
In AHWR, subsequent to energy absorption in GDWP in vapour suppression mode, the Passive Containment Cooling System (PCCS) provides long term containment cooling following a postulated LOCA. GDWP serves as a passive heat sink yielding a grace period of three days. The core gets submerged in water long before the end of this period.

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

- Elimination of high-pressure heavy water coolant resulting in reduction of heavy water leakage losses and eliminating heavy water recovery system.
- Recovery of heat generated in the moderator for feed water heating.
- Elimination of major components and equipment such as primary coolant pump and drive motors, associated control and power supply equipment and corresponding saving of electrical power required to run these pumps.
- Shop assembled coolant channels, with features to enable quick replacement of pressure tube alone, without affecting other installed channel components.
- Replacement of steam generators by simpler steam drums.
- Higher steam pressure than in PHWRs.
- Production of 500 m³/day of demineralised water in Multi Effect Desalination Plant by using steam from LP Turbine.
- Hundred years design life of the reactor.
- A design objective of requiring no exclusion zone on account of its advanced safety features.

The AHWR fuel contains 54 fuel pins arranged in three concentric circles surrounding a central displacer rod. The inner two circles contain 30 (Th-233)O₂ fuel pins and the outer circle contains 24 (Th-Pu)O₂ fuel pins. The central rod contains dysprosia in zirconia matrix. The fuel also incorporates a water tube for the spraying of ECCS water directly on fuel pins during a postulated LOCA. AHWR fuel is currently designed for an average burn-up of 24 GWd/t. Its design makes it amenable for reconstitution, if desired to facilitate a further extension of burn-up in the (Th-233)O₂ fuel pins in future.
The AHWR fuel cycle will be self-sufficient in $^{233}$U after initial loading. The spent fuel streams will be reprocessed and thorium and $^{233}$U will then be recycled and reused. The AHWR fuel cycle has enough flexibility to accommodate a large variety of fuelling options. Incidentally, the thorium fuel cycle also presents low proliferation risks, a factor considered significant by several nations for export of nuclear technology. A quantitative analysis of the AHWR fuel cycle substantiates this feature.
### Important Design Parameters of AHWR

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor power</td>
<td>920 MW $\text{t}$, 300 MW $\text{e}$</td>
</tr>
<tr>
<td>Core configuration</td>
<td>Vertical, pressure tube type design</td>
</tr>
<tr>
<td>Coolant</td>
<td>Boiling light water</td>
</tr>
<tr>
<td>Number of coolant channels</td>
<td>452</td>
</tr>
<tr>
<td>Pressure tube ID</td>
<td>120 mm</td>
</tr>
<tr>
<td>Lattice pitch</td>
<td>245 mm (square pitch)</td>
</tr>
<tr>
<td>No. of pins in fuel cluster</td>
<td>54</td>
</tr>
<tr>
<td>Active fuel length</td>
<td>3.5 m</td>
</tr>
<tr>
<td>Total core flow rate</td>
<td>2230 kg/s</td>
</tr>
<tr>
<td>Coolant inlet temperature</td>
<td>259 °C (nominal)</td>
</tr>
<tr>
<td>Feed water temperature</td>
<td>130 °C</td>
</tr>
<tr>
<td>Average steam quality</td>
<td>18.6 %</td>
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<tr>
<td>Steam generation rate</td>
<td>414.4 kg/s</td>
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<tr>
<td>Steam drum pressure</td>
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</tr>
<tr>
<td>MHT loop height</td>
<td>39 m</td>
</tr>
<tr>
<td>Primary shut down system</td>
<td>40 shut off rods</td>
</tr>
<tr>
<td>Secondary shut down system</td>
<td>Liquid poison injection in moderator</td>
</tr>
<tr>
<td>No. of control rods</td>
<td>13</td>
</tr>
</tbody>
</table>
1.1 REACTOR PHYSICS DESIGN & NUCLEAR DATA

1.1.1 Reactor Physics Design of AHWR

The major design mandates are the production of more than 60% of the total power from $^{233}$U and incorporation of inherent safety features such as the negative coolant void coefficient. A composite cluster with (Th, $^{233}$U) MOX fuel in the inner region and (Th, Pu) MOX fuel in the outer region is designed to meet this desired fuel performance characteristics. Coolant void coefficient of reactivity can be made negative in a slightly under-moderated heavy water system at a lattice pitch of 220 mm. For the design pitch of 245 mm, the coolant void coefficient is significantly positive. Hence, a burnable absorber, dysprosium, is used in a centrally located multi-purpose displacer rod, to obtain the negative coolant void coefficient. Other inherent safety features include negative fuel temperature coefficient, negative power coefficient and low power density. In addition, core excess reactivity is low on account of on-power refuelling.

The basic lattice and composite fuel cluster has been designed for 245 mm pitch. The reactor has two independent, functionally diverse, fast acting shut down systems, namely, shutdown system#1 consisting of 40 mechanical boron carbide shut off rods and shutdown system#2 based on liquid poison injection. There are thirteen boron carbide control rods grouped into four regulating rods for fine reactivity control, four absorber rods for xenon override and five shim rods for power setback.

Reactor control and protection systems are designed in a fashion similar to that in currently operating PHWRs except that the reactivity devices occupy lattice locations in the AHWR. A notable feature of AHWR is the xenon override capability to restart reactor anytime following shutdown. Equilibrium core is fuelled to three burn-up zones to achieve a flattened power distribution. Core average discharge burn-up is 24,000 MWd/t.

The cluster design allows use of burnable absorber in the non-fuel multi-purpose displacer rod depending on the fuel type. Use of heavy water as moderator improves neutron economy and relatively soft neutron spectrum imparts flexibility to run it on different fuel-types. During the initial phase plutonium bearing fuel is used to generate $^{233}$U required for composite cluster of the equilibrium core. Equilibrium core will be self-sustaining in $^{233}$U and runs on composite cluster containing both $^{233}$U and plutonium. Two different values of enrichment i.e., 2.5% in the top half and 4.0% in the lower half are used in plutonium bearing fuel pins such that the axial flux peak is shifted away from the high steam quality upper region in the coolant channel. This improves the thermal hydraulic characteristics of fuel channel, with fuel dry-out occurring at a considerably high channel power in comparison to that with a uniform axial enrichment.

Initial core design of the AHWR, approach to equilibrium and refuelling strategy studies along with the development of the flux mapping system, power distribution control system and analysis of various spatial instabilities are being pursued. Physics design to achieve higher fuel burn-ups of about 40,000 MWd/t by lattice pitch reduction and removal of burnable absorber from the fuel cluster are important ongoing activities.

Multi-group data library has been updated for elements of thorium chain and for certain elements specific to AHWR such as burnable absorber dysprosium. Various uncertainties due to thorium data have been identified and experiments are planned in the AHWR critical facility to resolve them. Physics design and preliminary safety analysis for experiments in critical facility have been completed. Core power distribution has been optimized, taking into account the neutronic thermal hydraulic coupling. Studies have been carried out on generation of $^{233}$U required for equilibrium core, achievement of self-sustaining uranium cycle, recycling of uranium and its impact on overall fuel cycle including toxicity of spent fuel.
Nuclear Data and Needs for Thorium Fuel Cycle Studies

The design of innovative reactor systems needs integral data validation studies to generate accurate nuclear data. The nuclear data from experimental measurements need to be evaluated, for arriving at basic point data libraries in the Evaluated Nuclear Data File (ENDF) format. The condensed nuclear data libraries are then used with Monte-Carlo and deterministic codes for reactor physics evaluations of reactor systems. In the context of thorium-based systems, the nuclear data for thorium fuel cycle studies of the AHWR and the CHTR are of current interest. The nuclear data for ADS systems, for neutron energies beyond 20 MeV are also important. Presently Indian nuclear data activities generically encompass the user-oriented approach starting from data files distributed by the IAEA.

India sponsors a regional IAEA-NDS nuclear data Mirror Site (http://www-nds.indcentre.org.in) for the Asian region.

Nuclear data for Isotopes of Thorium Chain

The new concepts involving thorium systems require detailed basic nuclear data measurements and integral validation studies, as thorium has not received the required attention in the past. The nuclear data of the major and minor isotopes $^{230}$Th, $^{231}$Th, $^{231}$Pa, $^{233}$Pa, $^{232}$U, $^{233}$U and $^{234}$U in the thorium fuel cycle needs to be brought to the present level of quality that exists for the isotopes of the U-Pu fuel cycle. For instance, the self-shielded capture resonance integrals for $^{232}$Th are higher in "jendl3.lib" as compared to those in "endfb6.lib" in the final stage of the IAEA WIMS-D update project, by several tens of percent as compared to 0.1% target accuracy. The design of advanced reactor systems, demands accurate nuclear data, in the resonance region that affect plant safety related feedback coefficients such as Doppler and coolant void reactivity effects as a function of burn up.

Critical Facility (CF)

The AHWR simulations are based on assumptions and modeling approximations, which are sensitive to nuclear data uncertainties especially because of thorium fuel cycle. The evaluation of the lattice characteristics requires experimental validation to freeze the design and obtain regulatory clearance, before fuel fabrication. A multi-purpose Critical Facility (CF) has been designed and is in an advanced stage of construction. The sensitivity calculations illustrate that the critical height of the CF with AHWR representative core increases by 5 cm and 7 cm respectively when the "iaea.lib" and "jendl3.lib" libraries replace "endfb6.lib" WIMS-D library. For natural uranium core this is about 2 cm. The replacement of multigroup data of $^{232}$Th alone in "jendl3.lib" by "endfb6.lib" changes the k-infinity by 10.24 mk, "jendl3.lib" yielding a higher calculated value of k-infinity.
Compact High Temperature Reactor (CHTR)

The design of CHTR, which exhibits intermediate neutron spectrum, was strongly influenced by considerations of nuclear data and associated uncertainties during its evolution. The cross sections for several new materials, such as Er, Bi and Ga that were considered for CHTR show large discrepancies in different cross section libraries. It is mandatory to have a negative Doppler feedback effect in the core design. The initial choice of pure $^{233}$U as fuel had been revised due to the calculated positive Doppler reactivity feedback. Further, the calculated Doppler reactivity effect of $^{233}$U has a large uncertainty as the nuclear data of resolved and unresolved resonance region are highly uncertain. As the spectrum covers regions above the thermal range, accurate knowledge of various transport and inelastic cross-sections of various constituents, such as $^{233}$U, $^{232}$Th, Be, Er, Th, which affect the design significantly at high temperatures are required. Experimental work to demonstrate these systems are underway.

Thorium irradiation in FBTR (Kalpakkam)

In the 40 MWt Fast Breeder Test Reactor (FBTR), at Kalpakkam, fifty four thorium subassemblies (717 Kg) have been loaded in the 9th ring in the radial blanket after the nickel reflector. Plans are underway to consider loading additional 100 thorium assemblies in 7th and 8th rings as well. The $^{233}$U produced in this reactor will have low content (5 ppm) of $^{232}$U as compared to several hundreds of ppm in other situations such as a normal fast reactor core, ADSS core with thorium and thoria in PHWR. The reason for the expected low ppm of $^{233}$U in FBTR is understood as due to three factors influenced by nuclear data and physics considerations: Nickel reflector brings the neutrons below the threshold of $(n, 2n)$ reaction in $^{232}$Th. The effective $^{233}$Pa $(n, \gamma)$ cross section is much lower in a fast spectrum as the capture cross section falls rapidly with increasing energy. Thirdly, the accumulation potential of $^{233}$U produced is more in saturation in a fast spectrum making the ppm content of $^{233}$U in $^{232}$U much smaller.

PIE studies of irradiated thorium bundles in PHWRs

Identical loading of thorium bundles was used in KAPP-1 & 2, KAIGA-1 & 2 and RAPS-3 & 4 to attain flux flattening in the initial core. The thorium oxide used is about 400 kg in all the 35 bundles put together in a reactor. The bundles loaded in KAPP-1 & 2, KAIGA-1 & 2 and RAPS-3 & 4 have already been discharged from the core. Samples were obtained from one of the irradiated ThO$_2$ bundles and have been analyzed experimentally by alpha spectrometry for $^{232}$U and by thermal ionization mass spectrometry for $^{233}$U, $^{234}$U, $^{235}$U and $^{236}$U by two different groups in BARC. The previous analyses by two teams in BARC gave a factor of six to eight under-predictions in the production of $^{232}$U. The discrepancy was traced back to the fact that the effective one-group values of cross sections for isotopes of thorium fuel cycle and the use of assumptions in the ORIGEN code are not applicable to the irradiation of thorium in PHWRs. Simulation of the thorium experiment using the new WIMS-D libraries, has been successfully attempted. Sensitivity results of different modeling approaches such as single cell versus super-cell model and treatment of $(n, 2n)$ process (pseudo-fission versus explicit) to prediction of isotopic contents of urania have been obtained. The results are shown in Table.

Generation of integral data by gamma spectrometric analysis of the irradiated thorium fuel is also a part of this activity.

A direct consequence of $^{232}$U concentration in bred $^{233}$U from PHWRs, is its effect on radiation shielding modification in the AHWR critical facility. The outer thickness of the concrete has been increased by nearly 10% to compensate for the additional gamma dose emanating from the fuel clusters based upon experimental results of post irradiation analyses and using the new basic evaluated nuclear data files.

IAEA-CRP on nuclear data for thorium cycle

The active participation in the IAEA-Co-ordinated Research Program, on the "Evaluated Nuclear data for Thorium- Uranium Fuel Cycle" enables to share information and to benefit from the developments related to the use of thorium around the world. An experimental benchmark based upon a 30 kW, $^{233}$U fuelled research reactor KAMINI, at Kalpakkam, is under preparation. KAMINI is a low power research reactor designed and built by a joint venture of BARC and Indira Gandhi Centre for Atomic Research (IGCAR). KAMINI is the only reactor in the world operating with the $^{233}$U fuel. Preparation of a benchmark on thorium irradiation experiments and burnup measurements in PHWRs is also underway.
Reactor Technology & Engineering

Advanced Heavy Water Reactor

Use of updated nuclear data for safety analyses and operation of existing reactors

The Fuel Temperature Coefficient (FTC) of PHWR fuel, calculated by the new 69-group “IAEA.lib” library gives significantly different results at higher burn-ups and explains as a preliminary observation, the unexpected power rise that occurred in the KAPS-1 unit. In a PHWR, the FTC which is negative at low burnups becomes less and less negative and even turns positive at some burnup. The precise crossover point in burnup where the FTC becomes positive depends on many parameters such as the temperature range and 19 versus 37-rod cluster. The FTC is due to the combined effect of Doppler effect and fuel re-thermalization effect. Recent calculations of FTC of PHWR lattices, performed independently by several researchers, illustrate the following: The 27 group WIMS1981 library has a crossover point, for FTC at about 12000 MWd/t burnup; at about 9400 MWd/t with the same but 69-group library, at about 6000 MWd/t for a 19 rod cluster with the new “IAEA.lib” library and at about 4500 MWd/t for 37 rod cluster of PHWR with the “IAEA.lib” library. The crossover point of the FTC is not just the issue but how negative it should be, in order to overcome positive reactivity, that includes the positive xenon kill feedback whenever power transient occurs. The calculated coolant void reactivity using the new “IAEA.lib” library is observed to be lower than the earlier results obtained using the 1971 library. The KAPS-1 overpower transient could be explained only with the use of new IAEA multi-group nuclear data libraries.

International collaborations in experimental programmes

As multiple fuel cycles (e.g., U-Pu, Th-U), with the option of closing the fuel cycle are envisaged, the nuclear data requirements that are needed to develop the new systems with high burnup are demanding and include the entire range of actinides and fission products for multiple fuels. The neutron time of flight experiments to measure neutron induced reaction cross sections from 1eV to 250 MeV, in CERN (Geneva, Switzerland) and neutron transmission experiments using the 150 MeV electron linac at Pohang in South Korea, would help to reduce the existing uncertainties in simulation studies of new and advanced reactor concepts. The new measurements of nuclear data of these experiments and analyses including covariance error information are of importance. Experimental activation measurements to perform neutron source flux characterization in DHRUVA 3001 beam hole has been taken up in order to initiate neutron time.

XnWlup software for inter-comparison of WIMS-D multigroup cross section

An upgraded version of the computer program ‘XnWlup’ has been developed in Visual C++ to produce readily, histogram plots of the multi group cross sections of a selected nuclide, as a function of neutron energy. It also provides the comparison of the nuclear data of different nuclides from different libraries.

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>C/E Ratio</th>
<th>Single cell model</th>
<th>Super cell model</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>PFP</td>
<td>Explicit</td>
<td>PFP</td>
</tr>
<tr>
<td>232U</td>
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<tr>
<td>233U*</td>
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<td>232U/233U (ppm)</td>
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<td>1.22</td>
<td>0.80</td>
</tr>
<tr>
<td>234U</td>
<td>0.81</td>
<td>0.81</td>
<td>0.96</td>
</tr>
<tr>
<td>235U</td>
<td>0.73</td>
<td>0.73</td>
<td>0.92</td>
</tr>
<tr>
<td>238U</td>
<td>0.60</td>
<td>0.60</td>
<td>0.86</td>
</tr>
<tr>
<td>Total U (232, 233, 234, 235, 236)</td>
<td>0.93</td>
<td>0.93</td>
<td>0.98</td>
</tr>
</tbody>
</table>

Comparison of the Isotopic Contents in (J-11-9) Fuel Rod – 3rd Ring.
The upgraded version, Xnwlup3.0 has extra features as sought by the users and includes plots and inter-comparisons of self-shielded resonance integrals at various background dilutions and temperatures.

Nuclear Data for AHWR Analysis

Nuclear data is one of the major inputs for the physics design of advanced reactor systems utilising thorium, such as the AHWR. The basic experimental data of the available Th-U cycle available requires to be established with the accuracy of current day standards. The requirements of accuracy for the basic data of the individual isotopes for thermal reactor systems, will have to be improved in order to predict, for instance, $k$-effective within 0.5\%, feedback coefficients within 10\% and integral reaction rates within 1.0\%. The aim of the task is to generate a processed multi-group data-set for AHWR analysis, which would entail benchmarking, data processing, integral testing, front-end and back-end analysis and sensitivity studies of the systems using thorium. There are several ingredients in this task like, assessment of the energy group structure of thorium cycle evaluations, research towards indigenous data processing capabilities and development of experimental capabilities and nuclear data evaluation methodology.
The availability of the basic nuclear data for the thorium cycle has been assessed and the multi group processed library for isotopes has been updated. The thorium-fuelled lattices have been benchmarked to validate the available basic data. Further the nuclear data set with materials and material composites have also been updated for analysis of the advanced reactor systems. Experiments have been formulated to qualify the nuclear data, by validation through post-irradiation examination.

The review of the basic data isotopes of the Th-U cycle such as $^{232}$Th, $^{233}$Th, $^{234}$Th, $^{231}$Pa, $^{232}$Pa, $^{232}$U, $^{233}$U, $^{234}$U etc. existing as of now exhibit large discrepancies. The thorium fuel cycle data in the WIMS data set has been extended with $^{232}$U and $^{231}$Pa. The $^{232}$U content in the irradiated thoria bundles of PHWR has been estimated with the WIMS dataset and WIMSD code system. The ratio of $^{232}$U/$^{231}$U is calculated as ranging from 450 ppm to 590 ppm and is in good agreement with experimental results.

Sensitivity studies done for AHWR with different evaluated data sets like, ENDF-B/VI, JENDL 3.2 and JEF 2.2 show significant differences. For example, the $^{233}$U absorption reaction rates differ by 5% in thermal energy range and about 10-15 % in higher energy ranges.

The void reactivity studies for AHWR lattice also show a scatter of about 25 %. WIMS data set has been updated with isotopes of dysprosium for AHWR analysis.

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1.2 ADVANCED COMPUTATIONAL TOOLS FOR PHYSICS DESIGN

Space-time Analysis code for AHWR

The knowledge of the space and time dependent behaviour of the neutron flux is important for the reactor safety analysis under operational and accidental conditions.

A time-dependent diffusion theory code called ARK3 (Advanced Heavy Water Reactor Kinetics in 3-D) is being developed for the AHWR. The code ARK3 has option to use advanced Krylov subspace based solution techniques. The space-time analysis code simulates the transient, due to the disturbed reactor steady state, by numerically solving the time dependent diffusion equations. The code is coupled with a visualization tool to plot fluxes as a function of transient time, at any planar cross-section of the reactor core. It is validated against a HWR benchmark problem, which simulates power rise due to half-core coolant voiding and subsequent control action. Representative snapshot flux profiles in two central planar cross-sections when power is at a maximum value, are illustrated.

The code ARK3 is currently being used to analyze AHWR transients such as LORA, operational transient with xenon and validation of reactor physics software in AHWR simulator. This code will be coupled with the thermal-hydraulic analysis code for studying the combined effects of neutronic and thermal hydraulic behaviour.

ATES3 – Anisotropic Transport Equation Solver in 3-D

An accurate prediction of the time dependent multi dimensional & multi energy group neutron flux at successive time instants, is one of the main aspects of reactor physics design. There are primarily two main approaches: deterministic ($S_n$, Collision probability) and Stochastic (Monte Carlo). Often, the reactor core calculations are done with diffusion theory, which is an approximation of the neutron transport theory, a deterministic approach. But an exact transport theory treatment is necessary in several cases such as high leakage reactors, for fluxes at the boundary and beyond, shielding analysis, verification of the approximate methods etc.

Recently, a neutron-gamma transport theory code, called ATES3, has been developed in 3-D Cartesian geometry for steady state criticality and external source problems. Apart from conventional methods of solutions, the code makes use of a few advanced Krylov subspace based schemes. The code is written in Fortran-90 language and has modular structure. These features make it more understandable and comparatively easier to modify.
code ATES3 has been validated against a few international benchmarks and is being subjected for more rigorous testing. Figure below gives the material layout and the corresponding thermal flux shape for an LWR benchmark. The flux dip at the Control Rod (CR) location can be seen clearly.

As is well known, transport problems are highly memory and CPU time intensive problems, a single PC or workstation is not sufficient. Hence, it is very important to adopt the present code to parallel computers. Efforts are being made to parallelize the code on BARC’s ANUPAM parallel systems. Incorporation of methods of solutions and user-friendly advanced features like visualization tools etc. are being incorporated.

**Monte Carlo Technique: Code Development and Reactor Physics Simulation**

Monte Carlo, as a tool in numerical analysis has gained wide spread applicability over the past few decades. The advent of high speed computing machines has been mainly responsible for the continual development of Monte Carlo method. Used properly, Monte Carlo can give quick “first cuts” at difficult problems, that is problems which are intractable by the traditional analytical or numerical techniques.

The greatest advantage of the Monte Carlo method is the exact simulation of the geometry. In deterministic methods only some special geometry can be simulated exactly, for irregular geometry some approximations must be considered. Monte Carlo method does not take any approximations in defining geometry. For this reason Monte Carlo method is essential for reactor calculations which involves complicated geometry e.g. Secondary shutdown system of 500 MWe PHWR, hexagonal geometry of CHTR, Nuclear Power Pack, Pebble bed reactors etc. as well as for deep penetration problems.

The main objective is to develop a general geometry Monte Carlo code with burn up, which will be used for criticality calculations, safety evaluations, accelerator driven sub-critical system’s calculations, shielding calculations etc. with greater confidence and wider flexibility.

**Development of Random Number Generator**

Random number plays an important role in any Monte Carlo calculation. The accuracy of the results depends on the randomness of the random numbers, its uniformity and its cycle length.

To provide uniform random sequences having larger cycle length required for Monte Carlo calculations a Random Number Generator (RNG) with large cycle length ($2^{57}$) has been developed using bit manipulation technique. Some of its properties namely uniformity, Expectation Value, Variance, Frequency distribution, Auto-Correlation, Chi-square test etc. have been performed. It was compared with RANDU of PC in FORTRAN, RAND of PC in Basic, RAND of Honeywell DPS-8 System and RAN of PDP-11/23 and found to be superior among
Comparison of Auto correlation coefficient

<table>
<thead>
<tr>
<th>Types of RNG Used</th>
<th>Auto correlation coefficient between neighboring bits</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>N &amp; N+1</td>
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<tr>
<td>Current</td>
<td>-0.0001173</td>
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<td>Randy of PC in Fortran</td>
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<tr>
<td>Randy of PC in Basic</td>
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<td>And of Honeywell DPS-6</td>
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<tr>
<td>And of PCAP-1123</td>
<td>0.0009187</td>
</tr>
</tbody>
</table>

* Number of sample = 10⁷

Development of 69-group spherical geometry Criticality Code

Monte Carlo code for criticality calculation has been developed for spherical geometry with WIMS 69 group energy treatment. This code is being extended for AHWR/PHWR lattice cell with WIMS 69 group cross-section data.

Simulations of Reactivity Induced Transients for Thermal and Fast Reactors and Stability Studies

An accurate prediction of the consequences of an accident in a nuclear reactor is vital from the reactor safety point of view. This in turn requires the solution of coupled time-dependent neutron diffusion equations, time-dependent heat conduction equations and single and two phase coolant dynamics equations. All of these require large computer memory and computational time. Present day large-sized power reactors are neutronically loosely coupled. The looseness of the coupling is further enhanced by the deliberate flattening of the power distribution. The study of the neutronic transient behaviour under accidental conditions in such reactors requires accurate methods of solution of system of coupled multidimensional multi energy group time dependent neutron diffusion equation. Two distinct approaches exist for this purpose namely; the direct (implicit time differencing) and Improved Quasistatic (IQS) approach. Both the approaches need solution of static space energy dependent neutron diffusion equations at successive time steps.

A three-dimensional computer code 3D-FAST was developed based on Incomplete LU (ILU) preconditioned Biconjugate Stabilized method. The code was parallelized on ANUPAM distributed memory parallel system. The domain decomposition technique was used to create parallelism. The parallel
computational scheme was tested by analyzing a well-known Canadian PHWR benchmark problem, which simulates a loss of coolant accident.

The transient was simulated using two energy groups and $52 \times 52 \times 40$ meshes. Twenty-nine space and energy dependent calculations were done with time step of the order of 0.1 sec. Table presents the CPU gain due to parallelization. The code was used to analyze the inadvertent withdrawal of two control rods along with drainage of light water from the zone controller units (ZCUs) for 540 MWe PHWR. Figures show the variation of reactivity and power as a function of time for this transient.

The accident analysis of fast reactors is generally carried out in two phases. The first phase is generally called as pre-disassembly phase and the second one as disassembly phase. In pre-disassembly phase, the transient is analyzed up to coolant voiding and fuel melting. The disassembly phase calculations are carried out with reactivity rates estimated from coolant voiding and fuel slumping. These transients are terminated by the disassembly of the core, which introduces sufficient negative reactivity. The calculation of disassembly reactivity requires the solution of coupled neutronics and hydrodynamics equations. A computer code for pre-disassembly calculations, which calculates coolant voiding, fuel melting, fuel and clad deformation and molten fuel slumping, is being developed. For the disassembly phase a computer code DISA is developed. This solves point kinetics equations coupled with two

<table>
<thead>
<tr>
<th>Parallel System</th>
<th>No. of Slave processors</th>
<th>Solver CPU time (sec)</th>
<th>Speed-up by parallelisation</th>
<th>Parallel efficiency</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANUPAM-PC</td>
<td>Sequential</td>
<td>803</td>
<td>--</td>
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</tr>
<tr>
<td></td>
<td>2</td>
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<td>10</td>
<td>130</td>
<td>6.18</td>
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CPU times for Parallelised BiCGSTAB(ILU) for IQS Approach ($e = 10^{-6}$)

Fig. 5: Energy-pressure relationship in the ANL EOS

Fig. 8: A: Adiabatic, B: Improved Quasistatic
dimensional hydrodynamics equations. Figures show the equation of state for fuel used in DISA. The variation of net reactivity and power as function of time for a hypothetical transient in a typical fast reactor are shown in figures. It is planned to improve the neutronics model of pre-disassembly code by replacing the point kinetics calculations by multidimensional, multi energy group neutron diffusion code 3D-FAST. It is also planned to couple both phase calculations.

New concepts have emerged in the dynamics of nonlinear systems in last two decades. As part of our nonlinear studies in reactor physics we have studied some of these issues for a typical PWR. Here the dynamics refer to a single-phase coolant using point kinetics and a feedback through fuel & coolant temperature coefficient of reactivity. Typical scenario of limit cycle reactor operation were observed as a function of coolant temperature coefficient. The temporal behaviors can be identified for certain values of this parameter.

For a specific value of coolant temperature coefficient the critical state becomes an oscillatory state (limit cycle). This latter state constitutes a new operational regime for reactor dynamics. These studies contribute towards understanding safety and performance of reactors.
As a result of the IAEA coordinated research program entitled “Final Stage of the WIMS library Update Project” new and updated WIMS-D libraries are generated by processing evaluated nuclear data files such as ENDF/VI.6, JENDL-3.2 and JEF-2.2. These WIMS-D libraries provide knowledge about the various relevant neutron-nuclear cross sections data in the form of 69/172 neutron energy groups. In order to help the WIMS-D library users to quickly view the plots of the energy dependence of the multi-group cross sections of any nuclide of interest, a computer program ‘XnWlup’ is developed for MS-Win operating system using Microsoft Visual C++. It is also possible for the WIMS-D library users to compare the energy dependence of cross section data of various nuclides, different WIMS-D libraries and different temperatures.

The first version of this software ‘XnWlup1.0’ helps to obtain the histogram plots of the values of cross section data of an element/isotope as a function of energy. The second version of this software ‘XnWlup2.0’ is serving as an exhaustive equivalent handbook of WIMS-D cross section libraries for thermal reactor applications and used for comparing different WIMS-D compatible nuclear data libraries originating from various countries. The next version of this software ‘XnWlup3.0’ was developed to plot the cross sections of a resonant nuclide using resonance integral tabulated data of WIMS-D library for the given background dilution cross section and temperature. Also the revised software ‘XnWlup3.0’ is now capable of plotting either the resonance integral data as a function of dilution cross section for a selected temperature grid point or as a function of temperature for a selected dilution cross section grid point for a given resonance energy group.
Illustration of plots of absorption resonance integral data of $^{232}$Th at 600 K and for various background dilution cross sections.
1.3 REACTOR SHIELDING AND IRRADIATION EXPERIMENTS

AHWR Shielding Experiments at APSARA

Shielding experiments essentially study the neutron and gamma streaming through duct geometries of shield models, to optimize and validate the actual reactor shield designs. Streaming experiments have been carried out at the shielding corners of the Apsara research reactor to optimize the end shield design of the AHWR. These experiments have generated data on neutron and gamma streaming through ducts of various sizes and shapes, present in the AHWR end shield. These duct geometries represent various configurations of lattice tubes and ventilation ducts.

The experiments included reference model without any streaming path, model simulating ECCS duct, model simulating stepped ventilation duct, lattice tube simulation model-1 (empty inside cavity, water filled cavity and inside cavity filled with shield plug with and without water), lattice tube simulation model-2 (similar to the model-1 except that it is surrounded with a layer of steel balls and water). Results of the lattice tube simulation model-2 experiments, in which lattice tube, end fittings and shield plug have been simulated are given in the illustrations.

Measurements have been carried out along the center of lattice tube filled with air, water and shield plug. Radiation streaming through 3mm air gap between lattice tube and end fitting has also been studied. In all these experiments shield models simulating only one lattice tube is used. Experiments with a more representative model which has four lattice tubes and in-between space filled with steel balls and water is underway.

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Material Irradiation in Cirus / Dhruva Reactors for AHWR Program

Carbon Irradiation in Dhruva Reactor

Samples of pyrocarbon have been irradiated in the Dhruva Reactor, to study the fast neutron fluence effect on its properties like wigner energy. A pyrocarbon sample has been irradiated in Dhruva reactor core, at location TR-15, 04 A for 1105.359 MWd. During the irradiation the fast neutron fluence, seen by pyrocarbon sample has been measured by activation method. In the measurement threshold neutron detectors like Ti, Ni and Fe have been used. After irradiation gamma activities of these irradiated detectors have been measured in a standardized high purity germanium detector coupled to a multi-channel analyzer. The fast neutron fluence (>1MeV) seen by the pyrocarbon sample was measured to be 1.28E+18 n/cm² at 100 MW power of the reactor. Further irradiations of pyrocarbon samples are planned at different fast neutron fluence levels.

Dysprosium irradiation

Dysprosium is used as a burnable absorber in the AHWR, as Dy₂O₃ in ZrO₂, to achieve desirable reactor physics characteristics. Natural dysprosium mainly consists of five isotopes with mass number 160 to164. Dy-164 has largest absorption cross-section for thermal neutrons, the others have much lower but significant absorption cross sections. It is a chain absorber and the end product Holmium is a stable isotope, which is also a thermal neutron absorber. Some of the isotopes are good resonance absorbers for epithermal neutrons. It’s suitability for AHWR is due to its slow burning characteristics and high radiation stability.

The cross sections of Dysprosium isotopes have not been validated systematically. Hence it has been planned to carry out irradiations of varying dysprosium content in ZrO₂ pellets in Dhruva or Cirus tray rods over a period of time. Tentatively it is desired to have three different dysprosium weight % enrichment of 3%, 5% and 7% in ZrO₂ pellets. Currently, pellets of Al₂O₃ (95%) and Dy₂O₃ (5%) have been made for initial studies.

It has been planned to carry out irradiations at low power along with thermal, epithermal flux monitors for short duration and another one for a longer duration for measuring isotopic composition change by mass spectograph. The characteristic gamma activity of the flux monitors will be measured by a HPGe detector and the data will be analyzed to get thermal and epithermal neutron flux distribution in and around the pellets.
1.4 REACTOR SAFETY ANALYSIS

Preliminary Safety Analysis of AHWR

Safety analysis is a major part of the process and licensing requirements of a nuclear power plant. The passive features of the AHWR cover entire range of normal operating conditions, transient conditions, accident conditions and beyond design basis accidents. The safety analysis exercise for the AHWR has been carried out in two parts namely deterministic safety analysis and probabilistic safety analysis. This article gives overview of the first facet. The deterministic approach requires establishment of set of Postulated Initiating Events (PIEs) as a starting point. The PIEs are selected from different categories based on phenomena encountered and AHWR specific reactor characteristics. This is followed by finalizing the reactor trip parameters, ECCS acceptance, fuel failure criteria and parameters required to build up the simulation model namely geometric, thermal-hydraulic and reactor kinetics with feedback. Safety analysis code RELAP5/MOD 3.2 has been used extensively to carry out all the analyses along with limited usage of subchannel analysis computer code COBRA IV.

Analytical computer code specific to AHWR plant model has been developed to address the variation in thermal-hydraulic parameters in each system expected from the PIEs. The plant model involves simulation of various systems and their components. This includes Main Heat Transport System (MHT), Emergency Core Cooling System (ECCS), Containment, Isolation Condensers (ICs), Moderator System and MHT purification system. Other important modeling for systems to be mentioned in the plant models are steam drum pressure-level controller and reactor regulation-protection system.
A set of 55 postulated initiating events have been analysed for 750 MWt and up rated 920 MWt designs to address plant behaviour to judge the adequacy of the design with respect to ECCS acceptance and fuel failure criteria. Contribution from the analyses were utilized for deciding

- Design pressure of the containment and main heat transport system
- Defuelling - refuelling time to avoid boiling crisis,
- Necessity of compartmentalization of ECCS header to avoid core starvation on ECCS header break,
- Assessment on hot shut down condition with ICs submerged in GDWP,
- Assessment of relief devices capacity to ascertain the integrity of pressure boundary
- Assessment of adequacy of ECCS along with mode of in core safety injection.

Studies are also carried out for optimization of ECCS design parameters and evaluation of moderator as an ultimate heat sink.

This whole task has been achieved with active participation of the analysis group, reactor trip committee, task force committee, design review committee, preliminary safety analysis review committee and safety documentation committee. In the light of number of design changes in 920 MWt design and re-assessment of reactor trip parameters, a fresh evaluation of the proposed reactor will be carried out with respect to ECCS acceptance and fuel failure criteria covering pre-decided PIEs. Various optimization studies are underway. Limited uncertainty analyses has been done to support best estimate evaluation.

Clad surface temperature transients at different axial locations for maximum rated reactor channel for event like 200% Inlet Header Break is illustrated.

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Level-1 PSA Study of the Advanced Heavy Water Reactor

Preliminary Level-1 Probabilistic Safety Assessment (PSA) study of the AHWR, has been carried out to obtain integrated statement of Core Damage Frequency (CDF). Important accident sequences suggest design modifications and to obtain insights about important dependencies and human interactions important to safety. Based on this study, design modifications have been suggested in emergency core cooling system, isolation condenser system, end shield cooling system and gravity-driven water pool recirculation system. This study has indicated that the core damage frequency of this reactor based on suggested modifications is of the order of $10^{-7}$ per reactor year. The dominating initiating event contributing to this CDF value is small LOCA. In arriving at this figure, it has been assumed that the passive systems are completely reliable. However, this needs to be ensured by carrying out thermal hydraulic studies. Also, the contribution from small LOCA to CDF will reduce if Leak Before Break criteria is incorporated in the design. Revised PSA study of this reactor is being carried out using the latest design documents.

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## Assessment of Leak Rate from the Cracked Pipe

The Leak Before Break (LBB) concept is employed in defense in depth concepts of reactor design to avoid any unstable failure in the pressure boundary, namely carbon steel pipes and Zirconium made pressure tubes. The LBB in the pressure boundary can be ensured when detection, confirmation and location of the leak are carried out and the reactor is placed in a depressurized condition before the crack exceeds the critical crack length. The LBB concept has been applied for designing the AHWR reactor.

Variation of Leak Flow Rate with Crack Angle
Comprehensive theoretical and experimental study is going on to support the LBB concept applied to Primary Heat Transport System and steam lines for the AHWR and the PHWR. The theoretical study includes elasto-plastic model development for crack opening area and thermal-hydraulic model development for estimating critical flow rate through crack and slits. Over the years various flow models are developed and validated against the published data specific to Pressurised Water Reactors (PWRs). Prediction of crack flow with crack angle variation for PHWR Steam Generator (SG) outlet pipe is furnished (Fig.). Influence of different flow models namely Henry’s Homogeneous Non-Equilibrium Model (HHNM) and Homogeneous Frozen Model (HFM) on crack flow rate is also shown. A Comparison of the computed flow with flow from detectable Leakage Size Crack (LSC) gives an idea of the order of semi crack angle which gives a detectable crack flow.

An experimental setup is built at Mechanical Engineering Department, Jadavpur University, Kolkata to estimate the crack/slit flow under high pressure and high temperature conditions simulating the pressure and temperature condition of primary heat transport system of the PHWR and the AHWR. The maximum operating parameters are 90 bar and 250º C. The facility consists of three systems namely the heating system achieved through oil-fired thermic fluid heater, the pressurization system through nitrogen system and the high pressure and high temperature system which consist of buffer chamber and test section. The system is instrumented with pressure, flow and temperature sensors. Safety devices and control logic interlocks have been incorporated to ensure safety in the loop. Four close circuit cameras continuously monitor the health of the system. Hot commissioning followed by some experiments in slits has been carried out. Series of experiments are planned with slits and cracked pipe to generate a robust database with different pressure and different sub-cooling to validate the developed computer code.

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- **Nuclear Containment safety Research for Beyond Design Basis Accidents**

For the ultimate load capacity assessment of Indian reactors, in-house finite element codes ULCA for static inelastic analysis and ARCO3SD for transient dynamic inelastic analysis have been developed. These computer codes have been extensively benchmarked with experimental test results of Sandia Laboratory, NUPEC, Japan and USNRC-sponsored containment model test and NUPEC, Japan sponsored seismic shear wall tests. Indian containments have been shown to have a factor of safety of more than 2 over the design pressure with these codes.

Recently Sandia Laboratory, USNRC and NUPEC, Japan co-sponsored a round robin analysis program to determine the ultimate load capacity of a 1:4 size steel lined Pre-Stressed Concrete Containment Vessel (PCCV) model which represents the Pressurized Water Reactor (PWR) containment of Ohi-3 nuclear power station in Japan. BARC ULCA code pre test predictions have been shown to be in excellent agreement with the experimental limit state test (LST) and subsequently the Structural Failure Mode Test (SFMT) results released in 2003 by the organizers.
The PCCV model has the design pressure ($P_d$) of 0.39 MPa and the best estimate of the ultimate pressure of 3.15 $P_d$, the lower bound conservative estimate of the minimum pressure of 2.8 $P_d$, which the test model would at least reach during the test and upper bound estimate of the maximum collapse pressure of 3.45 $P_d$; beyond which the test model is unlikely to remain in pressurized condition were predicted with the in-house code with 90% confidence level as per the requirements of the round robin analysis activity. The subsequent test carried out at Sandia showed that the model had the limit state test pressure of 3.3 $P_d$ and the structural failure pressure of 3.6 $P_d$. BARC computational results have been found to be among the best of all the predictions.

The pre-test predictions with BARC code ULCA published in international journals before the release of the experimental data are in excellent agreement with the test results.
Evaluation of Ultimate Load Capacity of AHWR Containment Structure

The nonlinear finite element analysis of the inner containment structure of Advanced Heavy Water Reactor has been carried out to determine its ultimate load capacity. The deformed shape of the containment at the ultimate pressure as shown in figure and the internal pressure versus deformation curve at the mid height of cylinder shown in figure obtained from the analysis illustrates a factor of safety of approximately 2.78 over the design pressure of 0.185 N/mm². The major safety-related highlights from the analysis are:

- The crack initiation starts at internal pressure of 0.462 N/mm².
- The through thickness cracking is observed at an internal pressure of 0.467 N/mm².
- The reinforcement yielding starts at an internal pressure of 0.510 N/mm².
- The 0.2 mm crack width is observed at an internal pressure value of 0.515 N/mm².

1.5 DESIGN AND ANALYSIS OF AHWR FUEL

Design of AHWR fuel

The AHWR fuel has been designed to meet the requirement of thermal hydraulics, reactor physics, fuel handling, and reconstitution (i.e., replacement of outer ring of irradiated (Th-Pu)O₂ fuel pins with fresh ones). The vertical pressure tube configuration has guided the structural design of the fuel assembly. The fuel assembly is 10.5 m in length and is suspended from the top in the coolant channel.

The AHWR fuel assembly consists of a fuel cluster sub-assembly and two shield sub-assemblies: shield ‘A’ and shield ‘B’. These sub-assemblies are connected to each other through a collet joint. The AHWR fuel cluster contains (Th-²³²U)O₂ and (Th-Pu)O₂ fuel. The fuel cluster has 54 pins arranged in three rings—the innermost ring of 12 pins of (Th-²³²U)O₂ with 3.0% ²³³U enrichment, middle ring of 18 pins of (Th-²³²U)O₂ with 3.75% ²³³U enrichment, and outermost ring of 24 pins of (Th-Pu)O₂ with Pu enrichment of 2.5% in the upper half and 4.0% in the lower half of the pin. The fuel pins are assembled in the form of a cluster, with the help of the top and bottom tie-plates, with a central rod connecting the two tie-plates. The central rod has a burnable absorber Dy₂O₃ in ZrO₂ matrix in the form of capsules. Each capsule consists of 12 pins. The central rod has provision for spraying of Emergency Core Cooling.
System (ECCS) water directly over the fuel pins in the event of Loss of Coolant Accident. Six spacers along the length of the cluster provide the intermediate pin spacing and stiffness to the cluster. Various components of the fuel cluster like dummy fuel pins, shields, top and bottom tie plates, spacers, central rod, collet joints have been designed and fabricated. A dummy fuel cluster has been assembled and pressure drop test, endurance test and vibration tests have been carried out for the cluster at Flow Test Facility and Integral Test Facility. A short length model of the fuel cluster showing different components of the fuel cluster has been made for display and for better understanding of the fuel cluster.

The injection of water from the holes of the central rod was visualised in a test set up and its flow versus pressure drop characteristics were obtained. A set up has been designed for studying the wetting of the fuel pins and effectiveness of spray cooling of ECCS water using electrically heated pins. Leakage of water from the flexi collet joint has also been evaluated.

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■ Fuel cycle analysis for AHWR

The AHWR follows a closed fuel cycle with the objective of utilisation of plutonium to a minimum and self-sustaining in \(^{233}\text{U}\). The \(^{233}\text{U}\) required for the equilibrium core is planned to be generated in-situ. Hence, the initial core of the AHWR would consist of all \((\text{Th-Pu})_2\text{O}_3\) fuel clusters. There will be a gradual transition from initial core \((\text{Th-Pu})_2\text{O}_3\) clusters to equilibrium core containing both \((\text{Th-Pu})_2\text{O}_3\) and \((\text{Th-}^{233}\text{U})_2\text{O}_3\) fuel. The process losses of fissile and fertile material in the reprocessing and refabrication should be considered while carrying out the fuel cycle evaluations. The fuel cycle analysis detailing the movement of fissile and fertile materials for various combinations of design parameters – from initial core to transition core to equilibrium cored is being carried out.

Near term fuel cycle for the AHWR

The AHWR fuel cycle for the near term, is a closed fuel cycle, envisaging recycle of both fissile \(^{233}\text{U}\) and fertile thorium. The fuel cycle time is eight years - four years of in-reactor residence time, two years of cooling, one year of reprocessing and one year of re-fabrication. For the initial few years, annual reload would consist of only \((\text{Th-Pu})_2\text{O}_3\) clusters. The spent fuel cluster before reprocessing would undergo dis-assembly for segregation of \((\text{Th-Pu})_2\text{O}_3\) pins, \((\text{Th-}^{233}\text{U})_2\text{O}_3\) pins, structural materials and burnable absorbers. The \((\text{Th-}^{233}\text{U})_2\text{O}_3\) pins will require a two stream reprocessing process i.e. separation of thorium and uranium whereas the \((\text{Th-Pu})_2\text{O}_3\) pins will require a three stream reprocessing process i.e. separation of thorium, uranium and plutonium. Part of the recovered thoria will be recycled into the reactor immediately by using it for the fabrication of \((\text{Th-}^{233}\text{U})_2\text{O}_3\) pins. The rest will be stored for 17-20 years by which time it would be similar to freshly mined thorium in radioactivity and hence will be used to fabricate \((\text{Th-Pu})_2\text{O}_3\) pins.

Long term fuel cycle for the AHWR

India’s three-stage nuclear power programme, based on closed fuel cycle, comprises Pressurised Heavy Water Reactors (PHWRs), Fast Breeder Reactors (FBR) and systems for thorium utilisation.
Natural UO₂ from PHWRs produces plutonium which along with Depleted Uranium will be used in FBRs for power generation and fissile material multiplication. The plutonium required for the AHWR will come from the PHWRs. Over a long period of time, part of the plutonium from FBR may also be used in AHWR. The plutonium from reprocessing of AHWR fuel has low fissile content and will be sent to FBR.

Partitioning of waste is planned for AHWR. The minor actinides like Np, Am and Cm would be separated from the waste. Similarly higher decay heat materials like Strontium and Cesium and long-lived fission products like Iodine and Technetium would be separated from the bulk of the waste. The R&D work on partitioning has already commenced. The partitioning technology developed will be equally applicable to both PHWR and AHWR. Transmutation of minor actinides like Neptunium, Americium and Curium and long-lived fission products like Iodine and Technetium, segregated from bulk of the waste from AHWR and PHWR, is planned for the long-term fuel cycle for AHWR. The design of AHWR offers flexibility to incorporate a wide range of fuel cycles, including those based on (Pu-Th)O₂ alone.
Fuel analysis for AHWR

The performance analysis essentially involves the thermo-mechanical analysis of fuel by modelling the various phenomena that take place during its irradiation. The analysis has been carried out by employing the methodology used for the well-established UO$_2$ fuel using GAPCON-THERMAL code. The available database on the thoria-based fuels is quite small. However, even the most conservative estimates indicate that the thoria-based fuels have an inherent advantage over the UO$_2$ fuel in terms of the two major performance bearing parameters thermal conductivity and fission gas release. Thermal conductivity is considerably higher and the diffusion coefficient, which is a measure of fission gas release, is lower for thoria-based fuel. The thermo-physical properties of (Th-Pu)O$_2$ fuel have been incorporated in the code for carrying out the analysis. The salient results of the analysis for various limiting parameters are given in the Table.

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1.6 CRITICAL FACILITY

A low power Critical Facility is under construction as part of over-all technology development program to validate the physics design of the thorium-based Advanced Heavy Water Reactor (AHWR) and for validation of safety parameters for 540/700 MWe PHWRs as well as advanced fuel designs for the PHWRs.

The nuclear design of the lattice and core of AHWR envisages the use of a novel 54-pin MOX cluster with different enrichment of $^{235}$U and $^{239}$Pu in Thoria fuel pins with a dysprosium displacer rod at the centre. The nuclear design of the lattice and core of AHWR envisages the use of a novel 54-pin MOX cluster with different enrichment of $^{233}$U and $^{239}$Pu in Thoria fuel pins with a dysprosium displacer rod at the centre. It is designed to have negative void coefficient of reactivity. Fuel assembly characteristics are evaluated by using transport theory codes WIMS or CLUB with 69 or 172 group library. The core calculations are done with few group diffusion theory methods. For new fuel assembly concepts, as in the AHWR, lattice and core characteristics have to be measured in Critical Facility to validate the above calculations.

The Critical Facility is designed to facilitate study of different core lattices based on various fuel types, moderator materials and reactivity control devices. The reactor is designed for a nominal fission power of 100 W with an average flux of 108 n/cm²/s. The design provides flexibility to arrange the fuel inside the core in a precise geometry at the desired pitch. Reactor criticality is achieved by the manual control of moderator level in the core.

Initially, three types of cores using heavy water as moderator and reflector will be studied. The three cores are based on different fuel types viz.

- 19 pin natural uranium metal fuel clusters to constitute the reference core.
- 54 pin (Th-Pu) MOX/ (Th-$^{233}$U) MOX clusters to constitute the representative AHWR core.
- 37 pin natural uranium oxide fuel clusters to constitute the 540 MWe PHWR core.

Reactor Structure

The reactor consists of a cylindrical Aluminum tank (Reactor Tank) of 330 cm ID and 500 cm height to accommodate the fuel, moderator and the shut-off rods. The reactor tank is adequately sized to provide a radial reflector thickness (heavy water) of about 40 cm around the core. The reactor tank, which is open at the top, is connected to a square box housing the lattice girders from which the fuel assemblies are suspended. Shut-off rod head-gears are also supported by these girders. The lattice girders are designed to provide the flexibility of arranging the fuel clusters in any desired square lattice pitch ≥206 mm. The reactivity addition/adjustment of the reactor is carried out manually through controlled addition of moderator heavy water to the reactor tank at a pre-determined rate, to ensure safe limits of reactivity addition rate. Reactor protection system comprises fast gravity-driven solid shut-off rods. During long-term shutdown of the reactor, moderator dumping will maintain.
Reactor Block

The reactor is housed in a concrete reactor block. The concrete wall of the reactor block provides shielding in radial direction. Shielding in axial direction is provided by two motor-operated movable shield trolleys provided at the top. The reactor block is located inside a reactor building.

The reactor tank, square box and lattice girders which support fuel and control assemblies are housed inside a cavity in the reactor block. The reactor tank accommodates the fuel, moderator and shut off rods. The tank is open at the top to facilitate connection with a square box through an elastomer expansion joint.

The square box houses the lattice girder assemblies. The bottom plate of the square box has a 3300 mm diameter opening in the centre to provide full access to the reactor tank. The top plate of square box has a 3350 mm diameter opening at the centre to allow access to the lattice girders. The opening is covered by a 25 mm thick revolving plate supported on a bearing, which in turn rests on the square box top plate. An oil seal is provided between the fixed top plate of the square box and revolving plate to confine the cover gas within the square box and to avoid ingress of atmospheric air and moisture into the heavy water contained in the reactor tank.

Lattice girder assemblies are designed to support the fuel assemblies and reactivity control devices and offer flexibility of configuring the desired core lattice, at the required pitch. The lattice girder assemblies can be spaced as desired with the help of centre zero scale provided along the side of stainless steel rail beams.

Moderator and Cover gas system

For the initial set of experiments heavy water will be used as the moderator and the reflector. Nitrogen gas is used as the cover gas for the heavy water. The system is designed to supply the required inventory of heavy water in a controlled manner to the reactor tank to attain reactor criticality for various core configurations / reactor experiments. Since the reactor is designed to operate at a very low power level, no dedicated core cooling system has been provided. The small amount of heat generated in the core is transferred to the moderator and cover gas and gets eventually dissipated into atmosphere by natural convection.
Control Instrumentation

For reliable neutronic power measurement and reactor protection, neutron detectors and associated electronics have been used with sufficient redundancy. The neutron detectors are located in the graphite fillers below the reactor tank as there is substantial difference in expected critical heights of AHWR and PHWR core configurations. For reactor start up from source range two pulse channels are provided. For power measurements and protection in intermediate & power range, six DC channels, consisting of three Log-linear channels (for providing Log-rate and Linear Power signals for manual regulation of the reactor power), two safety channels and a multi-range DC channel (having seven ranges with full scale values of 0.5 mW to 500 W), are provided. Adequate conventional instrumentation is provided to monitor and record the process parameters such as flows, pressures, temperatures, levels, etc. and to generate trips/alarms. An in-core flux mapping system consisting of an array of 25 LEU-based miniature fission counters, working in pulse mode, will provide three dimensional flux distribution in the core.
Shut-Off-Rod Head Gear under Endurance

Shutdown Devices

The fast-acting shut off rods are used as primary shutdown system. Fast shut down of the reactor on a trip signal is achieved by gravity fall of six-cadmium (Cd encased in Al) shut off rods into the core. The shut-down system is designed to provide sufficient negative reactivity insertion reliably, for fast reduction of reactor power to render the reactor sub-critical following a reactor trip. The shut off rods are normally parked above the core region. The total worth of the shut off rods is about 68 mk and can vary depending on the core configuration. Shut off rods can be withdrawn from the core, one at a time in a predefined sequence, only when the trip is reset, and are parked in their normal parking positions. On a reactor trip signal, in addition to the insertion of the shut-off rods, moderator is also dumped from the reactor tank up to a predefined level by opening two fast-acting dump valves to provide adequate shut down margin, independent of the shut off rods, in the most reactive core.

Experimental studies

Initially, three types of cores using heavy water as moderator and reflector are planned to be studied.

The AHWR representative core of the Critical Facility is a variant of the reference core, wherein, initially the central nine natural uranium clusters of the reference core will be replaced by 54 pin (Th-Pu)/(Th-233U) oxide clusters to simulate the initial and equilibrium AHWR core. Eventually the core will be made critical with (Th-Pu)/(Th-U233) oxide clusters alone. The central cluster in the representative core is expected to have the spectrum very close to actual AHWR spectrum.

In order to facilitate extensive measurements, the central cluster is designed for removal of fuel pins from the cluster, placement and retrieval of foil detectors from the removable fuel pins and features to fill/ drain the desired fluid into/ from the cluster for experiments with different coolants as required.

In critical facility, experiments are done in cold, clean conditions. Thus only simulation models and nuclear data are validated in these conditions directly. As the temperature of moderator in CF cannot be raised above 60°C, it will be difficult to measure fuel temperature coefficient. Possibility of carrying out these measurements by electrically heating central cluster is underway. Some typical measurements to be carried out are listed below:

Experiments planned for AHWR

- Critical height and level coefficient measurement for various lattice pitches involving both (Pu-Th) oxide and composite

<table>
<thead>
<tr>
<th>Lattice Core (RLC)</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reference</td>
<td>19 rod cluster of Natural Uranium</td>
</tr>
<tr>
<td></td>
<td>61 lattice locations (6 for SORs)</td>
</tr>
<tr>
<td></td>
<td>245 mm pitch</td>
</tr>
<tr>
<td>AHWR</td>
<td>54 pin AHWR clusters in Central 9 positions of RLC245 mm pitch</td>
</tr>
<tr>
<td>PHWR 540MWe</td>
<td>Six bundles of 37 pin natural UO2 fuel per channel</td>
</tr>
<tr>
<td></td>
<td>69 lattice locations (6 for SORs)</td>
</tr>
<tr>
<td></td>
<td>286 mm pitch</td>
</tr>
</tbody>
</table>
- Measurement of flux profile, reaction rates and the ring power factor inside the removable fuel pins of experimental fuel cluster.
- Measurement of Dynamic and static worth of reactivity devices
- Lattice cell parameters
- Assessment of coolant void reactivity
- Measurement of Worth of Dysprosium burnable poison in ThO₂ & (Th-Pu)O₂ and (Th-²³³U)O₂ in AHWR clusters.
- Reaction rate and initial conversion ratio measurements
- The flux distribution measurements with Cu/Au wires and foils
- On-line measurements by distributed set of Fission Chambers.
- Spectrum measurements by Lutetium and other activation detectors – High Purity foil materials (Cu, Au) for thermal, with Cd cover for epithermal and threshold detectors like In, Ni, Fe, Ti, Zn etc. ²⁷¹Np coated foils to be used with SSNTD for fast neutron measurements
- Special fuel foils for measuring various reaction rates.
- U foils of different ²³₅U/²³₈U contents
- Pu foils with different ²³⁹Pu/²⁴⁰Pu contents
- ²³³U foils with different ²³³U/²³⁴U content.

For experiments in the reference core, the metallic uranium cluster at the centre of the core has fuel pins with activation detector foils. Five fuel pins in this cluster are of removable type to facilitate insertion and removal of the activation detectors. Each removable fuel pin has six flux measuring foils. The flux measuring foil is clad in Al/Cd.

Fuel design for Critical Facility

The lattice physics experiments will be carried out in three phases, i.e., Reference core, AHWR representative core and PHWR core.

Natural metallic uranium cluster for reference core

The reference core will contain natural metallic uranium clusters at 55 lattice locations arranged at a square pitch of 245 mm. The metallic uranium cluster will be hung from the top of girders through an extension assembly consisting of a top adaptor and an extension rod. The cluster consists of 19 pins of uranium clad in aluminium arranged in two rings of six and twelve pins around a central pin. They are held between top and bottom tie plates, which are welded to the aluminium fuel tube. The fuel cluster will not be in contact with coolant. The top tie plate has provision for attachment of the extension rod through pins. All the fuel pins required for natural metallic uranium clusters have been fabricated for reference core of critical facility. Thirty eight assemblies have been made.

Experimental metallic uranium cluster

For reactor physics experiments in the reference core of critical facility, the metallic uranium cluster at the centre of the core will be of removable pin type for placing activation detectors in it. The cluster has provisions for the following:

- Five pins with provision for placing activation detector foils
- Foils wrapped around these pins at select locations
- Ø1.0 mm wire inside the cluster along its complete length
- Ø1.0 mm radial wire as chords, at three elevations.
Experimental Thoria Cluster

One thoria cluster containing 19 thorium dioxide pins will be loaded at the central lattice position of the reference core. Reactor physics measurements, including reaction rate, flux measurement, level coefficients, void reactivity etc. will be made with this experimental thoria cluster. The cluster will have provisions for placing activation detectors.
AHWR Representative Core

The AHWR representative core will be constituted by replacing central 3 x 3 array of natural metallic uranium clusters by AHWR fuel clusters. They will be hung from the top in a similar way to the reference core clusters. The experiments will be carried out with sets of clusters – 1st set consisting of nine all (Th-Pu) MOX cluster and 2nd set of nine clusters containing (Th-Pu)MOX and (Th-233U)MOX pins. One cluster from each of the sets will be an experimental cluster with features of removable pins and removable dysprosium assembly (central rod).

The design of all (Th-Pu) MOX cluster for AHWR-CF (AHWR fuel cluster for Critical facility) has been finalised. The design has been made amenable for remote assembly. A dummy cluster of AHWR-CF fuel made up of SS components has been fabricated. This cluster will be used to demonstrate the assembly and testing of the handling tools (for pins and for cluster as a whole). Short length solid SS bars are used to simulate fuel pins but other components like tie-plates, closure plug etc. are of full scale. Removal type of central rod has been made and will also be used for tool testing.

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- Design and Development of Shut-off Rod Drive Mechanism

The shut-off rod drive mechanism is designed with several advanced features. However, it is the requirement of designing drive mechanism within the given space constraints which makes its design a custom built. As the mechanism is custom built, its design is qualified and proven for reactor use through prototyping and subjecting it to life-cycle testing on full-scale test facility.

The prototype drive mechanism has been tested for more than 5000 cycles on full-scale test facility. Design basis report has been
Advanced Heavy Water Reactor

Gamma scanner to validate design parameters of low irradiated fuel pin of advanced heavy water reactor in the critical facility

A fuel pin Gamma scanner has been developed to validate physics design parameters of fuel pin for the Advanced Heavy Water Reactor in the Critical Facility. It is a fully automated, computer-interfaced radiation measurement system for scanning the irradiated fuel pin by a pair of Sodium Iodide (NaI) detectors and one Germanium detector (HPGe), for collecting collimated gamma spectra in three directions. The data generated will facilitate the optimization of the fuel pin design.

The system is designed to impart cyclic linear pull and rotate the fuel pin through predetermined angle and supersedes other engineering techniques using lead screws, cams etc. which are not viable for use in conjunction with long fuel pins. The device technique developed and validated as shown in the figure confirms with a fitting solution in particular holding the fuel pin hollow cluster to a predetermined non-superfluous force.

The gamma scanner developed is primarily suitable for low irradiated fuel pins. At this stage conceptualization and fabrication aspects of basic devices of gamma scanner like development of twin chuck, it’s up-gradation for adaptability to scan different fuel pins, computer interface, cyclic device sequence programming and its automated operations are validated and standardized. In future, this technology will be implemented in the following areas.

- Development of compact gamma scanning mechanisms in conjunction with counterweight techniques for underwater gamma scanning of irradiated fuel elements.
- Development of portable mechanisms to facilitate in situ Gamma Scanning of fuel pins encased in a heavy cast.
- Measurement of inside diameters of clusters of long lengths.
- Development of heavy-duty compact gamma scanning mechanisms in conjunction with counter weight techniques to scan fuel elements up to 100 mm in dia and 5 meters in length and weighing approximately 50 kg.

completed and also reviewed for safety qualification. Presently, nine drive units are under advanced stage of manufacture.

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1.7 DESIGN OF COMPONENTS AND PROCESS SYSTEMS OF THE AHWR

- Design of Process & Safety Systems of the AHWR

Process design of major reactor systems in the AHWR nuclear island and steam cycle involves finalisation of process parameters, design of process equipment, analysis for system behaviour under various conditions (normal, upset, accident, faulted), preparation of flow sheets, Piping and Instrument Diagrams (P&ID), system & equipment layout.

The Design Basis Reports (DBRs) and Detailed Project Report (DPR) of process and safety systems of 920 MW, AHWR have been prepared. Peer review of detailed project report of AHWR has been completed by NPCIL and review comments have been incorporated in the design.

- Main Heat Transport System

The Main Heat Transport (MHT) System of AHWR transports heat from fuel rods to steam drum. AHWR uses boiling light water as the coolant in the high-pressure MHT system and heavy water as the moderator in a low-pressure system. The coolant recirculation in the primary system is achieved by two-phase natural circulation, which depends on the density difference between the hot and cold legs of the primary loop. This mode of core cooling is adopted not only during normal operation but also during transients and accidental conditions.

The MHT system consists of a common reactor inlet header from which 452 inlet feeders branch out to an equal number of fuel channels in the core. The outlets from the fuel channels are connected to tail pipes, 113 of which are connected to each of the four steam drums. From each steam drum, four down comer pipes are connected to the common inlet header.

During normal operating conditions, the steam drum pressure is maintained at 7 MPa. The water level in the steam drum at nominal operating conditions is 2.2 m. The primary loop circulation rate maintained by the density difference is
approximately 2237 kg/s at nominal operating conditions. The average core exit quality is about 18.7 % for the rated reactor operating conditions.

The two-phase mixture leaving the core is separated into steam and water in the steam drum. The steam-water separation in the AHWR steam drum is achieved naturally by gravity without the use of mechanical separators. During hot shut down condition, the decay heat is removed using main condenser. However, in case of non-availability of main condenser, Isolation Condensers will remove the decay heat in passive mode.

**Emergency Core Cooling System**

The Emergency Core Cooling System (ECCS) in the AHWR provides cooling to the fuel in passive mode during first fifteen minutes of LOCA by high pressure injection from Advanced Accumulator and later for three days from the Gravity Driven Water Pool. The advanced accumulator is designed to provide a large amount of cold water directly into the core in the early stage of LOCA and then switches passively to inject small amount for a longer duration to quench the core. This passive switching is achieved by the Fluidic Flow Control Device of the advanced accumulator.

The system consists of four accumulators containing 240 m³ of water pressurised by nitrogen at 5.0 MPa pressure. During the event of large pipe rupture i.e., the inlet header rupture, MHT pressure falls due to blow down, and cold water from the accumulators enters the reactor core. The high pressure injection lines are provided with rupture discs, check valves and isolation valves. The low pressure injection from GDWP initiates after isolation of accumulators by closing the valves.
Gravity Driven Water Pool

The Gravity Driven Water Pool (GDWP) functions as a heat sink for Passive Concrete Cooling System, condenses the steam flowing through the isolation condensers during reactor shutdown and sources low pressure Emergency Core Cooling (ECC) injection for removing decay heat for three days following LOCA. The GDWP also functions as a suppression pool to cool the steam and air mixture during LOCA.

The GDWP is located in the dome region of the reactor building and contains 6000 m³ of water inventory. Stainless steel lining of 6 mm thickness is provided inside the GDWP. There are eight compartments in the GDWP which are interconnected to each other. Each compartment of GDWP contains one Isolation Condenser for core decay heat removal during shutdown and is provided with outlet nozzles at various elevations for flow of water into the core during LOCA. The Passive Containment Coolers are located below the GDWP for containment heat removal during and after LOCA.

The GDWP recirculation and cooling system consists of four loops. Each loop consists of a heat exchanger, a pump, a filter, an ion exchanger and a chemical addition tank for maintaining water chemistry. These loops operate in rotation with two of them operating at a time. A bypass purification line, consisting of a filter and an ion exchanger, is provided to remove the suspended and ionic impurities so as to maintain the crud concentration within limits.

To prevent biological growth in GDWP, its water will be continuously recirculated by the re-circulating pumps. The recirculation flow will be maintained by two pumps operating at a time on rotation basis. The GDWP re-circulation system takes care of re-circulating, filling and draining the water in each compartment. Each compartment has a separate inlet and outlet line for supply and drain of pool water respectively.
**Important Design Features of Process & Safety Systems to Improve Plant Performance & Safety**

**Moderator heat utilisation**

Heat generated in heavy water moderator and heat transferred from fuel channel to moderator is generally transferred to process water in circulation loop, which goes as waste heat. In AHWR, the moderator system is designed to utilize the optimum quantity of heat to improve the thermal power output at Turbine Generator (TG). Out of 58 MWt moderator heat, 36 MWt is utilised to heat the condensate feed water before the LP heaters in steam cycle. With heat utilization the condensate is heated from 44ºC to 68ºC, resulting in saving of steam extraction requirement of 51.72 T/h from LP Turbine and thereby generating 3.42 MWe power.

The moderator outlet temperature is increased to utilize the maximum heat. The heat utilization is optimized keeping the sufficient time margin (~3-4 h) for moderator to reach boiling temperature during station black out condition and also based on the limitations like size of the heat exchanger and temperature cross.

The economic benefit in terms of electric power gain is considerable compared to capital cost involved in additional to and fro feed water piping from turbine building to reactor building, additional heat exchangers and valves. The efficiency of steam cycle is 33.4% after utilisation of moderator heat.

**Passive Moderator Cooling System during Station Black Out**

Passive moderator cooling is designed to avoid boiling of moderator and to avoid release of tritium from moderator cover gas due to pressure rise in the event of Station Black Out (SBO). Feasibility study is completed and detailed design is under progress.

The passive moderator cooling system removes 2 MW moderator heat by a heat exchanger suitably elevated from the heat source center i.e. core. Both tube side heavy water
moderator and shell side Gravity Driven Water Pool (GDWP) water circulate in the heat exchanger by buoyancy force. GDWP with 6000 m³ water inventory serves as heat sink. Heat exchanger and the connecting piping are designed to minimise the pressure drop to achieve the required flow by natural circulation.

During Station Black Out (SBO) condition due to unavailability of forced circulation & cooling, moderator temperature starts increasing due to heat generation in moderator and heat transfer from hot main heat transport system (MHT) at 285°C. MHT temperature remains hot during station blackout without any crash cooling provision while Isolation Condensers (ICs) removing core decay heat by passive valves in operation. Analytical estimation shows a moderator bulk mean temperature of 90°C in 1 h 45 min and cover gas pressure reaches relief valve set pressure of 1.5 kg/cm² (abs.). The temperature further rises to 100°C in three hours and continues to rise with time in the absence of cooling provision. With the passive cooling provision and 2 MW moderator heat removal, the cover gas pressure and moderator temperature of the system is maintained below the relief pressure setting and saturation temperature.

Passive concrete cooling system protects the reactor V1 volume concrete surface from high temperature MHT, by natural circulation of water in tubes near the concrete structure. This passive concrete cooling system has eliminated blowers and associated power supply failure events. The concrete temperature is maintained below 55°C by the optimum number of cooling pipelines, over the conventional cabinet insulation panel surrounding MHT. The natural circulation of water from gravity driven water pool, in the pipes, maintains the concrete temperature. Corrugated plate outside the insulation panel gives a better heat transfer contact with the cooling pipes. Water circulation stabilizes due to density difference between the cold supply line connecting GDWP from outside V1 volume and cooling pipes, which picks up heat transferred through insulation panel.

Passive concrete cooling system design is completed and experimental facility is being planned to validate the analytical results.

Passive Shutdown Device

The Passive Shutdown Device (PSD) is an additional provision to effect shutdown in the event of over pressure in the MHT system due to the failure of wired mechanical ShutDown System (SDS-1) and liquid poison injection in moderator (SDS-2) due to any maleficient activity. PSD adds liquid poison to moderator from a pressurized poison tank, by actuating a valve passively driven by steam pressure. A rupture disc is provided between the passive valve and the steam drum for releasing the steam to valve by rupture action at over pressure above the trip set pressures for SDS # 1 or SDS # 2.
Advanced Heavy Water Reactor

The purification system cooling circuit is modified to utilise the optimum quantity of waste heat of 5 MWth to produce 300 m³/day Demineralised water by providing an isolation heat exchanger, which transfers heat from 100°C to 64 °C to Low Temperature Evaporation (LTE) Desalination Plant. This design feature lowers the heat load on process water besides producing 300 m³/day D.M. water.

**Thermal Analysis for Estimation of Rate of MHT Temperature Rise**

The heating of Main Heat Transport (MHT) system from 35°C to 285°C during start up from cold shutdown condition has been studied by utilising core heat from 2% to 5% Full Power (FP), after attaining criticality. Analytical estimates shows that the heating time varies from 5 hrs to 26 hrs for core heat varying from 2% to 5% FP and purification flow varying from 0 to 100%. These analytical estimates consider heat capacities of structural, fuel and water inventory, losses to surrounding moderator, top and bottom end shield, air surrounding MHT piping in V1 area.

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- **Design of Coolant Channel of the Advanced Heavy Water Reactor**

Coolant channel forms the heart of pressure tube type nuclear reactor. Coolant channel houses the fuel assembly along with shields and seal plugs. The coolant flows past the fuel assembly removing the nuclear heat from the fuel pins.
The AHWR coolant channel has many features which facilitates on-power fuelling, direct injection of cold water from emergency core cooling system to hot fuel pins in the event of LOCA, easy replaceability of pressure tube, suitable interfaces with Main Heat Transport (MHT) system, as well as annular gap to separate the hot pressure tube (285°C) from surrounding cold moderator (65°C).

**Design Description**

Coolant channel consists of cold worked Zr-2.5Nb pressure tube in the reactor core region, which is extended by stainless steel end fittings at the top and the bottom. The coolant channel is located and locked in its position by anti-torque hardware and yoke assembly attached to top and bottom end fittings respectively. Bottom end fittings are provided with feeder couplings to connect it with feeders, which are welded to Inlet Header. Lower ends of tail pipes are welded to top end fittings and top ends of tail pipes are connected to steam drum.

Fuel assembly is located inside the pressure tube and is inserted and taken out of the channel through the top end fitting. The top end fitting has suitable features to support and locate the fuel assembly along with the shield plugs in its designed location and to interface with the fuelling machine to carry out fuelling operations. After fuelling is over, the top end fitting is closed using a seal plug which butts against a seal face provided in the bore of the end fitting.
Operational Details

Light water coolant from MHT System enters the coolant channels through bottom end fittings, which are connected to feeders. Coolant at 259°C flows past the fuel assembly by removing the heat generated by nuclear reaction and flows out as steam-water mixture at 285°C through the tail pipe to the steam drums where steam is separated and sent to turbine.

Easy Replaceability of Pressure Tube

The estimated life of pressure tube in this reactor is about 30 years due to various degrading mechanisms such as irradiation creep, corrosion and hydrogen pick up. Easier replaceability of Pressure tube is taken into consideration in the design to reduce the duration of shut down of the reactor, man-rem exposure and the cost of replacement. Certain features are provided in its design for the above purpose and are described below.

Pressure tube

Replacement of pressure tube has been planned by removing it along with bottom end fitting, through the bore of the top end fitting. This has been achieved by keeping the bore of the top end fitting more than the maximum outside diameter of pressure tube, after considering the irradiation induced diametrical creep of 4% and the outside diameter of the bottom end fitting. In view of this and the rolled joint requirements, pressure tube is designed to have an outside diameter of 133 mm and thickness of 6.1 mm at the top end and 90 mm outside diameter at the bottom end.

Rolled Joint

Pressure tube is joined to top end fitting with a rolled joint, which can be assembled and tested remotely. The rolled joint can be detached by shock heat method. Top end fitting which is not planned to be removed during pressure tube replacement, is provided with an additional set of rolled joint grooves which could be used for making the second joint.

Feeder Coupling

Feeders are connected to the bottom end fittings by feeder couplings. This is a special compact mechanical coupling with a self-energising metal seal as sealing element which requires lower tightening load than other mechanical couplings. The effectiveness of sealing increases with pressure in this coupling. The bottom end of the bottom end fitting is provided with an in-built hub and the feeder is also ending with a suitable hub. These hubs are provided with a taper on its outer portion and the inner side of clamp is provided with a matching taper. When the clamp is tightened over the hubs, they become closer and the required amount of tightening load is applied on the sealing element. The feeder coupling has been designed and developed on the basis of performance tests.
Composite Sleeve rolled Joint

The tail pipe is of SS304L and is required to be welded to top end fitting. Hence, it is preferable to have top end fitting of SS304L in order to avoid a dissimilar metal welding in the nozzle region. Due to large difference in thermal expansion coefficients of SS304L and pressure tube material, a rolled joint between pressure tube and SS304L end fitting operating at high temperature is not feasible. Hence, a composite sleeve is designed consisting of outer sleeve of SS403 material which is having lower thermal expansion coefficient, shrink fitted over inner sleeve of SS304L material. At higher temperatures, outer sleeve will introduce compressive stresses on the inner sleeve, which will partially compensate for the relaxation of contact pressure, between the pressure tube and the inner sleeve.

Annulus Leak Monitoring System

The Annulus Leak Monitoring System (ALMS) is incorporated in the design of the coolant channel to ensure ‘Leak Before Break Criterion’. It is designed to detect leaks from pressure tube, its joints and from moderator system into the annulus between pressure tube and calandria tube sufficiently in advance so that the reactor can be safely shut down before any catastrophic failure occurs. The annulus is closed at the top end and open at the bottom.

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- Development of rolled joint detachment technology

One of the major design features of AHWR is the replaceability of pressure tube during the lifetime of the reactor. In order to demonstrate this aspect, a technique for detachment of rolled joint between end fitting and pressure tube is being pursued. The technique is based on radio frequency induction heating, in which the pressure tube in the rolled joint area is rapidly heated using an induction coil and then suddenly cooled. A schematic of the technique is shown in the figure. During heating, as the pressure tube temperature starts increasing and the stress developed reaches the yield stress, there will be no further increase of the contact pressure between pressure tube and end fitting. During subsequent cooling, a gap will be generated between the end fitting and pressure tube. After two to four cycles, the joint will become free. During subsequent heating, an axial load is applied, which will contract the pressure tube radially and finally it will come out of the rolling grooves. Variation of contact pressure with temperature...
in the case of 220 MWe PHWR calandria tube rolled joint, as estimated analytically is given in the Figure.

In order to estimate the various parameters involved and to optimize their values, a mock up facility is being set up. To reduce transmission loss, coaxial cable is being purchased, which will be used to transfer power from the induction-heating machine to the work coil. In order to apply axial load during detachment work, a tool has been designed, as illustrated in figure.

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Containment Isolation Devices for Nuclear Reactors

The peripheral RCC wall of the nuclear reactor is termed as containment and is the ultimate barrier between the reactor and the atmosphere. Utilities are fed in and out of the reactor through pipe penetrations. The largest such penetrations pertain to ventilation system. The dampers provided for closing these penetrations and bottling up of the containment are thus critical equipment of the dynamic containment system. Dampers provided in such penetrations are required to close in a perfectly leak-tight manner within few seconds in case of any abnormal situation. The safety and reliability of entire containment system depends mainly on these dampers.

Most of the existing systems utilize butterfly valves for this purpose. These butterfly valves are basically designed for fluid duty at a system pressure of 5 kg/cm² whereas in reactors the pressure of not more than 2 kg/cm² is expected even under severe accidental condition like LOCA. When these valves are used for low pressure systems handling atmospheric air the performance does not remain satisfactory. Unequal seal compression, unequal wearing of seal, different behavior of seal material in water and air, inadequate deflection and sealing force due to low pressure, accumulation of dust on sealing surface aggravating the seal compression etc. are the problems being faced in butterfly valves.

Hence there was a need to design these isolation devices specifically for the intended application instead of just trying to adapt available valves in an unsuitable manner. The isolation devices can be either active type i.e. through external actuation by means of an electro-pneumatic circuit or passive type i.e. self-actuating by the fluctuation in system parameters viz. pressure & velocity of air in the duct during LOCA. Work is being carried out on both the fronts.

On the active damper side an electro-pneumatically operating, lightweight, positive seal, airtight damper, with rotary cum axial motion was developed in TSD. The axial motion provides perfect sealing by overcoming inherent defects of butterfly valves. This damper has compressed air to open and spring to close on active signals. Thus it is termed as active damper. Initial design utilizes multi-four bar mechanism to achieve the rotary-cum-axial motion. Efforts are being made to reduce the number of pulling tool for detachment work.
linkages. The test results are highly encouraging.

On the passive damper side simple spring loaded disc critically designed to allow a particular range of flow is being developed. The disc automatically closes when there are instantaneous increases in air velocity in the duct.

The active damper along with passive dampers in series can provide a perfect solution to the need of positive and reliable isolation of any nuclear reactors. The major works involved are conception of design, development of prototype working model, development of testing set up, fabrication and testing prototype model, test data interpretations and subsequent design modifications. Presently work is being carried out to design, fabricate & test a damper suitable for the conditions specified in AHWR reactors and also to fabricate a damper to establish the design principles of Passive Dampers.
Advanced Heavy Water Reactor

1) DISC - 470ø WITH CEILING RING
2) SPRING HOLDING COLLAR
3) SUPPORTING COLLAR (HAVING SLOTS AT TOP SLOT LENGTH 150mm AND 4 Nos.)
4) SPRING
5) GASKETS

SECTION ‘A-A’

PASSIVE DAMPER (EXHAUST)

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1.8 FUEL HANDLING FOR AHWR

Development of Fuel Handling System of AHWR

Main requirement of fuelling system is to re-fuel the reactor by gaining access to the reactor top face during refuelling. The fuel handling system mainly consists of a fuelling machine, an inclined fuel transfer machine, a temporary fuel storage bay located inside the reactor building and a fuel storage bay located outside the reactor building. Main design challenges faced in the design and development of fuel handling system are related to the handling of 10.5 metre long fuel assembly, shielding requirement and seismic qualification of 19 meters tall and 600 tonne heavy fuelling machine.

Fuelling Machine

The function of fuelling machine is to remove and insert the fuel assembly in three pieces i.e., shield A, shield B and fuel cluster with a reliable operation. This is done with the help of separator assembly, magazine assembly and ram assembly of the fuelling machine. Magazine assembly temporarily stores the plugs and fuel and ram assembly operates the plugs for their removal, movement or installation. Ram assembly consists of three coaxial rams and outer ram travels up to 7.6 meters for removal of fuel assembly from the channel. Considering the shielding requirement and long and heavy structure of fuelling machine it has been supported at higher elevation to achieve fine alignment with the channel. Fuelling machine interacts with coolant channel through snout assembly making leak tight connection for refuelling. 3-D modelling and design has been completed for the machine. Presently manufacturing of prototype fuelling machine is in progress.

Trolley and carriage assembly support the heavy shielding structure and locate the machine coarsely to within ±3 mm of the particular channel to be refuelled.

Fuel Storage Bay

There is a small temporary fuel storage bay (TFSBx) located inside reactor building through which fuelling machine interacts for charging new fuel and receiving spent fuel. The permanent fuel storage bay is located outside the reactor building for storage and cooling.

Inclined Fuel Transfer Machine

Inclined Fuel Transfer Machine (IFTM) is used to transfer the fuel across the reactor building. IFTM is a tall machine connecting inside and outside fuel storage bays through containment walls. A water
filled pot containing fuel, guided in an inclined ramp, is hoisted up in the tilting leg and subsequently hoisted down to unload the fuel on the other side. The concept of IFTM for fuel transfer is most suitable because of less requirement of space inside reactor building, on line transfer, small containment penetrations, ensured cooling of fuel throughout the transfer and passive containment isolation features. Passive containment isolation is enabled by water head available in the inclined ramp in case of LOCA.

Fuel Handling System Hardware Components

Main hardware components include snout plug, seal plug and collet joint. Snout plug is a part of fuelling machine and provides leak tight closing end for the fuelling machine when it is not connected to any channel or fuel port. The necessary sealing is accomplished by a special mechanism using radial ‘O’ ring. Skinner seal, a metallic face seal, ensures a leak tight connection between end fitting and the fuelling machine during refuelling. Seal plug is a critical sub assembly of the coolant channel assembly and acts as a pressure boundary to prevent the escape of MHT steam/water. Collet joint has been used for joining different components of fuel assembly. All the above components have been tested in reactor-simulated conditions and design has been proven.

1.9 ANALYTICAL STUDIES AND EXPERIMENTAL VALIDATION

Investigations on the stability characteristics of the Advanced Heavy Water Reactor

The Advanced Heavy Water Reactor adopts natural circulation for removal of fission heat during start-up, power-raising and accidental conditions in addition to the rated full power operating condition. With several parallel boiling channels having different power and resistances connected between the header and the steam drum with very long feeder and tail pipes, the reactor may experience various types of instabilities during its operation from atmospheric to rated pressure and the power raising process.

Occurrence of the thermo-hydraulic instabilities may further induce power oscillations through the void reactivity coupling. Instabilities of any form are undesirable from the viewpoint of reactor operation, control and safety. It is required to predict the stable and unstable regions of the reactor operations during the design stage so that if instability is found, methods of suppressing or procedures to avoid them can be worked out. Further to this, it is also required to generate the stability maps considering the neutronic feed back effects at various conditions, which are useful for the design and operation of the reactor.
Modelling the natural circulation static and dynamic instability characteristics of the AHWR have been carried out using two computer codes TINFLO-A and TINFLO-S. These codes have been developed to predict the static instabilities such as Ledinegg type and the density-wave instabilities of the reactor respectively. Computer code TINFLO-S can simulate the interactions of several parallel boiling channels of the reactor to induce the out-of-phase instability in the reactor when operated under natural circulation conditions. To analyse the flow pattern transition instability in the reactor, the model considers the flow pattern transition criteria and flow pattern specific pressure drop models both in single and two-phase regions of the horizontal and vertical sections of the reactor.

Simplified models for coupling of neutron kinetics with thermal hydraulics in the AHWR have been developed. To study the interaction between different parts of the core through neutron diffusion, a coupled multipoint kinetics model has been applied in place of simple point kinetics model for the neutron dynamics. The model considers the reactor to contain ‘N’ number of sub-cores which are sub-critical, isolated by reflectors and influenced each other only through leakage neutrons number of which is proportional to the average neutron flux over each subcore. Each subcore may contain one channel or group of channels having the same power and resistances.

In the recent past, the out-of-phase instability is explained as a phenomenon in which the neutron higher modes are excited by the thermal hydraulic feed back effects. The higher modes are all sub-critical, which could result in out-of-phase oscillations depending on the sub-criticality of the harmonic mode and the void reactivity feedback. To simulate the out-of-phase instability in the AHWR core, a mathematical model has been developed based on the modal point kinetics model. The out-of-phase instability behaviour of the AHWR considering the coupled multi-point kinetics model and modal point kinetics model has been compared.

Influence of void reactivity feedback and fuel time constant on the thermal hydraulic stability behaviour of the AHWR has been analysed. Effect of delayed neutrons on the reactor stability has also been analysed. Constant decay ratio lines, which are indications of the stability margin of the reactor, were predicted at the rated pressure conditions of the reactor.

The results shown in figure indicate that the out-of-phase mode oscillations are more dominant as compared to the in-phase mode oscillations in the reactor because of the extra single phase friction in the down comers which stabilise the in-phase mode oscillations.

Both Type I and Type II instabilities were found to be dependent on the operating conditions such as the heat generation rate and subcooling. Type I instabilities occur at low power with initiation of boiling in the core when the quality is low. Under low critical, which could result in out-of-phase oscillations depending on the sub-criticality of the harmonic mode and the void reactivity feed back.
quality conditions, a slight change in quality due to any disturbance can cause a large change in void fraction and consequently in the driving head to induce oscillations. Whereas Type II instabilities occur at high power conditions when the quality is high. The two-phase frictional pressure loss may be high owing to the smaller two-phase mixture density. Having a large void fraction will increase the void propagation time delay in the two-phase region of the system. Under these conditions, any small fluctuation in flow can cause a larger fluctuation of the two-phase frictional pressure loss due to fluctuation of density and flow to induce the oscillations in the system.

The Type I instabilities is of concern for the AHWR operation, especially during the power raising process, start-up, set-back and shut down conditions. However, since the heat generation rate is less, it may not cause occurrence of CHF in the channels. However, the reactor operation at these conditions may not be possible and to avoid them, a suitable operation procedure needs to be worked out. On the other hand, the Type II instabilities occur at much higher power, beyond the reactor trip set point. Hence, it is not of concern for the AHWR operation.

The frequency of oscillations at different operating conditions are estimated and also shown in the same figure. It is observed that the frequency of Type II instabilities is much larger than that for Type I instability. Moreover, the frequency of oscillation is very less in the AHWR (<0.08 Hz) as compared to that normally observed in vessel type BWRs (>0.5 Hz) due to very long feeder pipes in the AHWR main heat transport system. Due to the low frequency of oscillations, the Type I and Type II thermal hydraulic instabilities can get suppressed with the negative void reactivity feedback inherent with the neutron characteristics of the core. To study the neutron field dynamics, a coupled multipoint kinetics model is used.

The codes developed are currently validated with in-house data and other commercial codes like RELAP5. As a future course of this activity, four equation drift flux model of the AHWR configuration will be used to assess its stability and compared with the homogeneous model. Stability behaviour of the AHWR considering carry-under will form part of the simulation studies.

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Simulations of AHWR start up procedure

The instabilities during the reactor start up from low-pressure, can be overcome by starting the reactor at higher pressures, by externally pressurizing the system. This can be achieved by introducing steam from external boiler into the steam drum. The start-up procedure for AHWR consists of stage-wise system pressurization up to 70 bar, by means of an external boiler and system heat-up with 2% reactor power. This has been simulated using a two-fluid, non-homogenous thermo-hydraulic computer code RELAP5/MOD3.2, widely used for reactor transient analysis.

Simulations have confirmed the stability of single-phase as well as two-phase natural circulation during the reactor start-up, with stage-wise external pressurization and low power heat up. This scheme requires an external boiler with a pressure rating of 70 bar. Further start-up case studies at lower system pressure have also been investigated. The results of these simulations are discussed below.

Variation of core flow during reactor start-up with an external stage-wise pressurization

Pressurization is carried out in such a way that cold pressurization limits for structural components is not exceeded. In this scheme
the boiling inception takes place at 70 bar and stable two-phase natural circulation is achieved.

**Start-up procedure with onetime external pressurization of system at 10 bar**

At lower pressure due to the static head reduction at the steam drum flashing takes place as indicated in the figure by void fraction in steam drum and at the core exit. It is then followed by the boiling inception in the core. Flashing at 10 bar however induces smaller amplitude oscillation but can be severe at further lower pressures. After boiling inception the system is closed by closing the boiler valve and with internal steam production system is pressurized up to 70 bar. Flow is found to be stable during two-phase natural circulation.

**Reactor start-up with stage-wise external pressurisation and heat up.**

**Reactor start-up from atmospheric pressure**

Flashing instability occurs with significant oscillation amplitudes. It is then observed to reduce at increasing pressures. Low-pressure (atmospheric) natural circulation start-up experiments conducted in High-pressure Natural Circulation Loop (HPNCL) indicated that the instabilities are associated with boiling inception. The heater power in this experiment has been kept at 20 kW. The natural circulation is found to become stable at higher pressures.
Reactor start-up without external pressurization (from atmospheric pressure)

Experimental results for low-pressure start-up in HPNCL.

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**Flow pattern transition instability loop**

Instabilities due to flow pattern transition in natural circulation reactors are reported to occur while operating near the slug flow to annular flow transition regions. The primary reason for this instability is that the frictional pressure loss in slug flow is more than that in annular flow. Therefore, while operating near the slug to annular flow transition condition, a slight increase in power leads to annular flow. Due to the lower pressure drop in annular flow, the flow rate tends to increase. The increased flow rate reduces steam production and hence the flow reverts back to the slug flow regime. Now due to the larger frictional pressure drop, flow reduces causing more steam production and the flow switches to annular regime once again. The process repeats itself and is known as flow pattern transition instability. The main objectives of the experiments were to generate data for

- bubbly flow to slug flow transition,
- slug flow to annular flow transition,
- void fraction and
- flow pattern specific pressure drop.

Experiments on steady state and stability behavior of two-phase natural circulation have been completed in four loops differing in diameter. The experiments have also generated data on the void fraction, pressure drop and flow patterns. Also, it has helped in the development and testing of the Electrical Conductance Probe (ECP) which is developed in-house, its performance has been compared with neutron radiography.

The steady state analysis shows a close agreement with the theoretical results. Void fraction assessment with the various correlations has yielded in identifying few correlations which can be used for low mass flux systems such as natural circulation systems. A photograph of the facility as it was erected in the Apsara reactor hall and few typical experimental results are illustrated.

The two phase natural circulation experiments performed in the APSARA loop (1/2” diameter) have been simulated using the two-fluid, non-homogeneous computer code RELAP5/MOD3.2 to find out the mass flow rate, void fraction, test section exit quality and pressure drop. The results have been compared with the experimental measurements. The comparison of experimental results with those that are calculated by RELAP5/MOD3.2.
It is essential to know the flow rate to establish the heat transport capability of natural circulation loops. A large number of scaling parameters are available in the literature. But practically it is very difficult to simulate all the given parameters between prototype and model. Another problem associated with the existing scaling laws are that they do not give the steady state flow rate directly whereas all of the parameters generally are dependent upon the flow rate. A generalized flow correlation is needed to simulate the steady state behavior with a single non-dimensional parameter.

A set of homogeneous Navier-Stoke equations have been solved to derive the correlation for Reynolds number (inertia force/viscous force) in terms of modified Grashof number, $Gr_m$ and contribution of loop geometry towards Friction number (effective loss coefficient for the entire loop), $N_u$. This correlation is valid for both uniform as well as non-uniform diameter loops.

To account for the density variation in the buoyancy term, a new parameter $\beta_h$, which is the volumetric thermal expansion co-efficient and defined as $\beta_h = \frac{1}{\nu_m} \left( \frac{d\rho}{dh} \right)_{p}$ as been used, where, $\nu_m$ is mean specific volume and $h$ is the enthalpy at that pressure and quality.

Experiments were conducted in three loops of inside diameter 10.21 mm, 15.74 mm and 19.86 mm respectively in a facility having the geometry as in Figure. The steam separator, the condenser and the associated piping (the portion inside the rectangular box in Figure.) were the same for all the loops. In addition, experimental data were generated in a 49.3 mm inside diameter High Pressure Natural Circulation Loop (HPNCL) shown in Figure. The developed correlation was tested against the data (generated with different loop diameter: 9.6-49.3 mm) and it was seen that there is a reasonable agreement (with an error bound of $\pm$ 40%) with the proposed correlation shown in Figure confirming the validity of the correlation.
Advanced Heavy Water Reactor

BARC HIGHLIGHTS

Reactor Technology & Engineering

Comparison of theoretical and experimental results

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- Integral Test Loop (ITL)

ITL is a scaled facility which simulates the main heat transport system (MHTS), Emergency Core cooling System (ECCS), Isolation Condenser System (ICS) along with the associated controls of the AHWR. The scaling philosophy for the ITL facility is based on the 3-level approach in which integral, local phenomena and boundary flow (mass and energy) effects are given due importance.

The integral (or global) scaling is based on the power-to-volume scaling philosophy. Care has been taken to preserve important local phenomena, which can significantly influence the integral behaviour. Typical examples are CHF (critical heat flux) in the core simulator, steam-water separation in the steam drum etc. Important boundary flow effects simulated are that due to emergency coolant injection, feed water injection, etc. The facility is a single channel test facility simulating the full elevation, pressure and temperature of the AHWR.

- Objectives of the Integral Test Loop

  - Generation of database for the performance evaluation of the following systems in the plant environment
  - Natural circulation in the MHT loop
  - Steam separation process in the steam drum
  - Fluidic device in the advanced accumulator
  - Gravity driven cooling system
  - Isolation condenser system and
  - Active shutdown cooling system
  - Evolution of a start-up procedure, generation of database for plant transients and accident scenarios like LOCA
A 3-D layout and few photographs of the various equipment of the facility
**Scaling Philosophy**

To check the adequacy of the scaling philosophy, simulation calculations were performed on the prototype and model for the same operating conditions and the results are graphically presented.

![Comparison of predicted flow rates for AHWR and ITL](image)

![Comparison of stability maps between AHWR and ITL with single channel for in-phase mode of oscillation](image)

![Variation of flow rate with power for different pressure](image)

![Variation of flow rate and S. D. pressure with time](image)

**Pre-Test analysis**

To gain an insight into system thermal hydraulics a pre-test analysis has been carried out using non-homogeneous, non-equilibrium two fluid thermal hydraulic code RELAP5/MOD3.2. The steady state and transient (e.g. startup and power raising) performance of ITL has been predicted. The predicted steady state mass flow rate was found to increase with power. With reduction in pressure, void fraction and buoyancy force increases, which increase the mass flow rate. The start up can be associated with following types of instabilities e.g. geysering, flashing and low quality density wave instability. Typical start-up transient at 2% power (40 kW) with an initial pressurization of 20 bar is shown in the Figure. As soon as the boiling starts at 25000 s the low quality density wave instability sets in. This instability dies out as the system pressure rises. The power raising from 2% to 100% (2035 kW) at 0.5 % per second shows stable operation.
Simulation of Steady state Behaviour

Few natural circulation experiments were carried out at various powers and pressures to simulate steady state and normal operating behaviour of the system.

Simulation of Station Blackout

During class III and class IV power supply failure the reactor is shutdown following a Secondary Feed Pump (SFP) trip and Isolation condensers (ICs) are valved in to remove the decay heat in the reactor under natural circulation mode. To simulate such a scenario few experiments (station blackout simulation) have been performed. In simulation experiment following SFP trip the system power is tripped and set to follow a programmed decay power curve through a ramp generator. During this the system pressure rises due to decay heat and at set pressure the isolation condenser valve (ICQOV-4) opens and lets the steam condense and allows it to flow to steam drum. Thus IC acting as cold leg for the system is able to remove decay heat effectively and maintains the system under natural circulation mode. Few typical results of the experiments performed are illustrated.

The facility has been installed and commissioned in all respects and experiments that are planned are listed as follows

- Station blackout at various system pressures and powers
- Loss Of Coolant Accident (LOCA) simulation with various header break sizes at various system pressures and powers
- Simulation and evolution of the start-up procedure for various system pressures and sub cooling
- Natural circulation experiments at different powers, pressures and subcooling
- Performance evaluation of the Active ShutDown Cooling System (ASDCS)
- Performance evaluation of the Fluidic Flow Control Device (FFCD) of advanced accumulator.

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Investigation on Critical Power of AHWR Fuel Bundle

The Critical Heat Flux (CHF) is an important parameter, which limits the power that can be extracted from a nuclear fuel bundle. The critical power of a fuel bundle is the bundle power corresponding to the CHF condition for a given operating conditions like axial power profile, pressure, mass flux and inlet subcooling. In natural circulation based reactors like AHWR, the other constraining parameter is the thermal hydraulic instability. These two parameters are important design consideration for the safe operation of the AHWR. Investigation shows that no upper instability threshold exists for high inlet subcooling. Hence, the maximum operating power is still limited by CHF. Therefore CHF in the bundle needs to be reliably ascertained to determine the available safety margin. Such a prediction needs to be carried out so that necessary corrective action/design modification in the bundle can be worked out to achieve acceptable thermal margin.

Suitable prediction methods such as empirical correlations, CHF look-up table and mechanistic approaches for the CHF have been assessed. A computer code has been developed to predict the critical power of AHWR bundle with CHF Look-Up Table (LUT). The Heat Balance Method (HBM) was adopted for the evaluation of critical power by LUT approach. The bundle correction factor used with LUT was modified to improve the prediction accuracy. The assessment of this approach was carried out based on the experimental data, on the rod bundle CHF data available in the literature. The comparison between the experimental data and the prediction showed that around 88.3% data was predicted with an accuracy of ±20% as shown in Figure.

The variation of the critical power with the subcooling is illustrated. The maximum operating channel power of 2.6 MW is also shown in this figure. Apart from this, the Janssen-Levy model has also been adopted to predict the Minimum Critical Heat Flux Ratio (MCHFR). A simplified mechanistic model has been employed in conjunction with the subchannel code for the prediction of CHF and the variation of film flow rate, deposition and entrainment rates are depicted. The Critical Power Ratio (critical power to the operating power) has been estimated to be 1.51 and 1.54 using mechanistic and LUT approaches respectively. The prediction reveals that there is an adequate thermal margin available in the bundle design of the uprated AHWR (920 MWth).
However, due to uncertainty in the prediction of the complex phenomena of CHF there is a need to substantiate the prediction of the thermal margin by experimental data in simulated geometrical and operating conditions of AHWR. In view of this, a CHF program has been formulated to generate experimental data on rod bundle CHF. A tubular test section for the CHF experimental set up has been installed and experiments are in progress. The fuel rod simulator for the CHF experiment has also been designed and is being fabricated. The Freon CHF data is planned to be generated and equivalent water data can be evaluated using the fluid to fluid modeling approach.

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Experimental Measurement of Pressure Drop Across Various Components of AHWR Fuel Bundle

The pressure drop is an important parameter for design and analysis of many systems and components. The driving head and the mass flux being lower in the natural circulation system than those in forced circulation systems, accurate measurement of pressure drop is essential. The fuel cluster design of AHWR with 52 rods and 54 rods were evaluated analytically as well as experimentally. The various components of the fuel bundle are the fuel rods, the bottom and top tie plates and grid type spacers. In view of the complex geometry of the flow cross section through the fuel channel it becomes necessary to generate data on pressure drop across the fuel bundle and its components experimentally.

Single phase water and two phase air-water experiments were carried out in flow test facility with 52 rods cluster (with five and six spacers) to generate pressure drop data for the bundle and its various components. The total pressure drop across the bundle with five spacers was found be 3-6% less than that with six spacers.

The single phase variation of fuel friction factor i.e., the resistance offered by the wall surface to the flow, the variation of loss coefficient i.e., resistance to flow due to shape or flow geometry change across tie-plates and spacer are shown in the illustration. The experiments were also carried out for Reynolds Numbers in the range of 7900 to 79000 with 54 rods cluster with six spacers for single phase and two-phase.

A two-phase multiplier ($\Phi_{2\text{LO}}$) for 54 rods fuel cluster was calculated by taking the ratio of two phase pressure drop and single phase pressure drop across the bundle. A correlation is derived using the experimental two phase multiplier data. The present correlation was able to predict the experimental data with an error band of 20%.

Single-phase pressure drop experiments have been carried out in the 3 MW Boiling Water Loop (BWL) for 52 rods cluster. The components like shield plug, collet and rod bundle were fabricated and installed in the coolant channel test section in 3 MW BWL. Single phase pressure data have been generated and analysed. A typical result for the pressure drop across shield plug of AHWR coolant channel is presented below. It can be seen that the shield plug loss coefficients reduces exponentially with the Reynolds number.
Advanced Heavy Water Reactor

Schematic of Flow Test Facility

54-Rod Bundle Assembly and Cross-sectional View

Fuel Pin OD = 11.2 mm, Number of Pins = 54
**Advanced Heavy Water Reactor**

**Variation fuel friction factor (single-phase)**

**Variation of loss coefficient across tie plates (single phase)**

**Variation of loss coefficient across spacer (single-phase)**

**Comparison of two-phase friction multipliers**

**Experimental set up for the two-phase pressure drop experiments in the coolant channel of AHWR**

**Loss coefficient of the shield plug of AHWR**

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AHWR fuel cluster vibration studies.

The fuel bundle vibration induced by high velocity coolant flow is an important issue that needs to be addressed for ensuring its satisfactory performance in the reactor. Normally the coolant flow has the potential to excite the bundle and the fuel elements inside it. This excitation can give rise to low amplitude vibrations that are enough to cause inter element rubbing. There are many reported incidences of fret related damages to fuel bundles due to rubbing in all type of reactors. The vibration studies on AHWR fuel cluster was taken up in an experimental facility to assess the fuel cluster/element vibration.

AHWR fuel cluster is a 4.3 meter long slender structure made up of 52 fuel rods of 11.5 mm diameter. The bottom and the top of the cluster have collets and the fuel tubes are held by spacers along its length to maintain gap between the tubes and to provide guide support to the fuel rods. The cluster is housed inside a coolant channel through which the coolant flows from bottom to top. The coolant flow, which is driven by natural circulation, induces vibration in the fuel rods. In order to characterise the dynamic behaviour of the cluster, study has been carried out to assess the fuel rod and cluster vibration in a simulated flow test facility.

The set-up shows the optical and laser devices used for direct measurement of fuel vibration through an optical window. The effect of single and two-phase (air and water) flow on cluster vibration has been studied. The frequency band of flow excitation and the level of cluster and tube vibrations have been identified. The experimental study has led to the conclusion that AHWR cluster vibrations are below five microns and tube vibrations are insignificantly low. Due to these observed level of vibrations, the possibility of putting just five spacers instead of original six spacers is under review. Such a design change is expected to reduce the pressure drop in the channel, which is a desirable feature for the natural circulation loops in AHWR.

The illustration shows the vibration spectrum of the cluster and the frequency band of flow excitation for five and six spacers in the cluster. The vibration spectrums of five spacers and six spacers cluster show similarity in its contents. The cluster modes around 5 Hz and 11 Hz can be clearly seen in both the spectra. There are no other indications that could be attributed to significant fuel element vibrations in both the cases.

The fluid fluctuation spectra for both the cases also show similarity except for minor variations of little consequence.
Fuel Cluster with six spacers

Vibration Spectrum

Fluid Flow
Development of Indirectly Heated Fuel Rods

The safety of nuclear reactor is to be ensured not only under normal operating conditions, but also under transient and accident conditions. Loss Of Coolant Accident (LOCA) is one of the postulated accidents, the course of which is strongly dependent on thermal hydraulic characteristics of the reactor core, comprising of fuel rods. The complex nature of the phenomena occurring during accidents calls for extensive experimental investigations. Electrically heated Fuel Rod Simulators (FRSs) are extensively used to simulate nuclear heating (particularly decay heat) in out-of-pile experiments. The designs of FRS are mainly categorized into directly heated fuel rod simulators and indirectly heated fuel rod simulators. In directly heated type FRS, current is passed directly through the tube which geometrically simulates the cladding of a nuclear fuel rod. This type of FRS finds application in experiments related to steady state heat transfer in single and two-phase flow of fluids, which are non-conductors of electricity. Very high heat flux can be achieved in this type of FRS. However, for unsteady state tests, simulation of stored heat is extremely difficult in such a design. For safety related experiments, fast transients are involved, the stored energy and transfer of stored energy in fuel pay an important role. Indirectly heated FRS, which simulates stored energy better, is preferred for such applications. In indirectly heated FRS, the heating element is kept inside the clad tube and the gap is filled up with the ceramic powders compacted to a certain density to achieve high thermal conductivity of the powders.

Indirectly heated Fuel Rod Simulators (FRSs) have been developed to perform out-of-pile thermo-hydraulic experiments. Two types of FRS (one with and the other without gas gap) have been fabricated and tested. The FRS with gas gap has been tested in 3 MW Boiling Water Loop (BWL) up to 35% of AHWR fuel rod rated power while the one without gas gap has been tested up to 130% of rated power.
Fuel Rod Cluster Simulator (FRCS): Design and Development of Directly heated Fuel Rod Cluster Simulator (FRCS)

Fuel Rod Cluster Simulator is a key component of thermal hydraulic test facilities to generate data on CHF, pressure drop and subchannel mixing. In addition, they are also required in simulated integral test facilities simulating nuclear reactor systems. In view of this, a direct electrically heated 54 rod Fuel Rod Cluster Simulator (FRCS) simulating AHWR fuel bundle has been developed. In the fuel cluster electrical heating of the clad simulates nuclear heat generation. The special safety feature of AHWR, like direct in-bundle emergency coolant injection has also been simulated in this FRCS.

The FRCS requires precise fabrication of spacers and fabrication of heater tubes by joining of nickel rod and inconel tubes. A special technique has been developed to install the thermocouples and to take out lead wires penetrating the pressure boundary. Subsequent to hydro-test and thermal cycling tests, the FRCS has been installed and operated in the integral test loop at operating pressure and temperature successfully.

Fuel Rod Cluster Simulator: a) Different Components b) Installed at Integral Test Facility

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Studies on steam drum thermal hydraulics in an Air Water Loop

In the normal operating conditions of AHWR the average core exit quality is about 17.6%, which corresponds to a void fraction of 81.36%. This causes a swelling in the steam drum i.e., an increase in the steam drum level and if the reactor is tripped suddenly the water level in the steam drum may fall below the top of the baffle plates, due to sudden collapse of voids. This may disrupt the natural circulation in the main heat transport system and can lead to an increase in the clad surface temperature of the fuel. Hence, studies are required to know the exact swelling in the steam drum. The other objectives of this loop are:

- Measurement of bubble and droplet distribution and
- Investigation of (a) carry-over phenomena i.e., entrainment of water droplets in the air going out of the pool and (b) carry under phenomena i.e., entrainment of air bubbles along with the water flowing to the down comer

Experiments are carried out in an Air Water Loop (AWL) to measure the void fraction (swell) in a pool of water by the swell level. Void fraction was estimated from the measured swell level and was compared with the calculated values from Kataoka Ishii correlation and good agreement is observed between these two.

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**Natural Circulation Flow Distribution Studies Set-up**

This experimental set-up has been established to study flow distribution in a natural circulation driven system like Main Heat Transport (MHT) system of the Advanced Heavy Water Reactor (AHWR). This set-up consists of ten parallel flow channels, each provided with individually controlled heat sources. Largely transparent construction of this set-up facilitates visualization of phenomenon like thermal hydraulic Instability, flow pattern transition etc., at atmospheric pressure and temperature up to 100 °C.

Although this set-up is built to study sensitivity to the process parameters of flow sharing between parallel coolant channels, modular construction makes it a versatile tool useful for a wide variety of other studies in the area like reactor start up procedures, possible channel flow reversal, effectiveness of flow pattern control devices and effect of change in system configuration.

For providing insight in to the flow distribution at different power levels flow, temperatures and power of individual channels are monitored. The channel flows are measured using a specially designed low-flow, low-loss venturi meters. The process parameters are displayed on-line in the mimic flow diagram by a PC-based data acquisition system.

**Investigations on parallel channel instability and simulation of void reactivity feedback in the Parallel Channel Loop**

Parallel channel instability is the controlling instability (i.e. having the least stable region) and is an important design consideration for the AHWR. Apart from this, two-phase natural circulation flow exhibits phenomena like flashing and geysering which occur predominantly at low pressure and hence is a concern during the start-up. To study these phenomena an experimental facility – Parallel Channel Loop (PCL) is being set-up. This facility will also simulate the void reactivity feedback effects of the AHWR. This is very important in the context of a boiling-water-reactor where power oscillation due to void reactivity feedback may either reinforce or suppress the flow instability. The main objectives of the Parallel Channel Loop are:

- steady state natural circulation behaviour with equally as well as unequally heated channels,
- generation of out-of-phase (regional) and in-phase (global) instability maps,
- void fraction measurement using conductance probe,
- simulation of neutronic feed back on thermal hydraulic oscillations,
• study of carryover and carryunder: demonstration using transparent sections in riser and downcomer,
• study of low-pressure two-phase instabilities like flashing and geysering,
• Effect of nano-particles on natural circulation and stability.

The pre-test analysis are carried out using RELAP5/Mod3.2 and the illustration represents a typical instability obtained using this code.

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- Experimental studies on Scaled Model of Advanced Accumulator with Fluidic Flow Control Device

A 1:5 scaled model of advanced accumulator with fluid flow control device has been designed, fabricated and installed at BARC. The fluid flow control device is a simple passive device, which allows initially large amount of flow from the accumulator and later reduces the flow automatically due to formation of vortex.

When the water level in the accumulators is above the standpipe the water enters the fluid flow control device through both the inlets i.e., stand pipe and side connection, and since the flow is smooth, a large flow of water is discharged from the accumulators. When the water level in the accumulators drops below the top of the stand pipe, the water enters the chamber through the side connection only, which is tangential to the chamber. This increases the flow resistance, due to formation of vortex, resulting in reduction of flow.

Three devices of different dimensions have been fabricated to evaluate their performance. The objectives of the scaled model are performance evaluation, verification of low characteristics, studies on water level transient in stand pipe at transition and expansion behaviour of nitrogen, Two sets of experiments have been completed for initial pressure up to 1 MPa. The graph shows variation of accumulator discharge flow with time.
Advanced Heavy Water Reactor

Scaled Model of Accumulator with Fluidic Device

Accumulator Flow Characteristics with Fluid Flow Control Device

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Studies on Thermal Stratification

Thermal stratification denotes the formation of horizontal layers of fluid of varying temperature with the warmer layers of fluid placed above the cooler ones. Thermal stratification is encountered in large pools of water increasingly being used as heat sink in new generation of advanced reactors like GDWP of AHWR in which Isolation Condensers are immersed.

Stratification influences heat transfer to pool to a great extent and heat storage capacity of the pool in the form of sensible heat is significantly reduced. It can also threaten the structural integrity if the pool is made of concrete. Hence thermal stratification is not desirable and we have to minimize its effect.

Experiments were carried out in a rectangular glass tank (440 x 100 x 300 mm) with an immersed strip heater for visualization of thermal stratification phenomena. Theoretical analysis was also carried out for this case using CFD (Computational Fluid Dynamics) codes TRIO_U and PHOENICS. For further investigation of this phenomenon, a case study has been carried out with side heated cavity containing water. This 2-D problem is solved to obtain the velocity and temperature profiles using CFD codes TRIO_U and PHOENICS. Simulation of whole isolation condensers will be carried out to study the thermal stratification phenomenon in GDWP pool. Based on these simulations, the configuration of isolation condensers will be proposed such that the effect of thermal stratification is minimized.
Temperature contours after 15000 s for constant heat flux of 500 W/m²

Velocity vectors after 15000 s for constant heat flux of 500 W/m²

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Studies on Passive Containment Cooling System (PCCS) of AHWR

Containment is a key component of a reactor system, since it is the last barrier designed to prevent large radioactivity release to the environment under accidental conditions. The Passive Containment Cooling System (PCCS) removes the energy released into the containment during a postulated Loss Of Coolant Accident (LOCA). Two alternative designs for PCCS are under consideration. One of the alternatives being considered is a system in which the containment steam condenses inside the vertical tubes immersed in a pool of water. In the second alternative, the containment steam condenses on the Passive External Condenser (PEC) which comprises of cooling tubes connected to a water pool above it via headers as shown in Figure. The containment steam condenses on the outer surfaces of the tubes and water from the pool circulates through these tubes by natural circulation. An important aspect of the working of PCCS is the potential degradation of the performance due to the presence of noncondensable gas in the vapor.

First design concept (steam condensation inside vertical tube)

Experiments have been carried out for steam condensation inside vertical tube in presence of noncondensable gas. Figure shows the comparison of the predicted local heat transfer coefficients inside a vertical tube with experimental data.

Second design concept (steam condensation outside inclined tube)

Design of the Passive External Condenser (PEC) for PCCS has been carried out. A computer code CONISA (CONdensation In Steam-Air mixture) has been developed to estimate the heat transfer coefficient, for steady-state free convective film condensation in presence of noncondensable gas where the steam condenses on the cooling surface. The model has been validated against experimental data available in literature. The results have also been compared with the correlations developed by Uchida and Dehbi. The variation of heat transfer coefficient with change in air (noncondensable gas) mass fraction in the steam-air mixture is illustrated. To validate this model, an experimental setup has been fabricated, installed and commissioned.
In severe accidents, other noncondensable gases like hydrogen can get mixed with the steam-air mixture making the treatment of multicomponent noncondensable gases having different mass diffusion coefficient important. The computer code CONISA has been modified to account for the multicomponent noncondensable gases mixed with vapour. A computer code CONFIN has also been developed for the passive external condenser to account for finned tubes.

Measurement of Reactor Power and Flow Using $^{16}$N gamma signals

The power and flow measurement technique using $^{16}$N signals is an on-line and non-intrusive method, independent of the flow regimes. Hence this measurement technique is of importance to the AHWR. The thermal power & flow measurement experiments have been carried out in Dhruva reactor. The results are given in the illustrations.

$^{16}$N is produced in the core by fast neutron activation of $^{16}$O of the coolant and decays with a half-life of 7.35 s emitting 7 MeV gamma rays. The $^{16}$N activity i.e., the radionuclide production and the thermal power are directly proportional to the neutron flux and hence $^{16}$N activity is used for on line power measurement. Coolant flow measurement is obtained from the transit time measurement, i.e., by cross correlating $^{16}$N noise signals from two detectors, placed along the flow path of the coolant.

Experiments have been carried out in the coolant loop-3 of the Dhruva reactor, by placing two NaI scintillation detectors (with out any shielding), 90 cm apart, below the coolant pipe line. The detectors were located 2 cm away from the surface of the coolant pipeline, positioned by mild steel platforms. The signals were captured and analysed in the FFT analyzer from the reactor hall. The power and the flow measurements have been carried out at different power levels from start up to 40 MWT. The dc output signals were found to be proportional to the power level. The measured transit time was 156 ms and the corresponding calculated velocity was $5.76\pm0.17$ m/s and the corresponding flow rate worked out to be $2.34\pm0.07\times10^4$ l/min, which agreed well with the actual flow rate of $2.18\times10^4$ l/min, indicated by the venturi flow meter. The experiments have been repeated with $^6$Li glass scintillation detector and similar results were observed, (BARC Report/2002/I/023).
Two phase flow measurement

The important addressed problems in thermal hydraulics of the Advanced Heavy Water Reactor (AHWR) are channel thermal power measurement and detection of critical heat flux (CHF). The process in Main Heat Transport (MHT) System of AHWR is partly single phase and balance is in two-phase. Direct measurement of two-phase parameters such as void fraction and its distribution, two-phase flow and finally thermal power output of the reactor is essential and a complex task. Research and development activities in these essential and difficult measurement fields have been taken up. Following are the instruments and measuring systems under development.

- Capacitance type on-line void fraction measuring system
- Void Fraction Measurement by Conductivity Probe
- Two Phase Mass Flux Measurement by Pitot Tubes Assembly and Gamma Densitometer
- Rotating Electric Field Admittance Probe

Development of Two-phase flow Instrumentation

Thermal power produced by fuel can be measured by measuring the void fraction in tail pipe and other parameters of interest along with suitable correlation. In order to fulfill this requirement, development of capacitance type void fraction sensor has been taken up as no suitable instrument as available for in-situ measurement.

Variation in void fraction, within the volume enclosed by two parallel plates of an electrical capacitor, leads to variations in the value of the capacitance accordingly. This phenomenon has been used for in-situ measurement of steam/water fraction in a metallic pipe, carrying high pressure and high temperature fluid medium. Capacitor elements are installed with enough electrical insulation such that the volume between the parallel plates only contributes to impedance under measurement.

In order to verify the principle of operation and performance of the sensor, an experimental setup was designed, fabricated and installed in the Heat Transfer Laboratory.
Two sets of sensors with angular aperture of 30° and 150°, 68 mm ID & 68 mm length have been installed. Output signal range for 30° sensor plates was found to be 2.98 pF to 66.41 pF and signal range for 150° sensor plates was found to be 4.40 pF to 110.02 pF with 100% air to 100% water. A maximum percentage variation in normalized output signal because of constant air bubble flow of 5 lpm at various (X-Y) positions was found to be 1.5%.

**Void Fraction Measurement by Conductivity Probe**

The knowledge of void fraction (fraction by volume of gas phase to a total given volume) forms an important part of two-phase analysis. In nuclear reactors, pipelines and other industrial operations, accidental rupture of a pipe produces a two-phase flow of gas and liquid. In nuclear reactor design, measurement of various two-phase flow parameters is important to verify analytical procedures for predicting reactor behavior in a loss of coolant accident. Cross-sectionally averaged mixture density (or void fraction) and mass flux are the most difficult parameters to measure.

Most of the void fraction measuring techniques which are based on the effects of nuclear reactions, such as gamma attenuation, beta attenuation, X-ray attenuation, neutron diffusion, are not applicable inside the reactor cores where intense fields of all these nuclear radiation is predominant.

Among the non-nuclear methods of void fraction measurement, one of the most important methods is the electrical impedance method. The electrical impedance of a two-phase flow depends on the concentration and distribution of the phases.

Depending on the system, the impedance will be governed by conductance or capacitance. Based on conductance principle to measure local void fraction the Single point conductivity probe and Five point conductivity probes were developed.
Principle of Measurement

- The principle of measurement is based on the difference in conductivity between liquid and gaseous phase.
- When probe tip in contact with liquid phase, the circuit between two electrodes is closed, whereas the circuit is opened as soon as probe tip touches the bubble.
- The time averaged local void fraction is given by

$$\alpha = \sum_{i=1,2,3 \ldots N}^N$$

Where $N$ is total number of samples and $(i)$ is the number of samples indicating presence of gas at the probe tip.

Construction of single point Conductivity Probe

This type of probe is constructed by using S.S wire of 1 mm diameter and insulated by using teflon sleeve as shown in Figure, for measurement of bubble time, one end of the wire is shaped conically and kept un-insulated and the other end of the wire is connected with supply. The metal pipe line through which two phase flow is flowing acts like a second electrode.

Five point Conductivity Probe with Teflon sleeve as insulation

This type of probe is constructed by using 5.5 wire of 0.5 mm diameter and insulated by using Teflon sleeve. The probe has five pointed electrodes which face into the flow. The tip of wires form the bubble detecting electrodes. One electrode is at the center of the pipe and two pairs are opposite each other at radii of 0.707 and 0.913 times the inside radius of the pipe. The cross-section of the pipe can be regarded as consisting of a circle and two annuli all of equal area, with one electrode monitoring the circle and two electrodes monitoring each of the annuli.

Two Phase Mass Flux Measurement by Pitot Tubes Assembly and Gamma Densitometer

Mass velocity and void fraction measurements are required in transient two-phase steam – water flow experiments related to thermal hydraulic studies on reactor safety. Thus experimental measurements play a key role in providing information for design, analysis and predicting system behaviour. Hence, design and development of simple, rugged and inexpensive pitot tubes based flow sensor for cross section averaged mass velocity measurement and measurement of chordal void fraction and average mixture density by gamma densitometer has been carried out. The design criteria followed are: (a) Local measurement near the centre region of the horizontal chord gives good result to predict the chordal mass flux in most flow regimes due to the uniformity of the local mass fluxes along the chord. (b) The cross section averaged mass flux calculations using pitot tubes along the chord were in good agreement if horizontal mixture density was used in place of local mixture density and (c) it is relatively easy to measure chordal average mixture densities by gamma ray attenuation.

Sensor Design

The present design consists of five pitot tubes located across the pipe section and positioned in a vertical line, one at the centre of...
the pipe, one each at 0.286 diameter and 0.429 diameter above and below the centre of the pipe to ensure that all the flow regimes are covered. Five differential pressure transmitters measure the velocity heads for the pitot tubes. The pitot tubes assembly is installed in the two-phase horizontal section of High Pressure Natural Circulation Loop as shown in figure for sensor validation. The two-phase velocity profiles obtained across vertical chord are shown in the graph. The gamma densitometer consists of Cs-137 source with principal photon energy of 0.662 MeV, NaI(Tl) scintillation detectors, single channel analysers with associated electronics for chordal average void fraction measurements. The present device is developed for simple operation and data interpretation. The Cs-137 gamma source is ordered through BRIT and commissioning of gamma densitometer is in progress.

This pitot tubes-based sensor is rugged; simple, reliable and easy to operate. It is a sturdy sensor for adverse high temperature and high-pressure steam water applications. The sensor introduces only very little disturbance in flow path, which is very important for two phase natural circulation studies. Flow regime identification is possible using chordal void fluctuations.

**Development of Rotating Electric Field Admittance Probe**

Electrical impedance techniques have proven attractive for many applications because of their generally fast response to void variations, high signal output and simplicity of operation. The important design criteria are the sensor shall not introduce disturbance to the flow, it shall not cause pressure drop in the system and the measurements data shall predict various flow regimes in two-phase mixtures. The present novel electrical probe, which is being developed to measure void fraction in a pipe of circular cross section, meets these requisites. In the present design the electrodes are mounted coaxially inside and form part of the pipe wall to eliminate the disturbance to the flow. Further the electric field, which is perpendicular to the flow, is rotated electronically to distribute it throughout the sensor volume.

**Sensor Design**

The sensor consists of six stainless steel electrodes separated by ceramic insulators. These electrodes and insulators form the inside perimeter of the sensor flow area. The three electrode pairs are excited such that the signal for each pair is 120 degrees out of phase with the other. A rotating electric field is thus generated within the sensor volume. Three identical admittance measurement circuits are connected to the electrode pairs and the absolute values of the signals from these circuits are summed. The resulting signal

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is proportional to the admittance between the sensor electrodes. A reference sensor, which produces a signal proportional to the admittance of the single-phase liquid, is used to compensate for the changes in the admittance of the liquid due to variations in the temperature and concentration of impurities. The relative admittance of these two sensors gives the void fraction of the two-phase mixture. The sensor details are given in the figure. The probe electronics measures magnitude and phase of the admittance. By measuring relative variations in conductive and capacitive components and fluid impedance particularly the phase angle, the flow distributions can be identified. The phase velocities can be predicted from the transit time signals obtained from the pair of probes by random signal analysis methods.

This technique is simple and relative simplicity of operation/data interpretation makes it feasible for real time measurements. Also the probability density functions of relative admittance fluctuations can provide flow pattern discrimination. Since it is a volume average technique it will give better results and because of faster response, can be used for transient studies.

The fabrication of low temperature sensor for air water calibration is completed and fabrications of signal conditioning modules are in progress.

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- **Super Critical Water Natural Circulation Loop (SCW-NCL)**

Super critical fluids have the advantage that there is no phase change above the critical point (CP) eliminating the occurrence of the critical heat flux phenomenon. The large variation in thermal expansion coefficient near the CP can be exploited for designing natural circulation-based Super Critical Water Reactors (SCWRs). From the viewpoint of design, it is essential to identify the operating parameters near the CP such that high circulation rate and hence high heat transfer rates are achievable during natural circulation. In addition, it is desirable to operate such loops in a stable condition, which requires identification of the stable and unstable zones by a stability analysis. Experimental determination of stable and unstable zone in loops of different diameters is planned.

A computer code has been developed to carry out steady state and stability analysis of a super critical water Natural Circulation Loop (NCL). Using the computer code, steady state analysis has been carried out to obtain the parametric effects on the natural circulation...
flow rate in a supercritical loop. The loop has been designed for 300 bar pressure and 400°C temperature.

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1.10 EVALUATION OF DESIGN BY INPRO METHODOLOGY

BARC participated in the IAEA initiated International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) which involved evolving a methodology for evaluation of innovative nuclear reactors and fuel cycles. The AHWR has been accepted as a case study to test the adequacy of INPRO methodology, to suggest any changes, if needed and to evaluate AHWR using INPRO methodology. The methodology consists of application of basic principles, user requirements and criteria in six areas of economics, environment, & sustainability, safety of nuclear installations including fuel cycle facility, waste management, proliferation resistance and cross cutting issues.

The basic principles, user requirements and the criteria in all the above-mentioned six areas were reviewed and modifications were suggested. The evaluation of INPRO methodology for the AHWR comprised of:

- identifying the salient design features of the reactor,
- listing the attributes affecting each of the design features and
- evaluating effect of INPRO parameters on each of the attributes.

This procedure was well appreciated by the agency. The case study has been completed and a final report has been sent to IAEA containing the description of the plant, near and long-term fuel cycle of AHWR, review of the Basic Principles, User Requirements and the Criteria, salient design features of AHWR and case study evaluation.

The following table is part of the evaluation of INPRO methodology for AHWR in the area of fuel cycle. The different design features of AHWR related to fuel cycle are listed and their effects on different areas of INPRO parameters are indicated.

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<th>Main design Features</th>
<th>Economics</th>
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<th>Safety</th>
<th>Waste Management</th>
<th>Proliferation Resistance</th>
<th>Cross-Cutting Issues</th>
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