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 feedback 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 dynamic 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 feedback 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 occur depending 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 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 neutronic 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 RELAPS. 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 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), \( N_f \). 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}{\text{\( \nu \)}} \frac{\partial \rho}{\partial h} \), as been used, where, \( \nu \) 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

- Steam Drum
- Isolation Condenser
- Advanced Accumulator
- Integral Test Loop (ITL)
- Fuel Channel Simulator
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.

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 MWe).
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 ($\Phi_{2}^{LO}$) 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.
Schematic of Flow Test Facility

54-Rod Bundle Assembly and Cross-sectional View

Fuel Pin OD = 11.2 mm, Number of Pins = 54
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

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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 play 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.

Radiograph

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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.
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 into 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 neutron feedback 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.
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.

Passive external condenser experimental set-up

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- Measurement of Reactor Power and Flow Using $^{15}$N gamma signals

The power and flow measurement technique using $^{15}$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 Dhrupa reactor. The results are given in the illustrations.

$^{15}$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 $^{15}$N activity i.e., the radionuclide production and the thermal power are directly proportional to the neutron flux and hence $^{15}$N activity is used for on line power measurement. Coolant flow measurement is obtained from the transit time measurement, i.e., by cross correlating $^{15}$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 Dhrupa 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±0.17 m/s and the corresponding flow rate worked out to be 2.34±0.07 x 10$^4$ l/min, which agreed well with the actual flow rate of 2.18 x 10$^4$ l/min, indicated by the venturi flow meter. The experiments have been repeated with $^4$Li glass scintillation detector and similar results were observed, (BARC Report/2002/I/023).

Typical $^{15}$N noise signals from a pair of detectors installed near the pipeline of Coolant loop # 3 of Dhrupa at 40 MWT.
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.

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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. 
Design of industrial grade instrument for measurement of void fraction inside the metallic pipes, has been done as a spool piece of the straight piping with flanged end connections with high pressure and high temperature rating as per ANSI code. Ceramic pipe sheath is provided inside the metallic spool piece and four parallel plates of SS316 are installed on inner surface of the ceramic pipe, thereby ensuring the electrical insulation and with out any disturbances to the fluid flowing in the pipe. Hardware components for assembling the instrument are under advanced stage of procurement.

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

\[ \alpha = \frac{1}{N} \sum_{i=1}^{N} a(i) \]

Where N is total number of samples and \( a(i) \) 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 S.S 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

Two phase velocity profiles across the vertical chord

Sensor details
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 supercritical 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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