The present nuclear power programme in India is based mainly on a series of 220 MWe pressurised heavy water reactors. These reactors use pressure tube technology with natural uranium as fuel and heavy water as moderator and coolant. Currently, there are fifteen PHWRs in operation including 540 MWe TAPS 4, the first of its kind in India. This chapter highlights the current R&D activities being carried out in the field of physics design & safety assessment, fuel management studies, on-line flux mapping systems, fuelling machine testing, design of control and shut off rod mechanisms, experimental studies and severe accident programmes.
2.1 FUEL MANAGEMENT STUDIES TO OPTIMIZE FUEL UTILIZATION IN INDIAN PHWRs

The optimised fuel management makes the best use of the fuel and therefore reduces the unit energy cost. Hence different possible alternatives have been studied to optimise the fuel utilisation in Indian PHWRs. A few of them have been successfully implemented in the power reactors. Fuel management code TRIVENI is being used in all the power stations. The equilibrium core optimisation studies have been performed by TAQUIL and the snapshot core calculations were performed by TRIVENI.

The following burnup optimisation studies have been carried out to improve the fuel utilisation in Indian PHWRs.

- Full power operation with corner adjusters fully withdrawn from the core.
- Reactor operation at reduced power with peaked flux distribution.
- Regular use of Depleted Uranium (DU) fuel at full power operation.

- Use of depleted uranium in the initial fuel loading of the new reactors and the reactors started after en-mass coolant channel replacement activity.
- Use of depleted uranium and deeply depleted uranium (about 0.3% w/w) fuel in the initial fuel loading of TAPS-4 reactor (540 MWe).

Lattice calculations for MOX-7, MOX-97, MOX-888 clusters and equilibrium core burnup optimisation studies have been carried out for the PHWRs. Feasibility studies have been carried out for the use of thorium MOX cluster such as MOX-Th24, MOX-Th20 in PHWRs. The core calculations with regard to the utilisation of MOX-7 cluster as well as MOX-888 cluster along with natural uranium at large scale in PHWRs have also been carried out.

The operation in the peaked flux distribution scheme has been implemented at all the PHWRs and more than 30% saving in natural uranium has been achieved. Usage of depleted uranium in PHWRs has been implemented at RAPP-3&4 & KGS-1&2 and 30% saving in natural uranium has been realised. Test irradiation of 50 bundles of MOX-7 is under progress at KAPS-1. Existing PHWRs can use 40% of MOX-7 or MOX-888 along with natural uranium.
2.2 REACTOR PHYSICS DESIGN AND SAFETY ASSESSMENT OF PHWRS

- Criticality of 540 MWe PHWR
Safety of the physics design and commissioning procedures have been evaluated for TAPS 3 & 4 units, of the 540 MWe PHWR design. Significant changes in design of control and safety systems have been incorporated for the first time in these reactors, due to their larger size, compared to the standard 220 MWe PHWRs. Technical specifications for operation has been evolved for first approach to criticality and Phase – B commissioning of TAPS Unit # 4, the first reactor of this kind in the country.

- Maximisation of discharge burn-up of PHWR fuels
The design support and safety analysis for the changes in the fuelling strategies for smooth transition from two burnup zone pattern (Flat flux) to one zone pattern (Peak flux) for maximising the discharge burnup have been carried out. The evaluation of trial irradiation of Pu MOX bundles in KAPS-1 reactor has been completed. Presently, the design discharge burnup of PHWR bundle is 15,000 MWd/t U and few bundles were irradiated beyond this limit up to 20000 MWd/t U, to obtain physics data such as the fall in the power production, changes in the reactivity after refueling and also metallurgical details through post irradiation examination (PIE). The extended irradiation of natural uranium bundles in two specified channels have been analysed.

- Evolution of criticality procedures for re-tubed reactors and upgradation of Computer code “Triveni”
Criticality procedures have been evolved for startup of reactors after long shutdowns like en-mass coolant channel replacement and these procedures have been approved by safety and regulatory bodies (SARCOP and AERB) and have been implemented in the approach to criticality of MAPS, Unit-2.
Modification of the fuel management code TRIVENI as and when required and inclusion of special features such as the supply of latest nuclear data files for better estimations of reactivity coefficients is one of the important ongoing activities.

Performance evaluation of Self Powered Neutron Detectors

Data collection and analysis of the linear response with power changes and dynamic response of SPNDs (Self Powered Neutron Detectors) following transient change of power in KGS – 1 and RAPS-3 was another valuable exercise taken up to validate and gain confidence in the usage of SPNDs for regulation and protection in 540 MWe units. Usage of SPNDs for incore monitoring of neutron flux during power operation would be a necessity in large size reactors and hence has a lot of relevance for the AHWR as well as 700 MWe PHWR.

2.3 ON-LINE FLUX MAPPING SYSTEM (OFMS) DEVELOPMENT FOR TAPS-3&4

Large power reactor systems, such as 540 MW(e) PHWRs, are known to sustain power tilts following minor local reactivity perturbations. If these tilts are not controlled, they can lead to unacceptable power distributions. Therefore, it is essential to have in-core neutron flux measurements and also to process them to get knowledge of the operating state of the power reactor with the help of the On-Line Flux Mapping System. This system is a virtually continuous power regulating scheme to maintain the core power distribution closer to the design intent.

Software for obtaining the power distribution from the in-core neutron flux measurements has been developed.

The physical OFMS system

Flux Mapping (FM) detectors

A total number of 102 Vanadium type in-core Self Powered Neutron Detectors (SPNDs) are provided in 26 vertical assemblies. These are called the Flux Mapping (FM) detectors. They are one pitch (28.6 cms) in length and made to measure the point thermal flux at their locations. The electrical signal generated in the FM detectors is due to the β decay of V. The relevant reaction that takes place is as follows.

\[ \text{V}^{\text{51}} \ [\text{stable}] \xrightarrow{\text{\beta}^-} \text{V}^{\text{52}} \xrightarrow{\text{n,y}} \text{Cr}^{\text{52}} \ [\text{stable}] \]

The response of FM detector lags behind the flux by about 5 minutes due to β decay of V.

Flux Mapping (FM) Software

In OFMS software, the instantaneous neutron flux is expanded as combination of pre-determined flux shapes whose combining coefficients are determined from the on line flux measurements by employing the Least Square method. The LS procedure reduces the flux mapping software in just two matrix multiplication. The first multiplication gives the instantaneous values of the combining coefficients for modes corresponding to the FM detector readings and the other multiplication gives the flux map from the combining coefficients. Flux map thus generated is used for calibrating the Cobalt SPNDs, used for controlling the reactor.

Validation of OFMS

Detailed programs for validation of OFMS software includes the following:

Static configuration validation

The flux map generated is tested against large number of theoretical simulation of possible reactor operating states. In this validation, the detector readings are estimated from the flux shape itself with random errors sampled from Gaussian distribution superimposed on detector readings. Such detector readings are fed to OFMS software to get the flux which is then compared with the actual simulation. In this simulation no dynamic feed-backs such as xenon, temperature etc. are considered. Gross global feed-backs (as determined by the cross-sections) are considered. This validation is used for determining the size of basis set for expansion.
**Dynamic configuration validation**

Under this scheme, the OFMS software is extensively tested in dynamic close loop environment. A space-time kinetics code with dynamic RRS feed-back, xenon feed-back and temperature feed-back is developed for this purpose. A detector model for both vanadium and cobalt SPNDs is also included to give realistic estimate of detector readings. This validation gave the insight to the adequacy of the FM system.

**Experimental validation**

As part of validating the OFMS software, flux mapping experiments have been carried out by simultaneous irradiation of copper wires in five pre-selected in-core lattice positions of CIRUS at low power (~230 kW). Some salient features of OFMS software were validated during the course of these experiments.

The present design of OFMS configuration is arrived at from the above validation. There will be a total number of 29 flux shapes (Fundamental + 18 lambda modes + 8 shapes of adjuster bank withdrawal + 2 shapes for control rod insertion) in the expansion set. The reactor regulating system will supply the adjuster rod and control rod bank status to OFMS. OFMS will then select required number of flux shapes from these 29 shapes depending upon the devices’ status. OFMS has also built in algorithm to consider the effect of failed detectors.

The errors in FM are sensitive to the difference between the fundamental mode in which the reactor is operating and the actual mode set used in OFMS software. Therefore the complete set of 29 flux shapes will be changed depending upon the core irradiation status (at 0 FPDs, 40 FPDs, 100 FPDs, 400 FPDs and the equilibrium core). This is done to minimize the prediction errors in flux map and to get better knowledge of the operating state of the reactor.

The Hardware implementation and testing of OFMS was carried out at ECIL using generated detector signals from the dynamic core simulation code. The onsite commissioning of the OFMS is completed at TAPP 4 site, Tarapur. The OFMS is loaded at site with initial core (0 FPDs) mode set. The in situ testing and fine tuning the response of OFMS will be undertaken as and when the full power operation of the reactor commences.

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### 2.4 FUEL ANALYSIS

#### PHWR fuel

In the context of the round-robin exercise carried out in BARC, analysis for the PHWR fuel bundles irradiated in KAPS has been carried out using the modified GAPCON code. The peak center line temperature evaluated was about 1600°C (for 51 kW/m linear heat rating) and cumulative fission gas release of about 3% has been calculated for a fuel burnup of 15 GWd/t. The analysis results have been compared with the post irradiation examination results.

#### MOX fuel

A 19-rod MOX bundle has been designed for loading in PHWRs and 50 of these bundles have been loaded in the KAPS reactor. MOX fuel for TAPS has been designed and twelve bundles have been successfully irradiated.

An analysis has been carried out using ANSYS to evaluate the fuel temperatures due to hot spot formed by Pu-agglomeration, for various Pu particle sizes at different heat generation rates for TAPS-BWR and PHWR (220MW) MOX fuel. The meshed finite...
An analysis has been carried out for MOX fuel pins irradiated at PWL CIRUS, as part of round robin exercise, using modified GAPCON code. The peak centreline temperature evaluated was in the range of 1300°C (for 42kW/m linear heat rating) and cumulative fission gas release of about 2.0% for fuel burnup of 16 GWD/t.

An analysis has been carried out for MOX fuel pins irradiated at PWL CIRUS, as part of round robin exercise, using modified GAPCON code. The peak centreline temperature evaluated was in the range of 1300°C (for 42kW/m linear heat rating) and cumulative fission gas release of about 2.0% for fuel burnup of 16 GWD/t.

2.5 Testing of Fuelling Machine of 540 MWe PHWR

The Fuelling Machine of 540 MWe PHWR Reactor is an important intricate equipment. For satisfactory performance of the assembled Fuelling Machine, Performance & acceptance testing of Fuelling Machine is carried out under the simulated reactor conditions.

- Fuelling Machine Test Facility (FMTF)

Fuelling Machine Test Facility has been set up and fuelling machine head is installed on the test carriage and the oil hydraulic water hydraulic systems have been commissioned. The fuelling machine head has been calibrated & commissioned. Necessary rectification and fine-tuning of fuelling machine head have been incorporated by suitable modification / changes of components.

The following tests were done in reactor-simulated condition:
- Performance testing: 6 times bundles loading & 6 times bundles receiving.
- Acceptance testing: 10 times bundles loading & 10 times bundles receiving.
- Cycling of Ram extension: 10 times loading and 10 times receiving.

The fuelling machine head has been qualified with successful completion of above tests. Full testing has been completed in very short period of 5 months.

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2.6 TESTING OF RAM ASSEMBLY OF 540 MWe PHWR

For satisfactory performance of the assembled fuelling machine, elaborate testing of various sub-assemblies of fuelling machine is carried out separately. Later, the assembled fuelling machine has been again completely tested before delivery to the site.

RAM assembly is one of the important subassembly of the fuelling machine and it is the first of its kind, to be used in 540 MWe PHWRs. It comprises of three telescopic rams namely B Ram, C Ram & Latch Ram. All these rams are operated through oil hydraulic motors. B Ram & C Ram are provided with water hydraulic back up. These rams are used for refuelling operations & various plug operations.

Ram assembly and the Oil & water Hydraulic system has been installed and commissioned. The calibration & commissioning of the Ram assembly in the Ram assembly test facility has been completed.

Necessary rectification and fine tuning of Ram assembly have been incorporated by suitable modification /changes of components. The B Ram Cycling equivalent to 300 channel refueling operation & Plug operations equivalent to 100 channel refueling operation have been successfully carried out to qualify the Ram assembly.

2.7 DESIGN AND DEVELOPMENT OF ADJUSTOR ROD, CONTROL ROD AND SHUT-OFF ROD DRIVE MECHANISM FOR TAPS 3 & 4

This development work has been taken up under an MoU signed between BARC and NPCIL. The control mechanisms for TAPS units 3&4 are designed with a number of advanced features like modular construction giving ease of maintenance, 90% free fall for shut-off rod/control rods giving high reliability and consistent rod drop performance, on-line test facility for shut-off rods to ensure rod availability on demand while reactor is under operation and partial release & re-arresting of control rods for reactor step back function.

Design, prototyping, testing & qualification on full-scale test set-up have been completed. Manufacture, assembly and testing of drive units for reactor use have been completed. They have been installed and commissioned at TAPS-4.
2.8 FISBE-EXPERIMENTAL STUDIES AND AUGMENTATION OF FACILITY FOR ACCIDENT SCENARIOS AND OPERATIONAL TRANSIENTS

FISBE is an integral test facility to simulate PHWR Primary Heat Transport System including Emergency Core Cooling System as well as secondary heat removal circuit. Maintaining the elevations same as in the reactor, power to volume scaling philosophy is followed in the design of the facility. Volume scaling in respect of 220 MWe PHWR units is 1:76.5. Time scale is preserved. The test facility is extensively instrumented to measure temperature, flow rate, pressure/differential pressure, level, void fraction etc., at various points in the loop. This facility will be used for LOCA and non-LOCA transient experiments. These experiments will help in the creation of a data base for the assessment and validation of thermal hydraulic computer codes used for predicting the transient behaviour during postulated accidents and operational transients.

Experiments have been conducted for single-phase and two-phase natural circulation and station blackout studies in the FISBE facility. Some of the results based on the data generated in the facility are given below. The facility is being augmented with simulated PHT Pumps, Fuel Cluster Simulators, Fast Acting Valves, Bleed Condenser, Regenerative Cooler, Jet Condenser, Pool Boiling Coolers, Low Pressure Accumulator and Secondary Feed Pump. The loop is instrumented with state of the art smart sensors and a PLC based control system. The loop is further equipped with a Fast Data Acquisition System (FDAS) for studying transient phenomenon.

Typical rod drop characteristics for shut-off rod for TAPS, units 3 & 4

Reactivity control mechanisms on deck plate of TAPS unit 4

Single-Phase Natural Circulation Results

Two-phase Natural Circulation Results
R&D for PHWR

Primary Circulating Pump

Low Pressure Accumulator

Quick Opening Valve

MIMIC

OPERATOR TERMINALS

PLC RACK

PC Based Control

Fast Data Acquisition System

Jet Condenser

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2.9 SEVERE ACCIDENT STUDY PROGRAM

Severe core damage phenomena can occur for a Beyond Design Basis Accident where multiple failure of safety systems has been postulated. A high degree of consequences is expected for a very low probability case. The world research covers mainly two parts namely severe core damage and containment loading with respect to energy release and radioactivity. In the field of severe core damage laboratory scale to full scale accident simulation are conducted to understand the complex phenomena arising from high temperature. Thermal-hydraulics, thermo-mechanical, high temperature chemical reactions, material relocation, etc. are the major fields studied extensively.

Efforts are made to understand different facets of the phenomena for Light Water Reactors (LWRs) as well as Pressurised Heavy Water Reactors (PHWRs) to assess the pressure boundary integrity for both the reactors and calandria vessel for the PHWRs. For the LWRs the understanding has been achieved with well experimentally
validated severe accident analysis computer codes namely ICARE/ CATHARE, ASTEC and RELAP/SCADAP and participation in the experimental validation program of PHEBUS-FPT0 experiment with code ICARE2. Some of the computational prediction are shown below for FPT0 experiment and fuel temperature transient for small break Loss of Coolant Accident with station Black out of VVER-1000.

For PHWR theoretical model developments namely Channel Analysis in Severe Accident (CHASA), Fuel Coolant Interaction (FCI) and Molten Pool (MPOOL) code are being developed. At present CHASA features in-channel thermal-hydraulic models, more emphasis on radiation among fuel pins and from fuel pins to pressure tube(PT), thermo-mechanical deformation model of pressure tube, calandriatube(CT) outer surface boiling crisis and high temperature chemical reaction model. Melting of the fuel pins, relocation and oxidation of the molten material, are the areas on which work is to be carried forward. Some predictions during the development period are cited below,

Two dimensional Computational Fluid Dynamics (CFD) Code development for Fuel Coolant Interaction (FCI) and behaviour of Molten Pool (MPOOL) arising from corium as a result of Beyond Design Basis Accident (BDBA) for PHWR has been carried out. The beginning of the FCI code development plant has been done with study on film stability followed by full scale code development. Effects of various fluid temperatures (degree of subcooling) on film growth are shown in the illustration.

To support the code development, experiments for studying pipe blowdown and vapor pull through in the reactor headers are planned for the initial phase of the accident. The next phase includes the experiments of channel heat up leading to thermo-mechanical phenomena in the pressure tube and boiling crisis of calandria tube with sagging and ballooning. Release of molten material from reactor channels and its distribution due to its neighbouring channels will be studied through the fuel coolant interaction experiments

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