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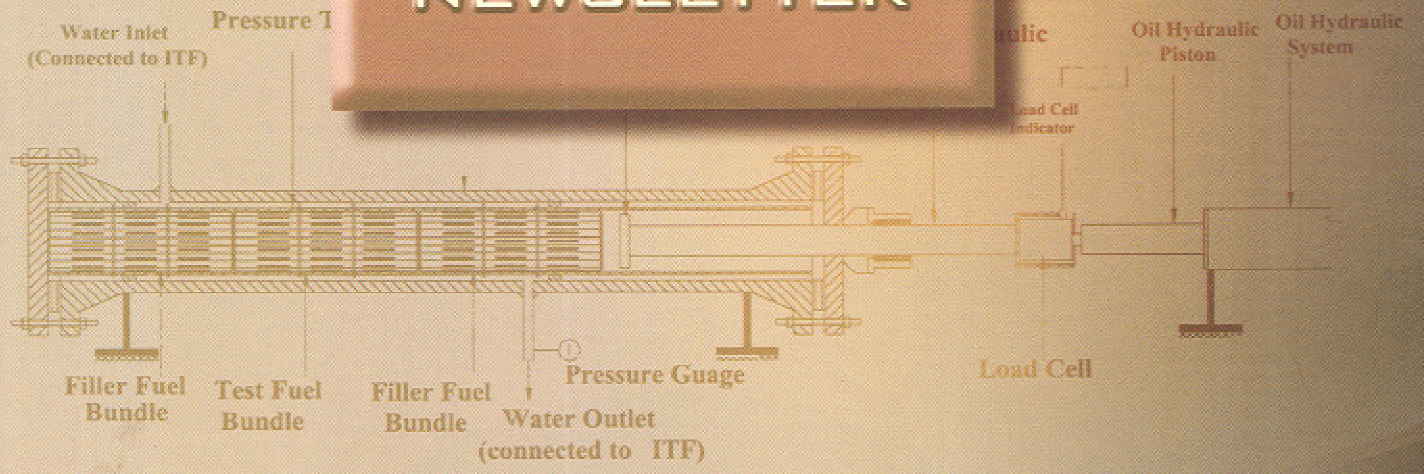
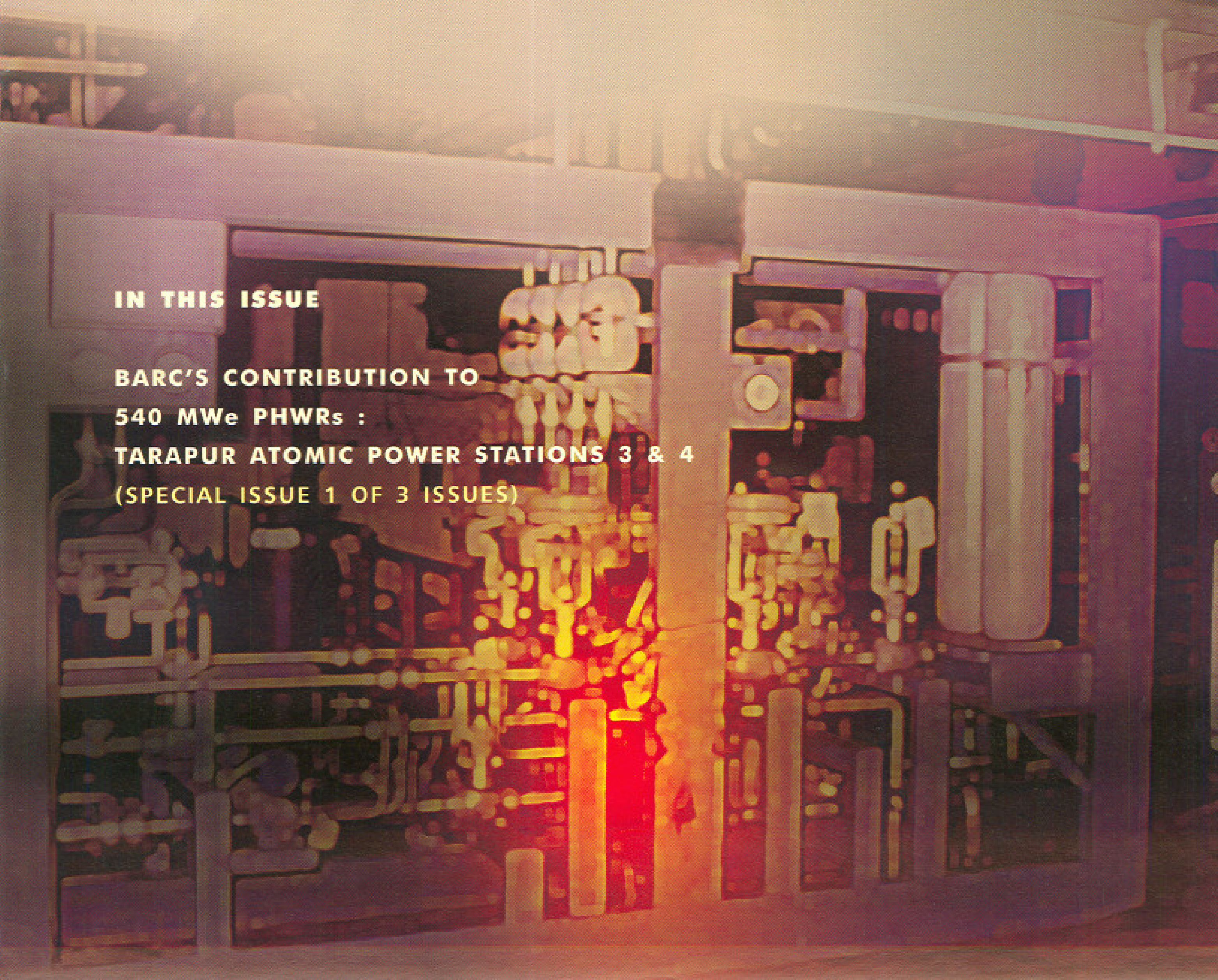
बीएआरसी न्यूज़लैटर

BARC NEWSLETTER





IN THIS ISSUE

BARC'S CONTRIBUTION TO
540 MWe PHWRs :
TARAPUR ATOMIC POWER STATIONS 3 & 4
(SPECIAL ISSUE 1 OF 3 ISSUES)

भाभा परमाणु अनुसंधान केंद्र
Bhabha Atomic Research Centre

पंडित जवाहरलाल नेहरू द्वारा परमाणु ऊर्जा संस्थान, ट्रॅाम्बे (जिसका बाद में श्रीमती इंदिरा गांधी द्वारा दिनांक 12 जनवरी 1967 को भाभा परमाणु अनुसंधान केंद्र के रूप में पुनर्नामन किया गया) का दिनांक 20 जनवरी 1957 को औपचारिक उद्घाटन किया गया था.

Atomic Energy Establishment Trombay (later renamed as Bhabha Atomic Research Centre on 12 January 1967 by Smt. Indira Gandhi) was formally inaugurated by Pandit Jawaharlal Nehru on 20 January 1957.

बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

CONTENTS

EDITORIAL

Power generation is one of the major peaceful applications of nuclear energy. BARC's main objective, is to provide R&D support, required to sustain this application. Right from the conceptualization of the nuclear power programme to the finalization of the design of the reactor and the peripheral components, BARC is involved, in various R&D aspects of power production.

Nuclear power production in India, involves a three-stage strategy of implementation. The first stage is the use of PHWRs. The objective of using PHWR technology is that, the limited Uranium resources available in India, are utilized in the best possible way, providing higher Plutonium yield into the bargain, which initiates the second stage of the power programme. This stage entails the use of Pu-239 (produced in the first stage) and U-238, in the fuel core of FBRs. Besides Pu-239, Th-232 is also used around the FBR core, which undergoes neutron capture reactions, leading to the formation of U-233. The same U-233 becomes part of the third stage of the Indian nuclear power programme. The vast Thorium reserves available in India, would be utilized along with the U-233 (produced in the second stage), in breeder reactors, which are under development.

Through totally indigenous R&D efforts, India has become self-reliant in PHWR technology. Apart from TAPS 1 and TAPS 2, all other power reactors are of PHWR type. BARC has extensive expertise, in the development of PHWRs in India. According to a study of published literature on PHWR research, (IAEA-CN-123/03/O/5; Vijai Kumar et al), the top ten rankings of authors are from India, specifically BARC. Much of the work on PHWRs, involved reactor fuels and reactor materials: modeling of fueling machine bridge and carriage assemblies of a typical PHWR system for seismic qualification; clad collapse test of prototype PHWR fuel elements; development of a TIG welding technique for end cap welding of PHWR MOX fuel elements; non-destructive testing in the fabrication of Zr-alloy tubes and PHWR fuel elements; experimental study of poison moderator interface movement for SDS 2 of a 540 MWe PHWR, to name a few. Three more research projects were exclusively devoted to TAPS 3 and TAPS 4; leak-before-break design of PHT elbows of TAPS 3 &4, using R6 method; drive-mechanism of SDS 1 and fault-tolerant, safety-related, computer-based process control system for TAPS 3 & 4. Besides these, BARC has made many more significant contributions to TAPS 3 & 4.

There was a need to highlight important technological innovations and to identify new areas of R&D. The July, August and September issues of the BARC Newsletter, are devoted to bring these achievements into focus. Considering the amount of material received, it became necessary to split it into three parts, to make it more manageable.

Vijai Kumar

PHYSICS DESIGN AND SAFETY ASSESSMENT OF 540 MWe PHWR

Baltej Singh, B. Krishna Mohan, Arvind Kumar, P. D. Krishnani and R. Srivenkatesan
Reactor Physics Design Division

The physics design work of 540 MWe reactor which started around 1984 and, training of the reactor physicists of NPCIL, were the responsibility of BARC. Over a long period of time, the Physics and Engineering design (by NPCIL) iterations of the core configuration, in-core physics, burnup optimisation, evaluation of reactivity devices, various incore monitoring algorithms, control and safety schemes, software and hardware of reactor regulation and protection systems, analysis of fuel management strategies and different computer code developments and their validation have been accomplished and the design was completed. Initially the core was designed for 500 MWe but subsequently, because of margins available in the MCP the reactor power could be increased to 540 MWe.

A number of computer codes were developed and/or evolved, to suit the requirements for design analysis and reactivity predictions of the 540 MWe PHWR. These are classified into the following five categories.

1. Computer codes for lattice calculations (CLUB)
2. Computer codes for Supercell Calculations of Reactivity Devices (BOXER).
3. Computer codes for Core Calculations (TRIVENI, TAQUIL and FEMINA)
4. Computer Codes for Simulation of Xenon Transient and RRS (TRIXEN).
5. Computer code for online flux mapping system and 3D-kinetics (3DFAST).

The calandria of the 540 MWe reactor, is a horizontal cylindrical vessel similar to 220 MWe PHWR, with its diameter being about 8 metres and length about 6 metres. Indian design of the reactor consists of 392 channels, whereas PICKERING –A (CANDU 500) design has only 390 channels. However, the safety and control system features were chosen to be similar to the then current design of CANDU 600. Moreover our reactor was made symmetric by addition of two more channels (H-01 & H-22) compared to Pickering.

In order to have better operating (bundle power) margins, the 37 rod cluster design was opted for as compared to the 28 rod cluster, used in CANDU 500, However, it cost a burnup loss of about 500 MWD/Te. Moreover, the choice of 37 rod cluster, which provides more margin on account of linear heat ratings, would help to raise the reactor power, without changing the cluster design. In addition to this due to non-boiling option of the coolant with 37 rod clusters, the PHT pressure increase, resulted in slightly thicker pressure tube (wall thickness of 4.5 mm as compared to 4.34 mm in CANDU 600), leading to an additional burnup loss of 180 MWD/Te.

Lattice Calculations

The lattice calculations for these advanced cluster designs was started, with the indigenously developed state-of-the-art code CLUB, based on multigroup integral transport theory. The computer code CLUB was a unique code which is treated cluster geometry exactly with multigroup cross section libraries. It was based on a combination of small-scale collision probability and large-scale interface current technique. This judicious combination yielded a computationally very efficient and an accurate method.

The computer code CLUB made use of the available WIMS 69-group cross section library. As the computer memory and the time required were both very large with the earlier computers, the 69-group cross section library was condensed to 27/28 group library, using a typical spectrum of a PHWR. Most of the lattice calculations were initially performed using the condensed version of the 27-group WIMS library. With the availability of more efficient computers in the 90's, the calculations were made using 69-group library. Later on, more cross section libraries were obtained from IAEA, under the WIMS Library Update Project. These cross section libraries contained 69 or 172 neutron energy groups and were generated from ENDF/BVI.8, JENDL3.2 and JEF-2.2 point data. There is one more library called IAEA library, which is obtained by taking the most recommended data for various isotopes from various point data sets. The computer code CLUB was modified to consider various new libraries obtained from the IAEA. The library generated from ENDF/BVI.8 point data has now been used, for lattice calculations of TAPS-3/4.

A large number of experiments were performed with 7, 19 and 28-rod fuel clusters and different coolants. Only one experiment has been performed with a 37-rod fuel cluster. These experiments were analyzed with the computer code CLUB and the results were generally found to be very encouraging. BARC participated in the IAEA's co-ordinated research program on "In-core Fuel Management Benchmarks for PHWRs" where the results of lattice calculations for 37-rod fuel cluster calculated with CLUB were compared with other international codes. The results of initial criticality obtained during startup of new reactors were also found to be extremely satisfactory.

Control System

For reactor regulation and control, different reactivity devices were provided. There are 17 adjuster rods (ARs) normally kept fully in, during operation and their purpose is to provide positive reactivity, whenever it is required. There are 14 zone control units (ZCUs) with partially filled with water with a capability to provide positive reactivity as well as negative reactivity to control the spatial flux distribution. Normally, the water level in the ZCU is maintained at about 45 % full level (FL). There are four control rods (CRs) normally kept out of the core, which are capable of providing negative reactivity, during operation.

Adjuster Rods

Adjuster rods are kept fully in during normal operation, could be withdrawn in banks (total 21 rods grouped in 8 banks) to add total positive reactivity of about 16 mk (12 mk for half an hour xenon over-ride and 4 mk for compensation of reactivity due to non availability of fuelling machine for about 10 days). On reassessment, it was concluded that, full recovery from a trip (Xenon override from 100%FP) need not be a basis for AR reactivity requirement. Instead, step back for different scenarios was considered from 100% FP to 65% FP and xenon over-ride requirement was worked out to be 8 mk. Thus it was decided to remove 4 out of the 21 ARs. As a result, a new banking scheme for 17 AR's with 8 banks was worked out, which maintains the symmetry of power distribution. Moreover, the location of these rods was chosen to achieve better flux flattening and power distribution.

The ARs do a flux flattening role in this reactor, unlike those in 220 MWe reactors, even though their worth is nearly the same. In 220 MWe thick shells (6 cm OD and WT 1.6 cm) are employed, while in 540 MWe they are thin (7.34 cm OD and WT 1.6 cm) and distributed in three planes, in the center of the core, achieving sufficient flux flattening.

Zone Control Units (ZCUs)

A distributed zone power control is vital for power distribution stability, since this neutronically large sized reactor, is unstable against xenon-induced power oscillations, in the azimuthal and axial directions. The first two higher modes do not have a large eigenvalue separation from the fundamental mode, the eigenvalue of the first azimuthal mode and first axial mode is 15 mk and 22 mk less than the fundamental mode, respectively. Thus seven radial zones each in two axial planes are provided for spatial control. The bulk power as well as power tilt control are performed by fourteen Zone Control Compartments (ZCC) by changing the level of partially filled light water. These ZCCs have been designed to provide ±3.5 mk. The location and length of each ZCC was optimized to achieve efficient and effective control. The response of the reactor to the action of various reactivity control devices, was studied in detail and corresponding S-curves were generated. As a result of this analysis, the location and size of some of the zone controllers were changed to meet control requirements.

Control Rods

The four control rods are always kept out and can be dropped or driven, in for step back or set back respectively, into the core, giving about 10 mk of negative reactivity to compensate for power co-efficient. In indian reactors, their positions have been shifted as compared to CANDU 600 and are located near ZCUs to increase their worth. These four rods are grouped in to two banks.

Shut Down System

As per PHWR safety philosophy, there are two independent shut down systems. The independence aimed are material and geometry of devices, direction and principle of actuation, initiating systems and even hardware and software of the electronics units. This safety philosophy was demonstrated and successfully implemented in 220 MWe PHWRs like NAPS and KAPS. There are two shut down systems (SDS 1 & SDS 2). In 540 MWe reactors these satisfy the safety requirements.

SDS 1

This system has 28 SRs made of SS-Cd-SS shells, which fall under gravity with spring to provide initial acceleration. They give about 72 mk when all the 28 rods actuate. However, for safety analysis, two maximum worth rods are assumed to be not available. Thus with 26 SRs, the worth comes out to be 52 mk. In the indigenous design, position of four rods has been shifted compared to CANDU 600, to improve the total worth.

SDS 2

This system consists of six poison nozzles in the horizontal direction (west to east) through which gadolinium nitrate is injected at high pressure, directly into the moderator. The initial rate of addition is aimed to be similar to the worth of SDS 1 (72 mk) as the total worth of the system works out to be about 300 mk.

Spatial Xenon Control

Xenon spatial oscillation occurs at constant power, due to prompt positive and delayed negative xenon feedback with a period of oscillation of about a day. The problem was tackled both by direct simulation and linear stability analysis. It was found that, xenon stability depended strongly on the core configuration. For the linear stability analysis, an accurate modeling of the reactor core was used and growth factor of instabilities was calculated, for various core states and power. It was also found that only the first azimuthal mode was unstable leading to diverging oscillations, while the first axial mode was close-to-stable with converging oscillations at full power. Below 60% full power, all the higher modes were seen to be stable. The problem of the control of xenon stability by zone control system was also studied in detail, by direct time-domain simulations. And to test the effectiveness and accuracy of the core flux control scheme and the simulation model, a benchmark problem was prepared under the IAEA CRP on core fuel management code packages. It was seen that the instabilities could easily be controlled, by using a simple proportional control algorithm.

Development of COPPS

The large core size has its implications on core safety too. The analysis of various Loss Of Reactivity Control (LORC) incidents showed that, distortions in flux could be highly localized and that the ex-core ion chambers may not be good representatives of reactor power. A reactor of this size needed a safety system such as a Regional Overpower (ROP) protection trip system, that could detect local overpowers. However, in 540 MWe design, partial boiling is not permitted. Hence the requirement of a complicated system like ROP was felt unnecessary and an equivalent simpler system, COPPS (Core Over Power Protection Stystem), was designed. The input to this system is obtained from cobalt SPNDs, strategically located in the core. Operating margins for the more frequently occurring flux shapes were maximized and a scheme for the calibration of SPNDs was worked out.

Instrumented Channels

To get the best estimation of reactor power, forty four instrumented channels distributed symmetrically in the core were identified where channel flow, channel in-let temperature and channel out-let temperatures were measured. The average thermal power is evaluated and fed to the COPPS system, to correct for the bulk power of the reactor.

Online Flux Mapping System

For flux mapping and monitoring the flux shapes in the reactor, 102 vanadium SPNDs have been used. These SPNDs are located in the form of an irregular grid in the core, based on the analysis of correlations between the change in local flux and change in the channel power for a large number of distorted flux shapes in comparison with the nominal flux shape. The detector signals have to be processed on-line, to obtain flux and power shapes in the entire reactor. The required software has been developed on a modal synthesis approach. Large amount of effort has gone into the generation of the fundamental and fifteen higher harmonics. Two different schemes have been developed for this purpose: one based on synthesising the 3D modes for nominal reactor configuration and second for obtaining them from 3D diffusion code, numerically, by an eigenvector elimination process. Once these basic modes are available, some perturbation modes depending upon the possible control RD configurations, estimated directly from 3D simulations and are directly added to the set. An online flux mapping algorithm, based on a least square method was developed and successfully implemented. Indigenously manufactured cobalt and vanadium SPNDs were loaded in the central thimble locations of KGS 1 & 2 and RAPS 3 & 4. Usage of these SPNDs in TAPS 3 & 4 was recommended after studying their performance.

Burnup Optimisation

All the reactivity devices are kept in or get inserted into the zircaloy guide tubes. These guide tubes are anchored to the bottom of the calandria vessel, by SS housing assemblies and employ inconel springs, to keep the guide tubes upright (about 1300 Kg of SS & 190 Kg of inconel). The initial physics optimisation study, carried out with two burnup zones with nominal position of the reactivity devices, (ZCC partially filled and AR's fully in) along with loads due to all the guide tubes, resulted in average exit burn up of about 6800 MWD/Te. The optimisation did not consider the loads due to detectors (SPNDs) and anchors of the guide tubes of reactivity devices in the bottom reflector. The subseqent reoptimisation after including loads due to the anchors of guide tubes revealed that bottom anchors affect the power distribution leading to a top-to-bottom tilt and load of these anchors was estimated to be about 1000 MWD/Te. This leads to the re-examination of the various reactivity requirements from the point of view of control as well as detector requirements. Also, the presence of seventeen adjusters in the core, produces a dip in the center. Therefore, analysis showed that the flux dip in the center due to adjusters and the top to bottom tilt can be corrected only if three burnup zones are considered. The burnup optimisation with three burnup zones resulted in average exit burn up of about 6900 MWD/Te.

Regulatory review

The Reactor Physics Design Division (RPDD) hired by AERB provided a complete regulatory review of TAPS 3 & 4 because of its expertise in the field,

In depth review of DBRs on Physics, Control and Instrumentation and PSAR was taken up and the suggestions made, were incorporated.

RPDD provided technical expertise to NPCIL, in preparing the document "Physics Design Manual" which included the lattice and core parametric studies using the latest IAEA cross-section library ENDF B-VI/6.8.

The different initial fuel loading configurations (which included usage of DU and DDU bundles) were studied and their follow-up till equilibrium is reached, was reviewed.

The experience of the RPDD was utilised in the design, safety review and commissioning of new systems like LZC, MLPAS and SDS 2.

The RPDD physicists played a lead role, in the safety review of the commissioning aspects of both TAPS 3 & 4 during the following activities

- Initial fuel loading
- Calandria filling – bulk addition of moderator (sufficiently poisoned)
- First approach to criticality
- Phase – B Commissioning
- Phase – C up to 50% FP and 90 % FP.

ANNOUNCEMENT

Forthcoming conference

DAE-BRNS Workshop on

Physics and Astrophysics of Hadrons and Hadronic Matter

November 6 -11, 2006

Venue :
Visva Bharati University,
Shantiniketan

A series of workshops have been planned in the area of hadron physics, hadronic matter and its applications to astrophysical systems in order to provide a forum for discussion for researchers working in these fields. The general format of these workshops is (1) lecture series and (2) seminars. The present workshop is third in the series.

Recent developments in the following topics will be discussed:

1. Hadron Physics
2. Nuclear Effective Field Theory
3. Neutron Stars.

Participants can send their applications on plain paper or email giving name, institute and field of interest by August 30, 2006.

Ph.D. students may please send a recommendation letter from their supervisors. Young researchers are very much encouraged to participate, limited financial support towards travel may be available for them.

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ANALYSIS AND DEVELOPMENT OF COMPONENTS, SYSTEMS AND DIAGNOSTIC TECHNIQUES FOR 540 MWe PHWRs AT TARAPUR

Reactor Engineering Division

Coolant Channel Related Development Work

Design and Development of Rolled Joints for Coolant Channel Components

Coolant channels of 540 MWe PHWR consist of a long Zirconium-Niobium alloy (Zr2.5Nb) pressure tube in the reactor core region extended by martensitic stainless steel (SS403) end fittings (Fig.1.). The pressure tube holds the fuel bundles and the primary coolant flows past these fuels removing the heat from the fuels. The pressure tube is surrounded by a zircaloy calandria tube, which separates the cold moderator from the hot pressure tube. The calandria tube is joined to the austenitic stainless steel (SS304L) calandria tube sheets at the ends. The calandria tube and the pressure tube are joined at its ends by rolled joints.

Zero Clearance Pressure Tube Rolled Joint

Appropriate clearance required for assembly were provided in earlier rolled joints between pressure tube and end fitting. However, these relatively large clearances resulted in higher residual stresses in pressure tube in the rolled joint region. These higher residual stresses would possibly cause initiation and propagation of crack by delayed hydrogen cracking under certain circumstances. RED has developed a family of rolled joints commonly referred to as Zero clearance rolled joints. In these joints, the fit between pressure tube and end fitting is from close running to interference. Assembly of these rolled joints will result in residual stresses lower than the threshold for initiation of delayed hydrogen cracking while meeting the requisites of the rolled joint.

RED has developed the zero clearance pressure tube rolled joint for 540 MWe PHWR, as shown in Fig 1. The rolling parameters were optimised by extensive experimental program. The joints were qualified by various tests. A joint sectioned to examine the flow of material to the grooves is shown in Fig. 2.

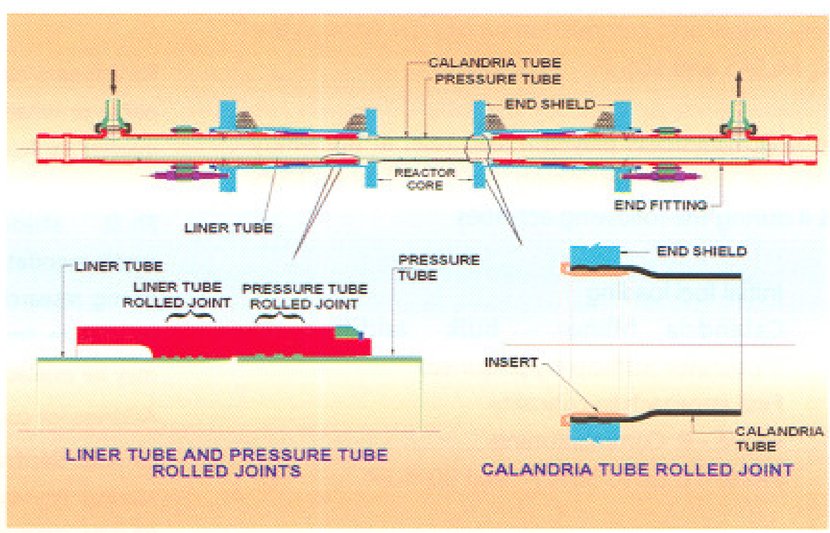

Fig. 1: Coolant Channel of 540 MWe PHWR

Fig. 2: Zero clearance pressure tube rolled joint spool and a sectioned joint

Liner Tube Rolled Joints

The liner tube, which is made from martensitic stainless steel (SS410), is rolled to end fitting by a conventional rolled joint. This joint is subjected to high axial loads during certain accidental conditions. This joint was developed during which different groove geometries were tried and number of grooves were varied and the rolling parameters were optimised. Joint comprising of four trapezoidal grooves, (Fig.1), is selected and percentage tube wall reduction were finalised. These joints were qualified by various tests.

Calandria Tube Rolled Joints

Calandria tube has high diameter to wall thickness ratio (95), where the conventional rolled joint will not work. Landed sandwiched rolled joint is developed to join the calandria tube to calandria tube sheet, where the tube is sandwiched between calandria tube sheet and martensitic stainless steel (SS410) insert, as shown in Fig. 1. Various joint configurations were tried out before arriving at the present configuration. The joint strength in these joints is usually found to be exceeding the tube strength during tests (Fig. 3).

Design and Development of Rolled Joints of other Core Components

The diameter to wall thickness ratio of Liquid Zone Control Rod tube was higher than usual conventional rolled joints but spatial constraints were not permitting introduction of other rolled joints. Rolled joints for Liquid Zone Control Rod and Horizontal Flux Unit, were

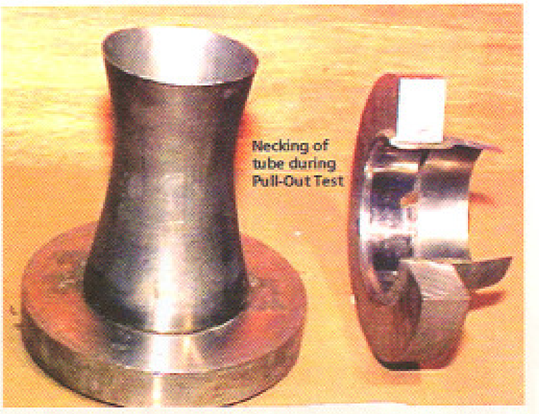

Fig. 3: Pulled out calandria tube rolled joint and sectioned joint

also designed and developed by RED and 17 Liquid Zone Control Rod rolled joints and 15 Horizontal Flux Unit rolled joints were assembled and supplied to NFC for integrating with the rest of the components before supplying to TAPS 3 & 4. A Horizontal Flux Unit Rolled Joint assembly ready for dispatch to NFC is shown in Fig 4.

Rolled Joint Assembly Station and Test Facilities

Rolled Joints are assembled in the Rolled Joint Assembly Station (Fig. 5) and are thoroughly cleaned, inspected and are qualified by a set of non-destructive and destructive tests. Rolled joint parameters such as geometrical and dimensional tolerances of the components, groove geometry, number of grooves and percentage wall reduction, are optimised to meet the

Fig. 4: Horizontal flux unit rolled

joint requirements with minimum residual stresses on the basis of these test results. These test facilities are used for development of other rolled joints also.

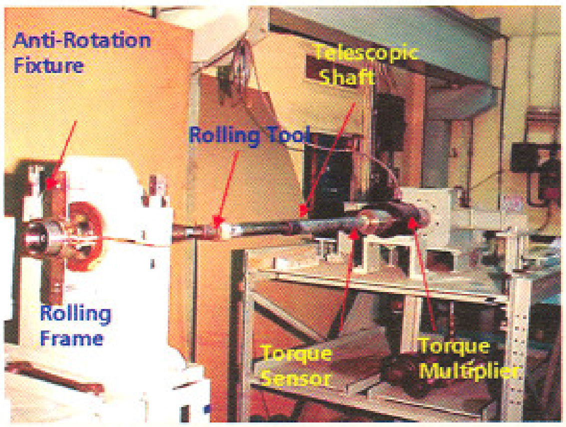

Fig. 5: Rolled joint assembly station

Rolled Joint Assembly Station

Rolled Joint Assembly Station consists of rolling drive - electric motor with reduction gears and torque multiplier, torque and speed sensor, rolling tool, telescopic shaft connecting the drive and the rolling tool, rolling frame where sleeve could be rigidly fixed, and associated supporting structures. It also has fixtures to hold the tube against rotation and axial movements and arrangements to measure the rotation and extrusion of the tube.

Helium Leak Test

All joints are initially tested for helium leak tightness at room temperature. This is done by vacuum method, in which high vacuum (10^{-4} Tor) is created across the joint and helium sprayed into a plastic jacket made around the joint. Helium Leak test at elevated temperature is also carried out in which plastic jacket is replaced with a metal jacket and the gas is cooled to prevent entry of hot gas to the helium leak detector.

Thermal Cycling Test

The rolled joints are subjected to thermal cycling during its life in the reactor. This test is done to assess any degradation on the integrity of the joint. The joints are subjected to a specific number of thermal cycles with intermittent helium leak. Each thermal cycle consists of heating from ambient temperature to operating temperature, holding at operating temperature for a specific period and cooling to room temperature.

Pull Out Test

The joints are subjected to pull out test (Fig. 6) to determine the strength of the joints. During the test, axial loads are applied on the job till the joint slips or the tube fails. High temperature Pull Out test is carried out by externally heating the sleeves to the required temperature before applying the load.

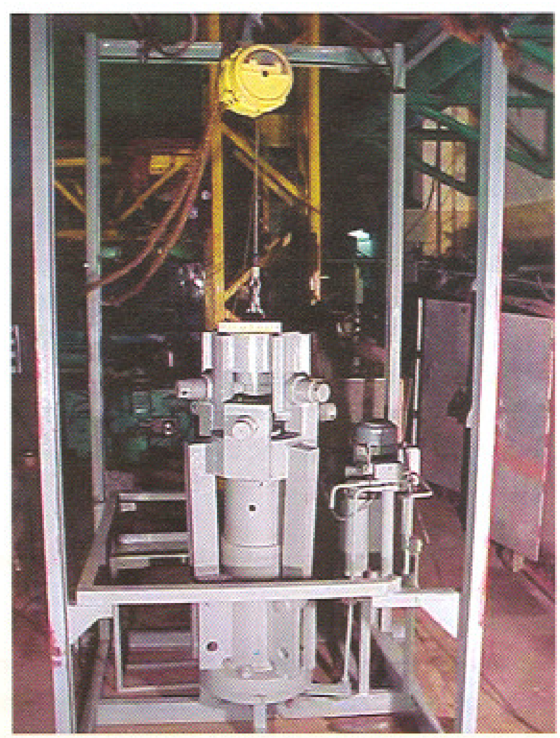

Fig. 6: Pull Out Test Facility

Sectioning Examination

Joint is sectioned to study the flow of material into the grooves, the percentage filling of the grooves and any deformation of the grooves. A sectioned calandria tube and pressure tube (Fig. 7).

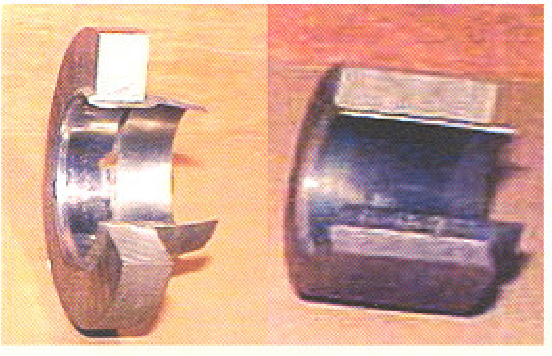

Fig. 7: Calandria tube and pressure tube after sectioning

Coolant Channel Test Facility

Coolant Channel Test Facility for 540MWe was set up at BARC. This facility is being used as platform for testing of fuel bundles, and design and development of ISI tools for coolant channel. Visual inspection of channel was also carried out. The gap between pressure tube and liner tube ends at the rolled joint area and rolled joint area are shown in Fig.8 and 9 respectively.

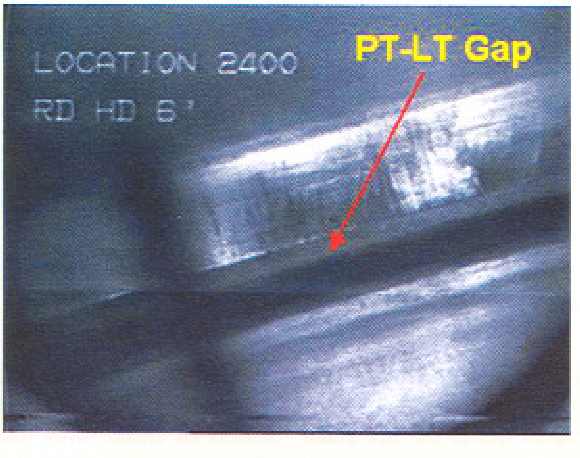

Fig. 8: Pressure tube-liner tube gap

Fig. 9: Pressure tube rolled joint area

Thermal Hydraulic Studies

Fuel Bundle Pressure Drop

Fuel channel pressure drop constitute about 60% of the total pressure drop in the PHT System of 540 MWe PHWR. Accurate knowledge of this pressure drop is essential for the sizing of the primary circulating pumps.The fuel bundle pressure drop data was generated in two different test facilities over the entire range of operation of the reactor. The friction feeder data for the 37-rod bundle is shown in Fig. 10. During loading of fuel bundles into the channels by the fuelling

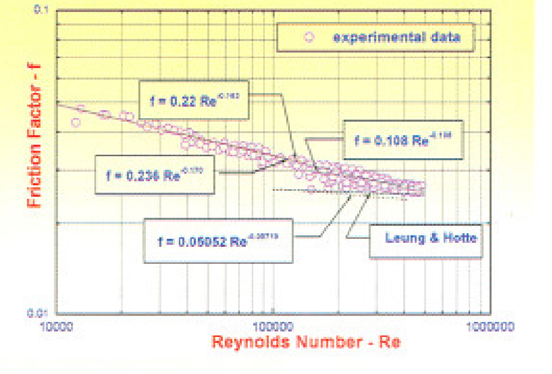

Fig. 10: Friction factor data for 37 rod bundle

machines, no special care is taken to align the bundles. Hence, the bundles can have different junction alignments in the channel (Fig. 11).

Experiments have been carried out in a transparent short channel with two prototype bundles (one fixed and the other rotatable) to measure the effect of misalignment at the junction on the pressure drop. The variation of junction pressure loss coefficient with angle of rotation is shown in Fig. 12.

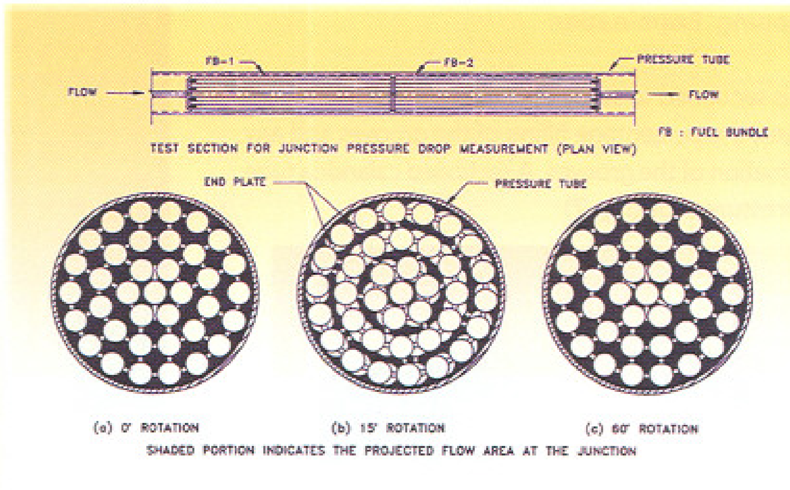

Fig. 11: Views of junction, from upstream of the flow direction for different angles of rotation neglecting bundle eccentricity

Sub-channel Analysis

Thermal hydraulic sub channel analysis is required to ensure that the temperatures in the fuel-channel are within the safe limits. Thermal Hydraulic sub-channel analysis of 37-rod bundle has been carried out for maximum rated channel & channel with maximum bundle power.

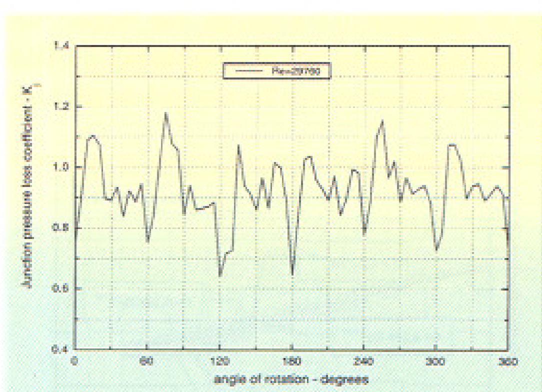

Fig. 12: Variation of junction pressure loss coefficient with angle of rotation

Thermal Hydraulic Study for the Effect of Radial Creep

Experiments have been carried out with channels of inside diameters 103.4 (0% creep), 105 (1.5% creep) and 107 mm (3.5% creep) respectively. Typical result is shown in Fig. 13. To study the effect of radial creep of coolant channel, sub-channel analysis has been carried out for 2.0%, 4.0%, & 6% diametral creep.

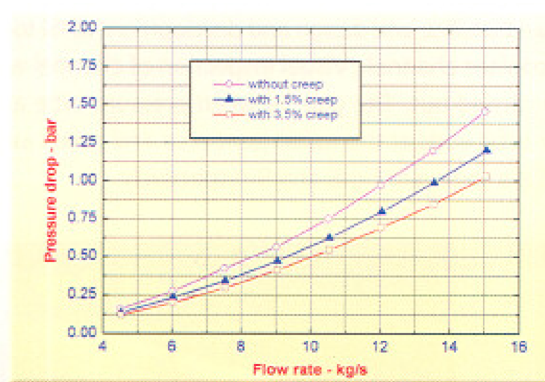

Fig. 13: Effect of radial creep on pressure drop

Calandria Model Studies

Studies on Moderator Inlet diffuser Nozzle

Experiments on full-scale models of moderator inlet diffuser nozzles (Type I and Type III) were performed. The experimental set-up is shown in Fig. 14.

Fig. 14: Set-up to test moderator inlet diffuser nozzle

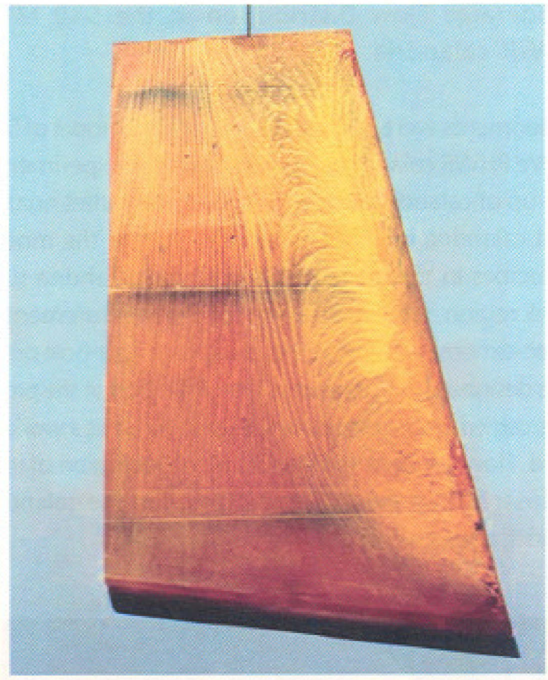

Fig. 15: Flow visualization at the partition plate of moderator inlet diffuser by tracer paint

The local velocities at the exit of the diffuser nozzle were measured. Flow visualization studies were carried out inside the diffuser. Fig. 15 shows the flow pattern near surface of the partition plate of inlet diffuser using the tracer paint. The visualization was also performed in the jet region of the diffuser.

Experiments were also carried out to determine the pressure loss coefficient for both the diffusers. A bottom hole was provided in the diffusers and flow rate through the bottom hole was measured. The velocity in the jet region of diffuser and near the conical skirt of the liquid poison injection tube was measured.

Flow visualization studies were also performed near the conical skirt. CFD simulation to obtain the velocity distribution inside the diffuser has also been carried out. Fig.16 shows the flow distribution inside and at the exit of the moderator inlet diffuser.

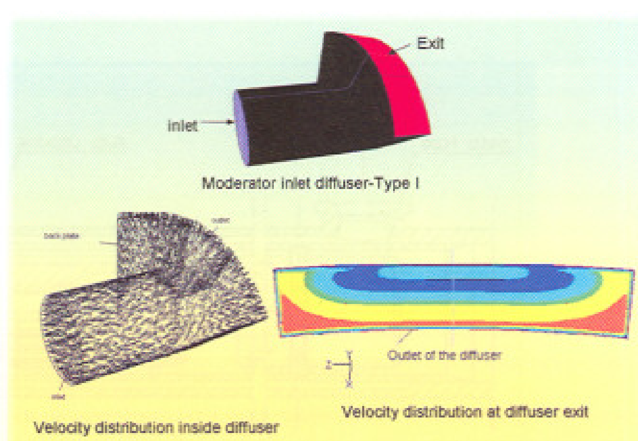

Fig. 16: CFD simulation of 540 MWe PHWR moderator inlet diffuser

Moderator Flow Distribution in the 540 MWe PHWR calandria.

Experiments were performed on a scaled model of 540 MWe PHWR calandria. Fig.17 shows the experimental set-up of calandria model. The inlet and outlet nozzles and calandria internals were simulated in the model. Velocities in the jet region and in the calandria tube bank region, were measured. For the measurement of three-dimensional flow in the jet region, a four-hole probe was designed and fabricated. The calibration of the probe was carried out in a separate set-up using air as a working fluid. Flow visualization studies in the jet region of inlet diffuser and in the tube bank region in the calandria model, were carried out.

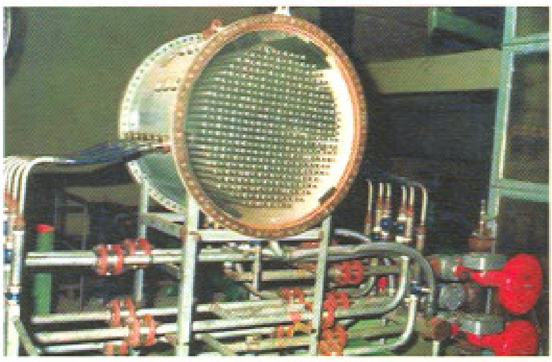

Fig. 17: Calandria model for 540 MWe PHWR

Safety Analysis

Various theoretical analysis and experimental studies have been carried out to ensure the safety of 540 MWe PHWR. A few of the safety studies carried out in RED, BARC are listed below

- Thermosyphon Studies for single phase condition.
- Thermosyphon Studies for two phase condition.
- Flow coast down transient and pump seizing studies.
- Large break LOCA analysis to confirm adequacy of ECCS injection and accumulator capacity.
- ECCS – estimation of recirculation flow and accumulator capacity.

Component Development

Fuel locator studies

A new component (Fig. 18) called the fuel locator has been introduced in the fuel channel of 540 MWe PHWR.

A computer code has been developed to compute the pressure drop performance & the flow distribution in the different parallel flow paths of this component. Fuel locator pressure drop experiments have been carried out for the entire range of operation of 540 MWe PHWR (Fig. 19).

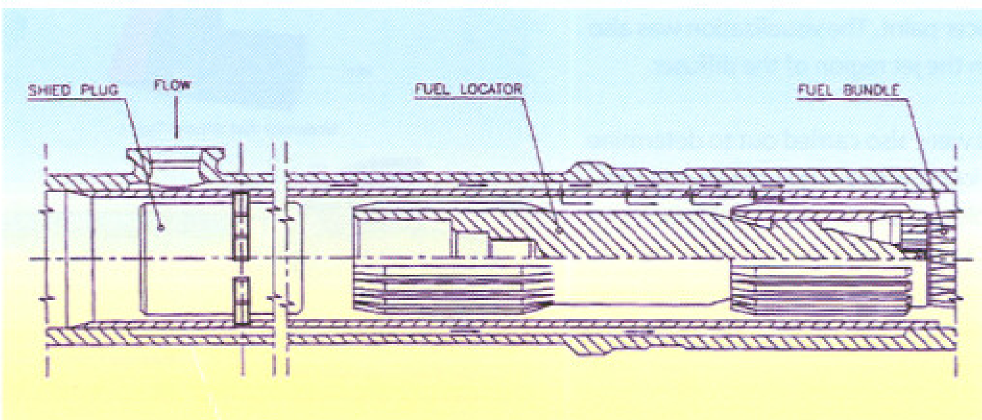

Fig. 18: Simplified sketch of fuel locator at inlet end fitting

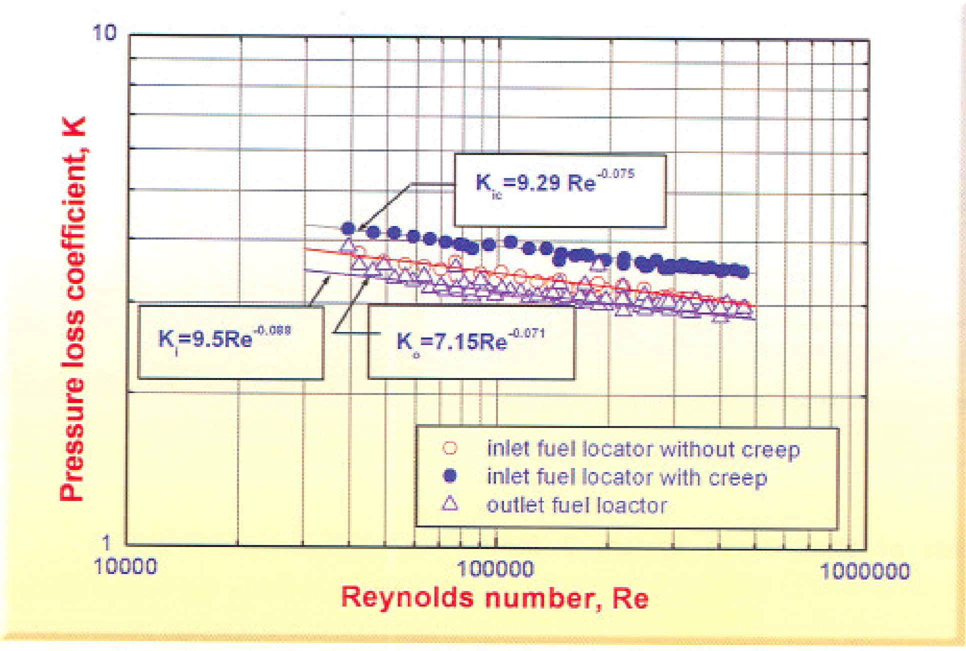

Fig. 19: Pressure loss coefficients for fuel locators

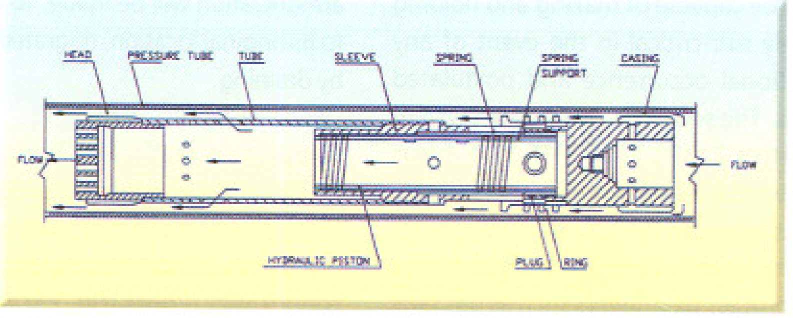

Fig. 20: Schematic of special extension

FARE Tool Development

The flow assisted ram extension (also known as special extension) (Fig. 20) tool is inserted in the 540 MWe fuel channel for refuelling.

A theoretical model has been developed for the analysis of the special extension and the predictions have been compared with the test data generated by RTD for different designs of the FARE tool (Fig. 21).

Pressuriser

Unlike 220 MWe units the heat transport medium in 540 MWe PHWRs design is kept in pressurised liquid state by a pressuriser backed up by feed and bleed control valves. Pressuriser thus becomes an integral part of the 540 MWe PHT system. Thermal Hydraulic Analysis for sizing, design of the pressuriser and dynamic analysis for integral system behaviour were carried out. Procurement assistance was provided for two part tendering. Participated in preparation of various operating procedures.

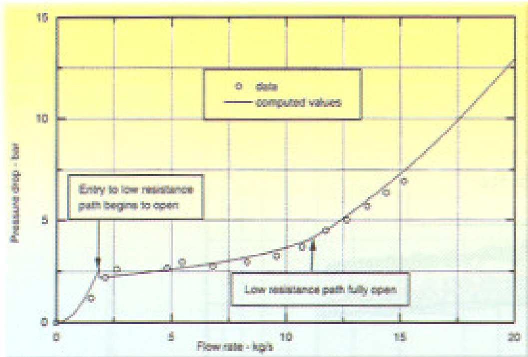

Fig. 21: Comparison of theoretical result with test data

Fig. 22: SDS 2 experimental setup

Developments of Shut Down System 2 for TAPS 3 & 4

The provision of Shut Down Systems is a mandatory requirement for safety of any nuclear reactor. A Shut Down System shall be capable of making and holding the core adequately sub-critical in the event of any anticipated operational occurrence and postulated accident conditions. The secondary shut down system i.e. Shut Down System 2 (SDS 2) of TAPS 3 & 4 was designed and developed at Reactor Engineering Division. Experimental studies were carried out on full scale mock up (Fig. 22) of one injection unit of SDS 2 to qualify the design and evolve process parameters such as gas tank pressure, poison tank level, poison discharge rate, poison injection time and poison jet growth measurements.

Experimental studies on Poison Moderator Interface Movement

In the Shutdown System 2 (SDS 2) of 540 MWe PHWR, there is no physical barrier between the poison and the moderator. Their own liquid in liquid interface separates these two fluids. Under normal reactor operation, the Poison Moderator Interface (PMI) will be at the 65 mm NB isolation ball valve (downstream of poison tank) in the U-bend region. However, interface moves towards calandria over a period of time due to molecular diffusion and also due to physical disturbance in moderator level. In order to monitor the Poison Moderator Interface movement, two online high-pressure conductivity probes are used. On interface movement towards calandria, there will be increase in conductivity and at safe value annunciation will be made. To bring the interface back to its original location, migrated poison will be removed by draining.

In order to meet the SDS 2 requirement of 80 bar rated probe, a developmental program for high-pressure conductivity probe was taken up and two numbers of temperature compensated online high pressure conductivity probes (Fig. 23) were developed.

Fig. 24 shows a high pressure conductivity probe (HPCP) used with a standard precision digital conductivity indicator & transmitter device, to study the interface movement in poison moderator interface setup for SDS 2.

Based upon above mentioned test results, installation scheme for two numbers of high pressure conductivity probes (HPCP) was worked out for SDS 2 of TAPS 3 & 4 Extensive full scale experimental studies have been carried out on poison moderator interface movement under simulated calandria level disturbances. The PMI experimental programme provided conductivity alarm values for TAPS 3 and 4. The study also revealed that there is no significant increase in PMI movement due to low pH poison solution and controlled calandria level disturbances. Based on experimental results, a poison-moderator diffusion model was formulated which truly predicts the change in poison concentration with time along the length of pipe heading towards calandria.

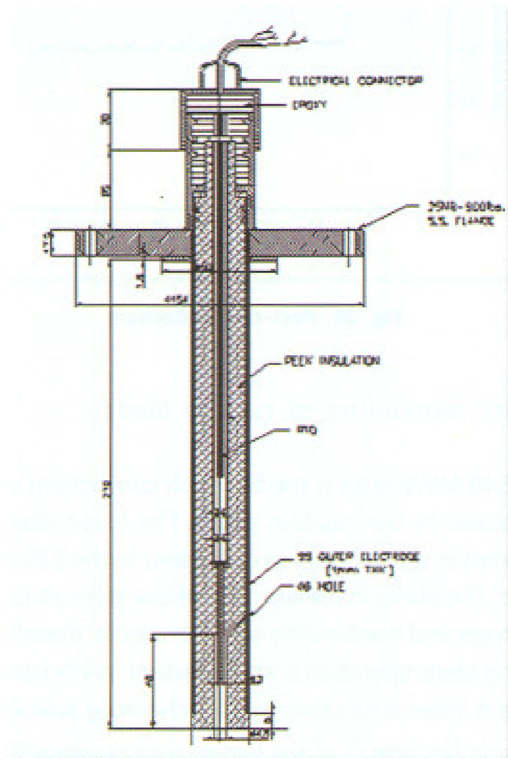

Fig. 23: High Pressure Conductivity Probe (HPCP)

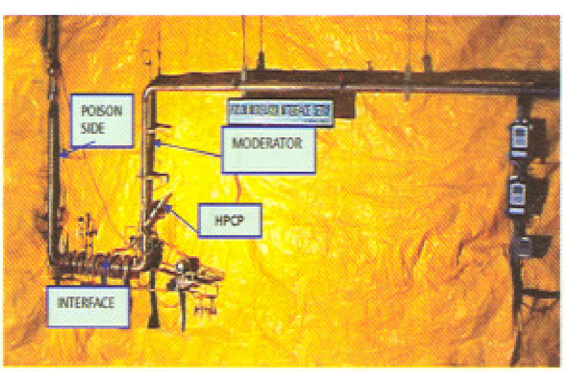

Fig. 24: HPCP under test at poison moderator interface setup

Development of Ultrasonic Ball Detection System (UBDS)

The SDS 2 of TAPS 3 & 4 consists of six poison tanks with their respective poison injection tubes with holes. From reactor safety consideration, the floating ball in poison tank is designed in such a way that it prevents the over pressurisation of calandria. To detect the presence and absence of polyethylene floating ball in poison tank having 22 mm stainless steel wall, a non-intrusive Ultrasonic Ball Detection System (UBDS) has been developed indigenously for TAPS 3 & 4.

Fig. 25: Ultrasonic ball detection system

PHWR Fuel Bundle Strength Test

Compressive strength testing of 37 element fuel bundles, as part of Fuel Type Tests was done, in an experimental set up connected to Integral Test Facility. These tests were carried out at different coolant pressures, temperatures, and support/ load conditions. These test conditions simulated the normal and abnormal conditions that the fuel bundle experiences during the refueling operation by Fuelling Machine. The fuel bundles used for the test were made by Nuclear Fuel Complex. The fuel bundles were pre & post test inspected by Atomic Fuel Division.

The schematic of the test set up is shown in the Fig. 26. The Test Set Up comprises of water hydraulic test rig and an oil hydraulic system. The oil hydraulic system generates the compressive load required for applying on the fuel bundles. The oil hydraulic piston has to overcome the hydraulic pressure inside the water hydraulic rig and then apply the load on the bundle. The end conditions were simulated by keeping the side stop on the blind flange side.

The post-test deflections in the endplate for a typical case are shown in the Fig. 27.

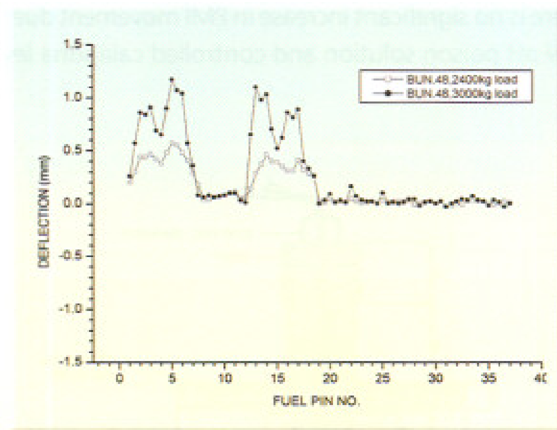

Fig. 27: Post-test deflection

The tests carried out on the fuel bundles showed that they are capable of withstanding the compressive loads expected during its life.

Vibration Diagnostic Technology

RED has provided state-of-art vibration diagnostic technologies for design verification of systems, root cause identification and machinery diagnosis during the commissioning stages in TAPS 3 & 4. These have been part of commissioning document submitted for safety clearance. The important among them are:

Health Monitoring of turbine blades

The 540 MWe rotor is the first such component under operation in our nuclear plant. The rotor design is different in several ways as compared to the 220 MWe rotors. The blades in the last stage of low-pressure turbine are longer and freestanding type. Vibration in these blades during plant operation is an important safety issue. At present, there is no commercial technology available to monitor the health of the blades during operation. In order to fulfill this strong industrial need, RED has developed an advanced diagnostic system for monitoring the health of the blades. The innovative and non-intrusive

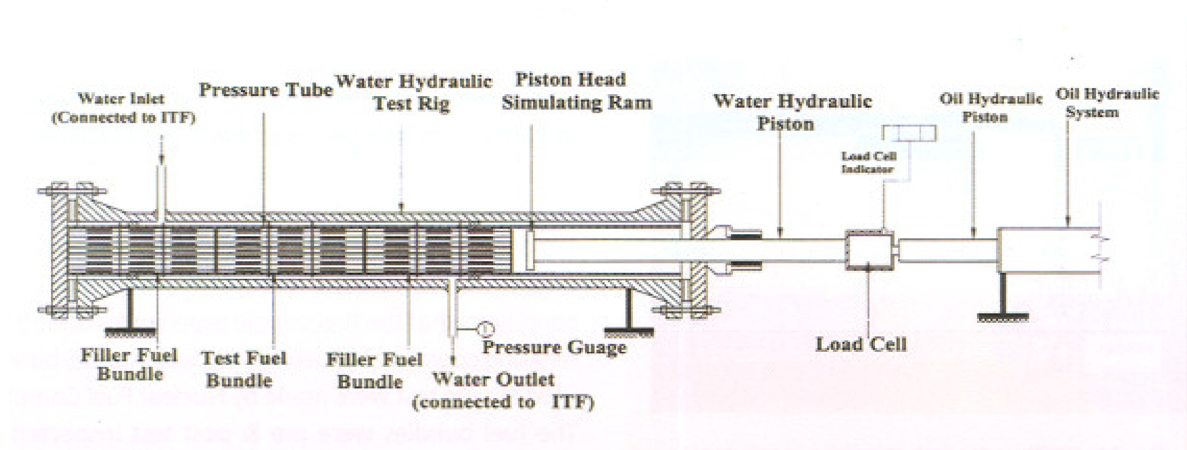

Fig. 26: Fuel bundle strength test setup

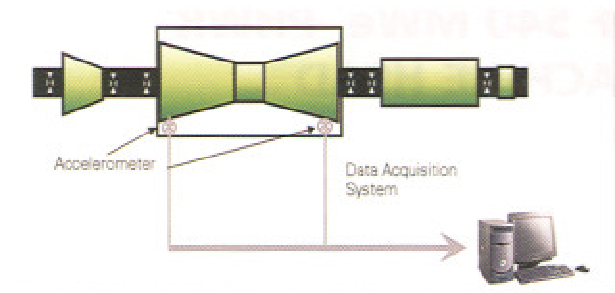

Fig. 28: Schematic of turbine blade monitoring

technique (Fig. 28) was used in TAPS 4 to closely monitor the blade parameters. The Campbell diagram of the last stage blades has been verified and found to be as per design.

Measurement of Poison injection time in TAPS 4

In TAPS 3 & 4, the shut down system (SDS 2) is based on poison injection into the reactor. When need arises, the SDS 2 gets triggered and the entire inventory of poison from the tank gets injected into the reactor. The time of total injection of poison is an important parameter from reactor physics point of view. Hence, it is important to measure the injection time accurately. The injection time was measured by sensing the shock caused by the impact of the synthetic sphere at the bottom of the tank. A data acquisition system was integrated for the purpose and used at the plant. This innovative technique has helped in measuring the injection time accurately. Safety committee has thus cleared the system design.

Pre-operational testing and qualification of nuclear piping

Piping in different systems of the plant carry high velocity fluid during plant operation.The internal fluid pressure tends to deflect and vibrate the piping span at different amplitude depending on the physical boundary condition. For long-term safety of these pipes, it is mandatory to assess the severity of piping vibration to ensure that the cyclic stresses are below the acceptable limits. Important piping like feeder pipe, moderator pipe and the steam lines have been assessed for vibration as per ASME code for nuclear piping.

Elements Fuel Bundle Vibration Measurement

The fuel bundles in the coolant channel are subjected to turbulent flow excitation especially at the entry of the channel. Due to flow-induced vibration, fret related wear in the bundle has been reported earlier by Canadians. In order to develop our own data, flow induced vibration test and analysis was carried out in a flow simulating test set up in RED. The effect of fuel locator on the bundle vibration has been

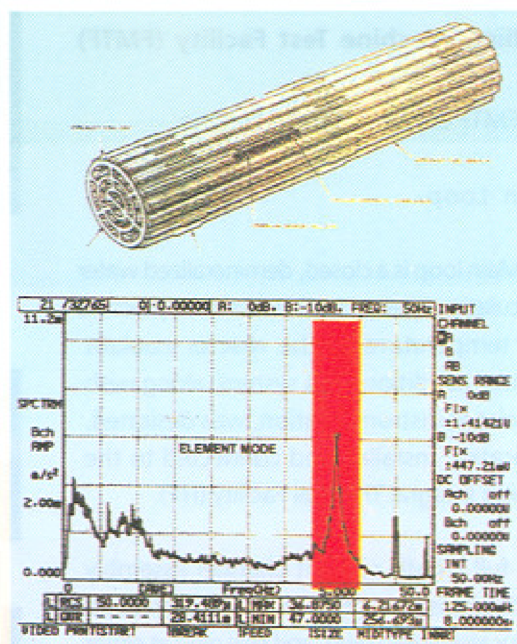

Fig. 29: Fuel bundle and vibration measurement data

closely monitored. It was observed that at a lower flow than rated, the fuel pins vibrate in its bending mode identified to be at 37 Hz. The observation is similar to what is reported from the test carried out originally by the designers.

QUALIFICATION OF 540 MWe PHWR FUELLING MACHINE HEAD

Refuelling Technology Division

The Fuelling Machine of 540 MWe PHWR, is one of the most important and intricate equipment. There have been many changes and modifications in the design of 540 MWe PHWR Fuelling Machine (FM) Head from 220 MWe FM Head. This calls for an elaborate and extensive testing of FM Head and its major sub-assemblies.

It was decided to set up a Ram Assembly test facility and Fuelling Machine test facility at BARC, for testing of Ram Assembly, Fuelling Machines and other R&D activities.

Fuelling Machine Test Facility (FMTF)

The FMTF consists of the following:

Main Loop

The Main loop is a closed, demineralized water recirculation system, to simulate flow, pressure and temperature of the reactor coolant channel condition. This system, along with associated instrumentation, was designed, fabricated, installed and connected to the existing Integral Thermal Facility (ITF).

One full length coolant channel assembly (Fig.1) consisting of thirteen fuel bundles and fuel locator, shield plug, and seal plug at either end was installed and connected to the main loop. Main components like pressure tube, End fittings, and various plugs were provided by NPCIL.

Head Water Supply System (HWSS)

Head water supply system (Fig. 2) supplies water to the Fuelling Machine Head, through valve station and actuator cabinet at required pressure and flow, at ambient temperature, for different operating modes of FM operation. This system is required for FM magazine, and for cooling of various shaft seals, to operate various actuators like Ram-B, Ram-C, guide

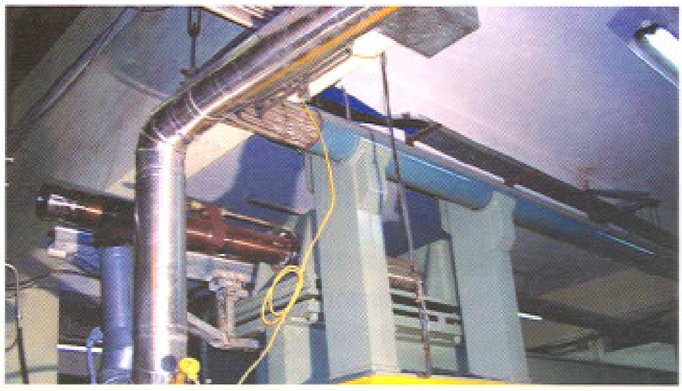

Fig. 1: Coolant channel of 540 MWe PHWR

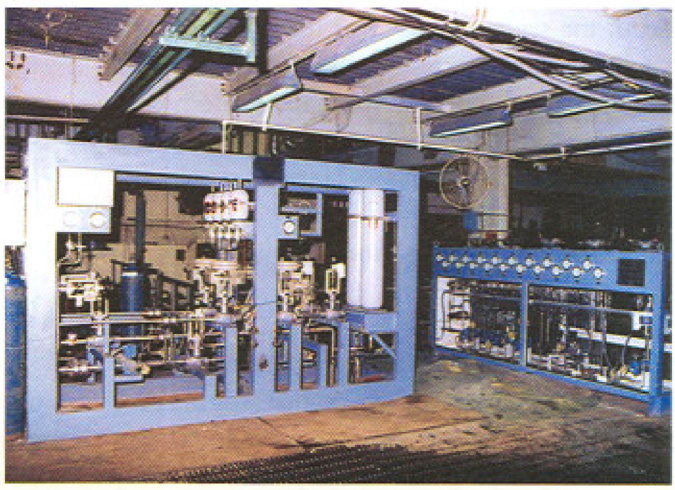

Fig. 2: Water Hydraulic System of 540 MWe PHWR

sleeve lock, snout emergency lock and separator assembly.

The system consists of a Triplex plunger reciprocating pump, control valve station, actuator cabinet, interconnecting piping and tubing and associated instrumentation and control systems.

Oil Hydraulic System

The oil hydraulic system (Fig. 3) consists of a power pack, valve panel and interconnecting high pressure piping, tubing and hose catenaries. It supplies oil at various pressures and flows to control the force, speed, direction and position of various actuators like Ram-B, Ram-C, Latch Ram, Magazine, Snout clamp mounted on Fuelling Machine and X-motion, Y-fine and Z-motion mounted on the Test Carriage.

Fuelling Machine Test Carriage

The fuelling machine test carriage (Fig. 4) has X-drive, Y-fine and Z-drive to align the FM head with the coolant channel. It consists of support columns, bridge assembly, trolley assembly, top beam assembly and upper Gimbal assembly.

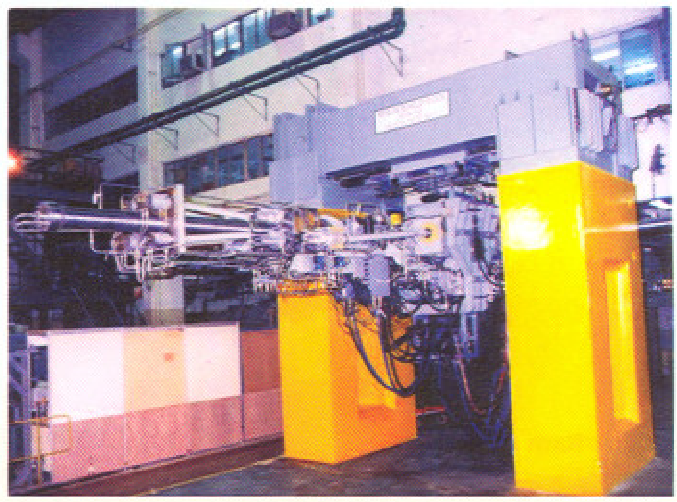

Fig. 4: FM Head and test carriage of 540 MWe PHWR

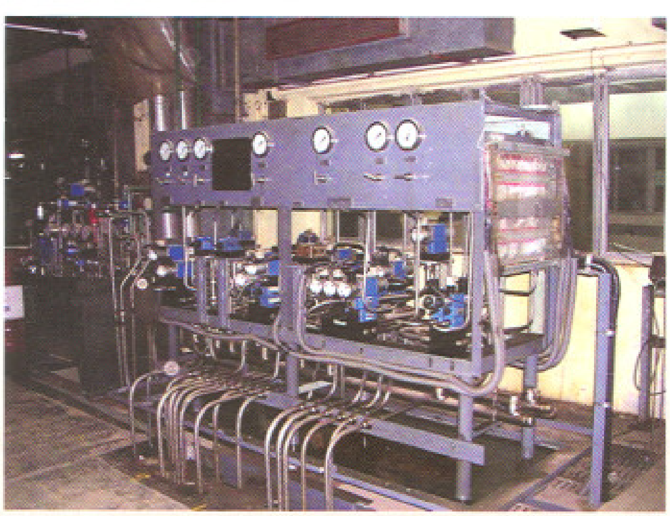

Fig. 3: Oil Hydraulic System of 540 MWe PHWR

Ram Assembly Test Facility (RATF)

A separate RATF was installed, to test the ram assembly (Fig. 5), which is one of the main sub-assemblies of the FM Head. The RATF consists of Ram assembly test rig, end fitting Trolley, oil hydraulic power pack, oil hydraulic system, back up water hydraulic system and associated instrumentation and control systems.

The Fuelling Machine of 540 MWe PHWR reactor is the most important and intricate equipment. Due to many changes in the design of 540 MWe PHWR Fuelling Machine Head from 220 MWe F/M Head, elaborate testing of F/M Head and its major sub-assemblieswas carried out. Ram Assembly is one of the important subassemblices of the Fuelling Machine Head.

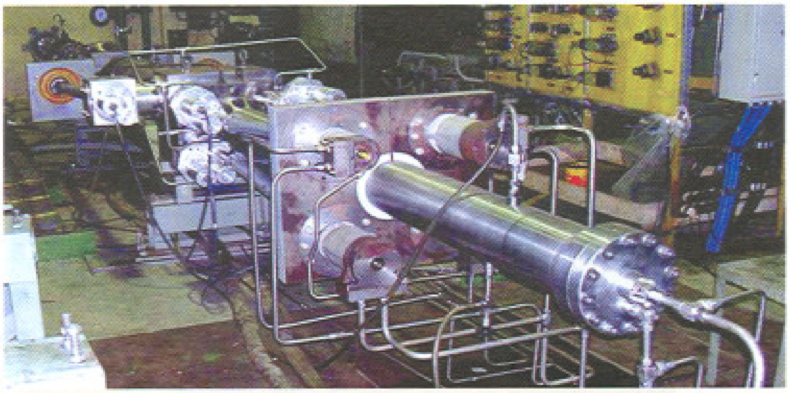

Fig. 5: 540 MWe PHWR RAM Assembly

Testing of Ram Assembly

The Ram Assembly was tested in Ram Assembly Test Facility (RATF). The Ram assembly mainly comprises three telescopic rams; B-Ram, C-Ram and Latch Ram. All these rams are operated through oil hydraulic motors. B-Ram and C-Ram are also provided with water hydraulic back up. These rams are used for refuelling operations and various plug operations.

Extensive testing was necessary to prove the design, to evaluate the life of different components and to specify the required surveillance when it is used for regular refueling in they Reactor. During testing, the following problems were identified, analysed and subsequently rectified and implemented in all the other FM Heads.

B-Ram stalling

The B-Ram was stalling, due to close tolerances between C-ram inner tube and outer tube, between Ram drive body and front housing, between B-Ram Rack Support and Bush and between B-Ram labyrinth and rear housing. This problem was solved by increasing the tolerances and modifying the labyrinth, using piston and piston rings of fiber reinforced polymer material. This system has been incorporated in all TAPS 3 &4 FM heads.

Failure of angular contact ball bearing

Inner race of angular contact ball bearing (147) of Seal Assembly of B-Ram Drive was found to be damaged due to the absence of lubricating oil. To rectify this problem, Gear Drive was modified for filling lubricating oil. This has been incorporated in all TAPS 3 & 4 F/M heads.

C-Ram stalling

C-Ram movement was sluggish and finally stalled. This problem was due to sharp edges of locking finger and low hardness of C-Ram extension tube. Sharp edges of locking finger were removed by rounding the edge. The problem was rectified after chrome plating of C-Ram extension tube.

After implementing all modifications, B-Ram Cycling equivalent to 300 channels refueling operation and plug operations equivalent to 100 channels refueling operation, were successfully carried out, to qualify the Ram Assembly.

The operation of emergency water hydraulic drives of B-Ram and C-Ram was checked and found satisfactory. Free cycling of B-Ram and C-ram and Seal plug/Shielding Plug operation in Magazine and in end fitting equivalent to 100 channels refueling operation was successfully carried out . The operation of Rams was found to be satisfactory.

Testing of FM head

The FM Head was tested in Fuelling Machine Test Facility (FMTF). Fuelling Machine (FM) head comprises of various important sub assemblies namely: Separator assembly, Magazine assembly and Snout Assembly. Since the FM head is also of a new kind of design, extensive testing was required to identify problems and to qualify for "On Power refueling" in the reactor.

During testing following problems were identified, analysed and subsequent rectification of all the problems was carried out.

FM magazine alignment and slippage of clutch

Due to slippage of clutch teeth of magazine drive, the magazine tube was not aligning properly with the Snout. To rectify the slippage problem of clutch, temporarily shims were put and the clutch was permanently engaged. Rigid coupling concept was also adopted in place of clutch in the magazine drive, to rectify this problem.

FM magazine rotation

The rotation problem was due to high torque requirement in unbalanced condition of magazine due to eccentric loading of plugs and bundles. This problem was solved, by increasing the PRV setting of the magazine.

Insufficient stroke of feeler and retractor of separator assembly

Due to insufficient length of separator assembly with respect to bundle position, incorrect feeler in gap feed back was seen. This problem was rectified by reducing the thickness of the spacer, below the separator assembly.

Due to insufficient stroke of Retractor, the fuel bundle was not going completely inside the magazine tube. This problem was rectified by reducing the height of the spring pin head.

Fuel pusher picking up problem from magazine

Due to improper engagement of Fuel Pusher's balls, it was not getting picked up from its defined position. As a short-term solution chamfer of groove of fuel pusher

ANNOUNCEMENT

FORTHCOMING SYMPOSIUM
DAE-BRNS
National Laser Symposium-2006
(NLS-6)

Dec. 5-8, 2006

NLS-6 will be held at the Raja Ramanna Centre for Advanced Technology (RRCAT), Indore. Contributed papers from scientists, engineers and research students (to be presented only as posters) are invited on the following topics:

(1) Physics & technology of lasers (2) Lasers in Nuclear Science & Technology (3) Semiconductor lasers & optoelectronic services (4) Laser materials; devices & components (5) Quantum optics (6) Ultrafast lasers & applications (7) Nonlinear optics (8) Laser spectroscopy (9) Laser plasma interaction (10) Lasers in industry & defence (11) Lasers in Chemistry, Biology & Medicine (12) Laser-based instrumentation (13) Lasers in material science

As part of the symposium, the Indian laser Association (ILA) will organize a two-day tutorial on Dec. 3 and 4, 2006.

Important Dates:

Receipt of manuscript	: Sept. 4, 2006
Intimation of acceptance	: Oct. 3, 2006
Submission of Registration form	: Nov. 6, 2006

For further details, please contact:

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Email: nls2006@cat.gov.in

was modified and the inner diameter of the fuel tube was reduced.

X and Y tilt correction

There was no change in tilt reading for approximately ±5mm offset of FM Head with respect to End Fitting in both directions. This problem is due to flexibility of end fitting, which moves up/down or sideways by 4.6 mm, due to clearance at journal bearing.

This problem was solved by reducing the Pre-load of levelling mechanism to 426 kgf. In this preload condition, response of tilt reading improved, but further reduction was needed to get desired reading. However, to compensate for the imbalance created by B-ram movement, at least 426 kgf pre-load is required. To overcome this problem, an automatic control of oil pressure with respect to position of B-Ram, in levelling mechanism using IRV (Instrumented relief valve) has been developed, implemented and successfully tested.

After implementing all modifications, the performance and acceptance tests, as required, were done in reactor simulated condition.

The Fuelling Machine Head was qualified with successful completion of the above tests.

Conclusion

The fuelling machine head, which is the first of its kind and of a new design, was calibrated and tested in record time, after commissioning of the test facility. Extensive testing of this assembly was carried out in a very short period. Acceptance testing of Fuelling Machine head, at reactor simulated condition was carried out, to qualify the fuelling machine for use in the reactor. Almost all major problems were resolved, so that NPCIL could use the Fuelling Machines at an early stage. As a result, commissioning time of TAPS 3 & 4 could be drastically reduced. Further endurance testing of all sub assemblies of Fuelling Machine Head is being carried out for evaluating long-term use.

ANNOUNCEMENT

Forthcoming Conference

Seventeenth Annual Conference of Indian Nuclear Society

(INSAC- 2006)

Nov. 21-24, 2006

India's growth and sustained development, depends on adequate energy supplies for all its needs. To address this crucial issue, the Indian Nuclear Society has organized the above conference at the Homi Bhabha auditorium, TIFR (Inaugural function) Mumbai and at the Multipurpose Hall (Technical sessions), Training School Hostel, Anushaktinagar, Mumbai. The theme of the conference is "Energy Foresight-India 2050." The conference will have invited talks from eminent scientists, planners, managers, and engineers in the field. Deliberations would be on: 1. Energy Foresight 2. Energy Financing and Investment 3. Environmental Impact 4. Emerging & Advanced technologies 5. Smart energy systems 6. Energy R&D foresight 7. Energy transmission & distribution 8. Energy Management 9. Efficiency Improvements & energy savings 10. Carbon sequestration and New Zero Emission and 11. Foresight for fuels & technologies for the transport sector.

For registration, accomodation and further details, one may contact;

The Convener	The Conference Secretary
Mr S.A. Bhardwaj	Mr R.K. Singh
Director (Technical), NPCIL	Control Instrumentation
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Webpage:http://www.indian-nuclear-society.org.in/conf/index.html

बीएआरसी न्यूज़लेटर
BARC NEWSLETTER

MICROCOMPUTER - BASED FAULT - TOLERANT PROCESS CONTROL SYSTEM FOR 540 MWe PHWRs TAPS 3 & 4

C.K. Pithawa
Reactor Control Division

Reactor Control Division, BARC and Nuclear Power Corporation of India Limited (NPCIL) entered into a Memorandum of Understanding in August 2002. This was for the development of a computer based real-time embedded process control system, for control by Primary coolant pressure, Pressuriser pressure, Pressuriser level, Bleed condenser pressure, Bleed condenser level and Steam generator pressure, for the 540 MWe Tarapur Atomic Power Plants 3 and 4 (TAPS 3 & 4). This is a safety related class IB system. Reactor Control Division (RCnD) has designed and developed the hardware and the software for this Process Control System (PCS), based on Dual Processor Hot Standby (DPHS) architecture. Based on the design information provided by RCnD, Electronics Corporation of India Limited (ECIL) Hyderabad, manufactured and integrated the system hardware (Fig.1). The system was commissioned in TAPS 4, in August 2004 and in TAPS 3, in November 2005. This article describes important features of the system, built in the design, to meet safety and reliability requirements.

Fig. 1: Integrated hardware panels of TAPS 4 : DPHS-PCS

Computer-based control systems for safety-related applications in nuclear power plants have to meet, not only the functional, performance and interface requirements, but also regulatory requirements like enhanced reliability, safety and security. These control systems act so as to maintain relevant plant parameters within set limits and prevent operational transients, leading to accident conditions. They form the first layer of safety in the operation of the plant in a safe manner and minimize the need, for actuation of safety.

These systems must be demonstrated to be safe and reliable with the appropriate level of confidence. Several design features such as fault tolerance, on-line diagnostics and self-supervision are to be incorporated in the computer system architecture, hardware design and software design to meet high reliability and high availability requirements. The software in safety related systems must be demonstrated, not only to be safe but also to have high level of integrity, which is defined as the quality of correctness, completeness, dependability and freedom from defects. Integrity has to be assured by developing software using systematic, carefully controlled, fully documented and reviewable engineering process with Verification and Validation (V&V) activities concurrently performed by an independent team.

DPHS-PCS Architecture

The Dual Processor Hot Standby (DPHS) - Process Control System(PCS) architecture consists of two independent and identical computer systems, System-I and System-II as shown in Fig. 2. Each system is fed with all field and operator's inputs, both analog and contact type. Each of these computer systems, employs two identical processors I-A and I-B, in System-I and processors II-A and II-B, in System-II. Each processor has its own local memory and watchdog timer. Each pair of processors shares common Input/Output (I/O) boards for all contact and analog signals. Sharing of common I/O boards is made possible, through the use of a bus selector board.

Both processors in each pair, perform all the control and logic functions, in an identical manner. In addition to this, exhaustive on-line diagnostic checks on memory, bus and I/O boards, are performed. Each processor of the pair compares its own computed output signals, with the corresponding computed output signals of the other processor, through dual port RAM and generates diagnostic information, based on any detected mismatches. If the diagnostics programs and/or any of the two watch-dog timers, detect conditions, that indicate that the system is faulty, then the control output connections to the control elements are automatically switched, from the faulty system, to the healthy standby system, through the Control Transfer Unit (CTU), built entirely with electromagnetic relays.

During normal operation of DPHS-PCS, the System-I gives control outputs, to half of the control elements, for which, hot standby is available from System-II. The other half of the control elements get their control output signals normally from System-II, while the hot standby for these outputs, are available from System-I. This arrangement reduces the impact of signal interruption, during the short transition period of a few milliseconds, whenever the entire control is transferred to one computer system. The design also provides additional feature, to transfer entire control on to one system through the operator's action, so that, the other system can be taken for offline test/maintenance.

To aid display of relevant information from the DPHS-PCS on the control panel as well as on the operator's console, two 6.4" panel-mounted video display units, (each driven by its own PC-based display computer) and a PC-based console are connected to the control computers, through a dual redundant Ethernet network as shown in Fig. 2. This enables efficient man-machine interface, without additional load on the control computers. The two control computers also exchange relevant information over the same network. Information is passed on, from the DPHS-PCS to the Computerized Operator Information System (COIS), through display computers, over a separate Ethernet network.

On-line diagnostic programs perform Finite Impulse Testing (FIT) on digital input boards, read-back of energization status in relay output board, testing of Analog to Digital Converters (ADC) with reference voltages and testing of analog output boards by reading back output signals. Cycle Redundancy Check (CRC) on Electrically Programmable Read Only Memory (EPROM) is also performed, to check the integrity of the control software. On detection of any fault, relevant diagnostic message is transmitted to the PC console.

Fault-Tolerant Features

The fault-tolerant features built in the DHPS architecture are as follows:

- A single sensor or input board failure, does not adversely affect, the control performance of the system. The triplicate input signals (digital and analog) are distributed in different input boards and representative signals, derived from healthy inputs, are used for process control. Such failure is always brought to the operator's notice through the PC console.

- The Watch Dog Timer (WDT) of each processor, monitors software flow and detects gross failures of the software.

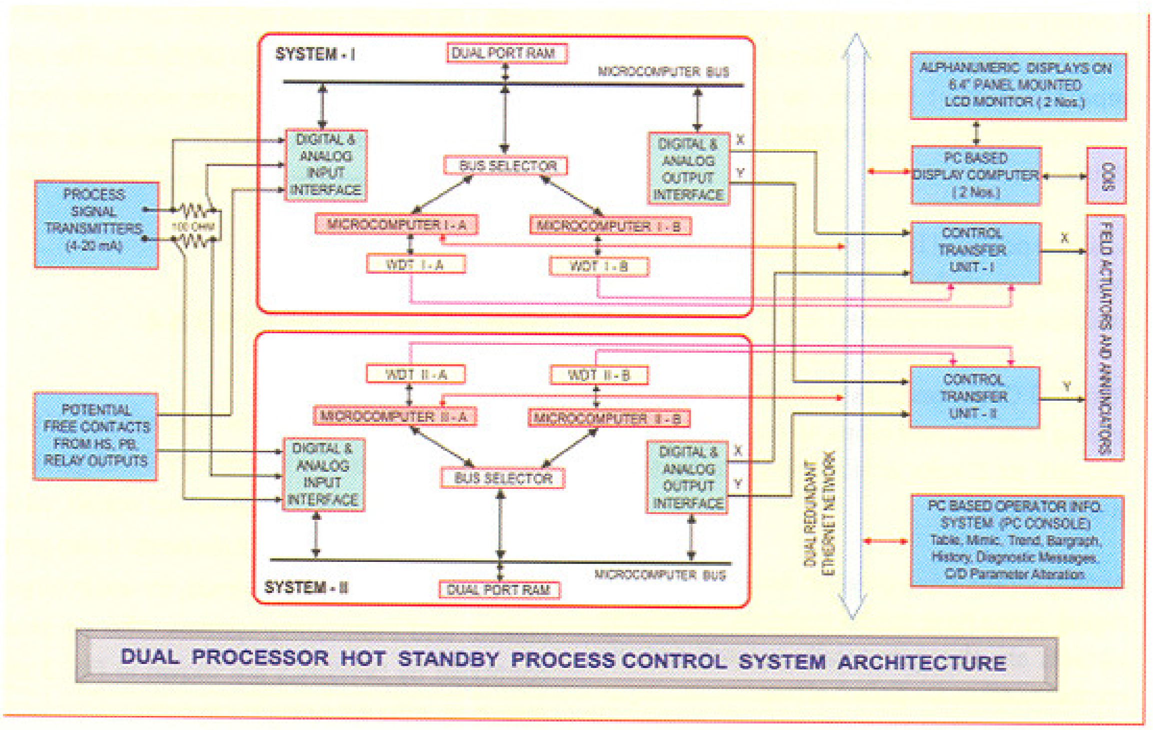

Fig. 2: Processor Hot Standby architecture

- The hardware modules have been designedwith features, to facilitate on-line diagnostics, that enable the system to detect faults, during run-time.

- The design provides redundancy in DC power supplies, AC poer connections, PC-driven panel mounted display units and dual redundant network.

- The alterable control and design parameters, are stored in lectrically Erasable Programmable Read Only Memory (EEPROM). Following every change in any of these parameters through the PC Console, the contents of these EEPROMs are updated. After every power on' of the system, these parameter values are equated to the corresponding one, from the other healthy operating control computer system, obtained through network communication. However, if the other system is yet to be started, then, the system reads the values of these parameters from its own EPROM, during the execution of initialization of the software. The current values of the control and design parameters can be viewed on the PC Console.

- On 'Power ON', the version number of software in EPROM, residing in each of the processors in the system, is compared. In case of a mismatch, further execution of program is stopped and a specific character is displayed on the processor board, facia panel, to indicate the mismatch. The version of software installed in all the four processors along with the software version of the PC console, display computer, network software and COIS interface software can be viewed on the PC Console.

- Control outputs to control elements with identical functions, are distributed among separate output boards, to minimize the consequence of a single output interface board failure.

- Separate potential free contact inputs from hand switches, push buttons and relays are made available for both Systems I and II.

- Two 100 Ohms resistors are used in 4-20 mA analog input current loops, to provide separate 400 mV to 2 volts analog inputs to System I and System II.

- Analog inputs have galvanic isolation and the digital inputs have optical isolation. The analog outputs are also galvanically isolated and the contact outputs are potential free.

- In case of fault in the control computer, the analog control outputs to control elements, are either frozen to values just prior to the fault in the computer system or forced to a pre defined value, based on the process requirement.

- Since the switchover unit comes as a single element in series with other redundant components, the overall system reliability is very sensitive to any possible failures in this unit. Hence its design is made simple and construction is rugged using good quality relays.

Operating Control Modes

Besides auto control mode, the DPHS-PCS provides the facility to select the control element/control valve, to be operable in both computer manual mode and external manual mode. When a control element is in computer manual mode, DPHS-PCS generates the corresponding ontrol signal, based on the operator's action, on the open/close push buttons for that element. In external manual mode, the control valves get their control signals from hand controllers, external to DPHS-PCS. The software design ensures bump less transfer, whenever the control mode is changed from external manual to computer manual and vice versa or, from computer manual to auto and vice versa.

DPHS-PCS System for TAPS 3 & 4

Each of the control computers receives 69 field analog inputs and 185 contact inputs. DPHS-PCS generates 26 numbers of 4-20 mA control output signals, which are passed on to the final control elements. It also generates 19 potential free contact outputs, for use in other plant control and monitoring systems. The six processes controlled by DPHS-PCS are shown in Fig. 3 and are described in the following paragraphs.

The steam generator pressure controller, regulates the steam generator (SG) pressure, by controlling the total amount of steam drawn from the SG, by action on the valves in various steam lines, which take the steam to the turbine or dump it into the condenser or discharges it into the atmosphere. The pressure measurements from four SGs and the primary coolant differential temperature across the SG, is made use of, to provide programmed variation of SG pressure set point with plant power output. Steam Generator Pressure Control (SGPC) system outputs are passed on to the Turbine control system and the Atmospheric Steam Discharge Valves (ASDV). The Turbine control system controls the steam turbine governor and the Condenser Steam Dump Valves (CSDV).

The primary coolant pressure controller, maintains the average reactor outlet header pressure around 100 kg/cm^2 irrespective of reactor power level and process transients. This is achieved, by regulating the opening of two pressuriser steam bleed control valves and ON/OFF control of eight pressuriser heaters, when the pressuriser is in service. When the pressuriser is isolated, the primary

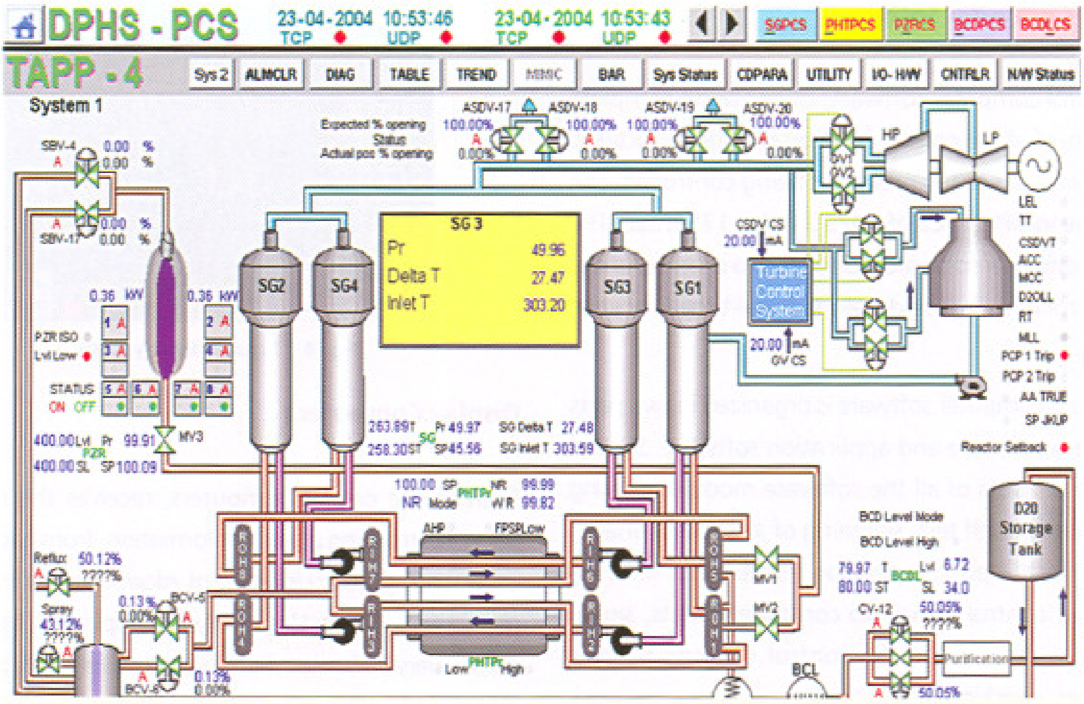

Fig. 3: Mimic showing 540 MWe PHWR processes controlled by DPHS-PCS

coolant pressure is controlled by regulating the opening of two feed control valves and two bleed control valves. At the time of primary coolant pressurization and depressurization, the set point changes over a wide range. In the 'Automatic Hot Pressurization' (AHP) mode, the set point is varied automatically as a function of Reactor Outlet Header (ROH) temperature, by selecting the 'AHP mode', through mode selection hand switch. The controller gain is varied non-linearly over the primary coolant temperature range to account for non-linearity in D_2O compressibility.

The Pressuriser level is required to be controlled only when the pressuriser is in service. The pressuriser level is controlled at 332.5 cm by regulating the opening of feed and bleed control valves. When the pressuriser is isolated, the pressuriser level is used for indication only and pressurizer pressure is controlled at 100 kg/cm^2 by pressurizer steam bleed valves and pressurizer heaters.

The Bleed Condenser (BCD) is meant for condensing the flashed steam from the PHT circuit through the bleed valves. The condensation is achieved by circulating colder D_2O through tubes inside the BCD (reflux control) and/or spraying colder D_2O into the BCD (spray control). The bleed condenser pressure is controlled at 34 kg/cm^2 by regulating the reflux flow and the spray flow through the reflux control valve and the spray control valve respectively.

The D_2O bled from PHT system into the BCD, is matched by an outflow of condensed D_2O, from the BCD to D_2O storage tank. The BCD level is maintained at 38.5 cm, by controlling this flow by means of level control valves. For Bleed Cooler Outlet Temperature (BCOT) less than 80°C, the valves are controlled, based on the BCD level measurement and for BCOT greater than or equal to 80°C, the valves are controlled,based on the BCOT measurement.

Control Computer Software

The control computer software, must always complete execution, of all the control function on time, as dictated by the dynamics of the processes being controlled. The cycle time in DPHS-PCS of TAPS 3 & 4 is 175 msec. The control computer software is designed to execute, under the control of in-house developed real-time Executive RT-XINU.

The control computer software is organized in two parts viz. system software and application software. System software consists of all the software modules dealing with power on self test, scanning of all process analog and contact inputs, operator inputs and sending computed control signals to control elements, signal validation, comparison of control outputs by two processors, checking of switchover conditions, network communication, alteration of control and design parameters and on-line hardware diagnostics. The system software is written in 'C' language.

The application software implements the process controller blocks, carrying out the required computations, to generate the control signals, as per the desired control algorithms. A controller accepts a set point and a measured quantity as inputs, performs control signal calculations following any combination of Proportional, Integral and Derivative algorithms and gives out the control signal.

Human Machine Interface (HMI) on Display Computers and PC Console

The information of the controlled processes is provided to the operator, through two panel mounted display units, each driven by its own PC-based display computer. In addition to this, a PC console is provided on the operator's console. The HMI on display units and the PC console are developed, using commercially available, off-the-shelf, Windows-based HMI package.

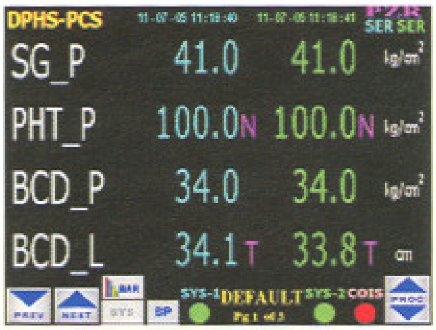

Fig. 4 : Typical display page

Display Computers

Each of the display computers, receives the network packet containing process information, from each of the control computers. Updating of information on the panel-mounted 6.4-inch Liquid Crystal Display (LCD) units is done, every 500 msec. Each panel mounted display unit, is provided with a touch screen interface. Both System-I and System-II data are displayed simultaneously. The two display units display the information in tabular or bar-graph format. Multipage display of parameters is provided. The display pages are organized based on the process. A pair of soft buttons provided at the bottom of each display page, allows the operator to select a process of interest. With another pair of soft buttons, the operator can surf through different pages of the selected process.

A soft button is also provided so that the operator can toggle between the bar-graph/tabular formats of display. The healthy status of network communication with control computers and network link to COIS gateway, is displayed in the form of soft LEDs, Relevant information of DPHS-PCS is also passed on to the COIS, through the display computers. Fig. 4 shows a typical display page of the display unit.

PC Console

The PC console, receives the network packet, containing process and system information, from each of the control computers. Updating of information on 21 inch monitor of the PC console, is done every second. Nearly 3000 tags of the HMI package database, are programmed to manage information of all the processes, controlled by DPHS-PCS. The PC console screen is divided into three parts such that the menu buttons are displayed at the top, the latest three alarms/messages are displayed at the bottom and the rest of the area is used for information display. Using the sub menu buttons, the operator can monitor the parameters of a selected process in different formats such as tables, mimics, bar graphs, real time and historical trend. Sub menu buttons are also provided to display controller details and control and design parameters of the selected process. The operator can also monitor detailed network status, system status and diagnostics messages with the help of related sub menu buttons.

The PC console provides a multilevel password security mechanism such that, different privilege levels are assigned at every level. Utility menu provides several important functions such as history and diagnostics data transfer to removable media, control and design parameter change, help, viewing of logged information, security functions like login, logout and password change. The PC console also maintains a log file, in which all the transactions carried out by the operator in the process of Control Parameter (CP)/Design Parameter (DP) changes, are stored.

The PC console decodes the diagnostics messages received from control computers and generates a relevant message string for display and logging. It has provision for logging the last 100,000 diagnostics messages. A utility is provided, to transfer these diagnostics messages onto a removable media, so that, off-line analysis of the diagnostics messages can be carried out.

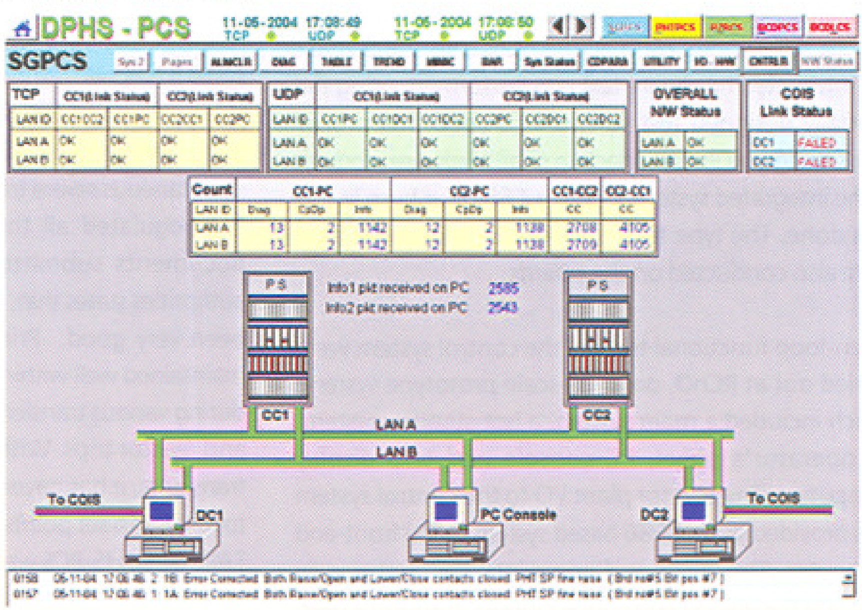

Fig. 6 : Detailed network status display on PC console

Data Communication through Network

The two control computers and the three PC-based nodes, are networked over a dual redundant Ethernet network, using optical fiber at physical layer. Redundancy in the Ethernet network is achieved, by using two network cards at each node, connected through redundant media. The network interface at control computer is realized, with RCnD-designed Intelligent Ethernet Module (IEM), along with an in-house developed sub-set of TCP/IP stack on it. On PC-based nodes, commercially available Ethernet network interface boards backed by the TCP/IP stack provided by Windows operating system are used, for network connectivity.

Control Computer Hardware

A family of microcomputer boards, with modular construction in double euro format and providing a variety of functions, was developed in RCnD. It is based on RCnD proprietary bus, for fault-tolerant, real-time control and monitoring applications. These microcomputer boards are used, for realizing DPHS-PCS. The Printed Circuit Boards (PCBs) are in the form of plug-in modules.interchangeable. The boards have been designed with features, to facilitate on-line diagnostics. Components are adequately de-rated, to increase system reliability. The layout and artwork of PCBs conform to Quality Assurance (QA) requirements of the Nuclear Power Corporation of India Limited. The reliability of microcomputer boards was assessed, based on MIL-STD-217/E. The assessed failure rates of microcomputer boards used in DPHS-PCS, lie between 0.02 to 12.21 failures /10^6 hours.

System Testing

RCnD prepared a detailed test plan and test procedure for testing integrated hardware/software and integrated system testing. The test on the integrated system, ensures compliance with functional, performance and interface requirements.

All the Printed Circuit Boards (PCBs) were individually tested, before system integration, on automated board facility. After integrating all the PCBs into the system, the integrated hardware test was performed, using the specially designed test software to check the proper functioning and interconnection of all hardware modules in the integrated system. Finally a 168-hour burn in test was done. The type tests for dry heat and damp heat were also conducted on the boards.

Open-loop functional tests of the control system were carried out at RCnD, on a full-scale prototype system, which included a main system, a hot standby system, an operator's panel, PC console and two display computers. The reactor plant I/O to the control system was provided by an 8086 based system called front-end computer. This front-end computer has features to simulate the analog and contact inputs, through a user-friendly interface, provided on a PC connected to it, over a serial link.

Documentation, Verification and Validation

The DPHS-PCS software has been developed as per Software Quality Assurance Plan following AERB SG/D-25 guidelines. At each stage of development cycle, appropriate documents were generated and subjected to independent review for verification. The documents generated during this process are : System Requirements (SR), System Architecture Design Document (SAD), Software Requirements Specifications (SRS), Software Architectural Design Document (SWADD), Software Detailed Design Document (SDD), Hardware Design Document (HDD), System Test Plan (STP), System Test Report (STR), System Verification and Validation Plan and Report (SVP and SVR), Hardware Installation and Commissioning Procedure, Software Installation and Commissioning Procedure, System Build Document and User Manual. Software verification and validation techniques are used, to assure a high level of software reliability, through protection against software design Errors.

Conclusion

The TAPS 4 DPHS-PCS system was commissioned in August 2004. The system was successfully used during hot-conditioning and pressurizer performance tests. The system has performed exceedingly well during several stages of power operation of TAPS 4. During load throwoff tests carried out at 50% and 90% full power, creating simultaneous severe transients in all the processes, DPHS-PCS regulated all the six processes very well. The documents submitted to AERB by TAPS 4 station authorities states that, the performance of DPHS-PCS has been very good. Primary coolant pressure has been maintained well within the small range of 100 kg/sq.cm. during various transients like turbine trips, load rejection and reactor trips. With the system experiencing several transients, it has never shown any indication of tending towards trip set points of PHT low or high pressure. The TAPS 3 DPHS-PCS was also successfully commissioned in November 2005.

बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

DESIGN AND DEVELOPMENT OF DRIVE MECHANISMS FOR ADJUSTER RODS, CONTROL RODS AND SHUT-OFF RODS OF TAPS 3 & 4

Manjit Singh
Division of Remote Handling & Robotics

BARC and NPCIL had entered into an MoU, for the design and development of reactivity control mechanisms for adjuster rods, control rods and shut-off rods for TAPS 3 & 4. This development work was taken up at the Division of Remote Handling & Robotics (DRHR), BARC.

Reactivity control mechanisms of TAPS 3 & 4 are electromagnetic type with sheave rope arrangement; however, design is significantly different from that used in Dhruva and 220 MWe PHWRs. The control mechanisms for TAPS 3 & 4 are designed with a number

Photograph showing reactivity control mechanisms on desk plate of TAPS 4 reactor 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 and re-arresting of control rods for reactor stepback function.

TAPS 3 & 4 utilize 28 shut-off rods (total rod worth 72 mk), 4 control rods (total rod worth 4 mk) and 17 adjuster rods (total rod worth 17 mk), total 49 drive mechanisms per unit. Shut-off rod drive mechanisms (28) form part of the Shut Down System SDS 1. The principal purpose of this system, isto provide shut down capability to the reactor, under normal as well as under any undesirable condition, including prolonged shut down. Four control rod drive mechanism forms part of the reactor regulating system. These are used for rapid power reduction if required (reactor step-back function), to bring average zonal compartment level into normal operating range and to compensate for reactivity gain, following transition from full power to hot standby condition. Seventeen adjuster rod drive mechanisms form part of reactor regulating system. These are used for flattening of neutron flux distribution, Xenon over-ride and reactivity shim (during extended fuelling machine outages).

In case of shut-off rod/control rod, total rod travel is 6600 mm while that for adjuster rod, it is 5900 mm. Weight of absorber rod for shut-off rod/control rod is about 50 Kgs. while that for adjuster rod, it is about 15 Kgs. Rod withdrawal time for shut-off rod/control rod at maximum speed is 150 ± 10 sec, while that for adjuster rod (at maximum speed), it is 70 ± 10 sec. Reactivity addition rate is limited to 0.33 mmk/sec for shut-off rods, while it is limited to 0.10 mk/sec, for control rods/ adjuster rods.

The prototype control mechanisms were manufactured, assembled and subjected to design qualification and life cycle testing, on a full-scale test set up, at BARC. Life cycle testing for shut-off rod drive mechanism was carried out, with tank water temperature maintained at 80°C and with elevated temperature of 65°C at mechanism location. Prototype shut-off rod drive mechanism was tested for full drop for more than 6000 cycles and this was also tested for on-line testing, for more than 3000 cycles (rod was released for three times in one cycle, establishing on-line testing for more than 9000 times). The prototype control rod drive mechanism was tested for partial release tests (for 48% travel) for more than 5000 cycles. The prototype adjuster rod drive mechanism, was tested for more than 8000 cycles (motorized up/down movement). The design was qualified based on satisfactory performance and life cycle testing of prototypes at BARC. All the design drawings (general assembly, sub-assembly and component level), technical specification, development activity-wise performance test reports and life cycle test reports were issued to NPCIL. As per the design and drawings supplied by DRHR, BARC, manufacturing of total 120 drive units for reactor use, was arranged by NPCIL. Manufacturing was done at M/S Godrej and M/S L&T (60 drive units each).

Installation, commissioning and testing of all the drive mechanisms for both the units at TAPS 3 & 4 reactor have been completed. The photograph shows control mechanisms installed on the thimble above deck plate elevation of TAPS 4 reactor.

बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

PREPARATION OF GADOLINIUM NITRATE FOR PHWRs OF TAPS 3 & 4, FOR USE AS CONTROL MATERIAL

S.L. Mishra and H. Singh
Rare Earths Development Section

Gadolinium nitrate (Gd $(NO_3)_3 \cdot 6H_2O$) is being used, for the first time in India, in 540 MWe PHWR at Tarapur. TAPS initally required ~150kg gadolinium nitrate. It is preferred over the generally used element Boron, due to high neutron cross section and it works efficiently for reactivity control, through Moderator Liquid Poison addition System (MLPS) as well as for reactor Shut Down System (SDS 2). Low concentration of gadolinium (0.1-0.2 g/l) in heavy water, is sufficient to shut down, in a very short time, for which, other wise large quantity of Boron would be required. After use, the small amount of Gadolinium can be separated quickly from heavy water, by ion exchange process. For such use, a critical technical specification for Gadolinium nitrate is prescribed, which includes Na:100 ppm, Mg:50 ppm, Cl:50 ppm, other REs :900 ppm and isotopic composition of natural Gadolinium. The process used by REDS, BARC, for production of Gadolinium nitrate, consists of

The REDS team

i) production of high purity Gadolinium oxide (99.9%), ii) dissolution of oxide in nitric acid, iii) evaporation of the clear solution till the temperature reaches 130°C, iv) cooling of solution with constant stirring for formation of fine crystals, v) sampling, analysis and packing.

Purification of gadolinium oxide

In India, Gadolinium oxide is recovered from monazite, which contains 1.5 % Gd_2O_3 of the total rare earths oxide (REO). Purification of Gd involves both solvent extraction and ion exchange processes. IREL processes crude mixed rare earth chloride, using 2-ethyl hexyl 2-ethyl hexyl phosphonic acid (EHEHPA) to obtain i) lighter rare earth concentrate (LRE) La to Nd, ii) Middle rare earths concentrate (MRE) Sm-Eu-Gd and iii) heavy rare earths concentrate (HRE) Dy to Y. During processing of MRE with EHEHPA for obtaining Sm (>97%), a Gd concentrate (60-70%) is also produced Table 1. At present, a process has been developed in REDS, for purification of Gd from its concentrate (74.5%), adopting solvent extraction process and using EHEHPA as an extractant.The solvent extraction process involved two steps. In the 1st step, heavy rare earths upto Tb are extracted, leaving Gd and other lighter rare earths in the raffinate. In the 2nd

Table 1 : The composition of various fractions obtained in counter- current solvent extraction process

Constituent	% Individual rare earth oxide/ Total rare earth oxide						
	Mixed $RECl_3$	MRE	Gd conc	Raffinate-1	Strip liquor-1	Raffinate-2	Strip liquor-2
La_2O_3	22	-	-	-	-	-	
CeO_2	46	2-3			-	-	
Pr_6O_{11}	5.5	3-5			-	=	
Nd_2O_3	20	20-22	LRE-~2	LRE-~2.5	-	LRE-~4	
Sm_2O_3	2.5	4.5	13.8	14	0.024	83	0.2
Eu_2O_3	0.016	0.5	1.14	1.2	0.017	1.1	1.2
Gd_2O_3	1.2	12-15	74.5	82	51.4	11.5	98.5
Tb_4O_7	0.06	1	3.2	0.024	12.3	Traces	0.02
Dy_2O_3	0.18	Traces	2.27	0.04	13.3	-	0.01
HRE	0.5	traces	1-2	<0.02	15	-	0.005

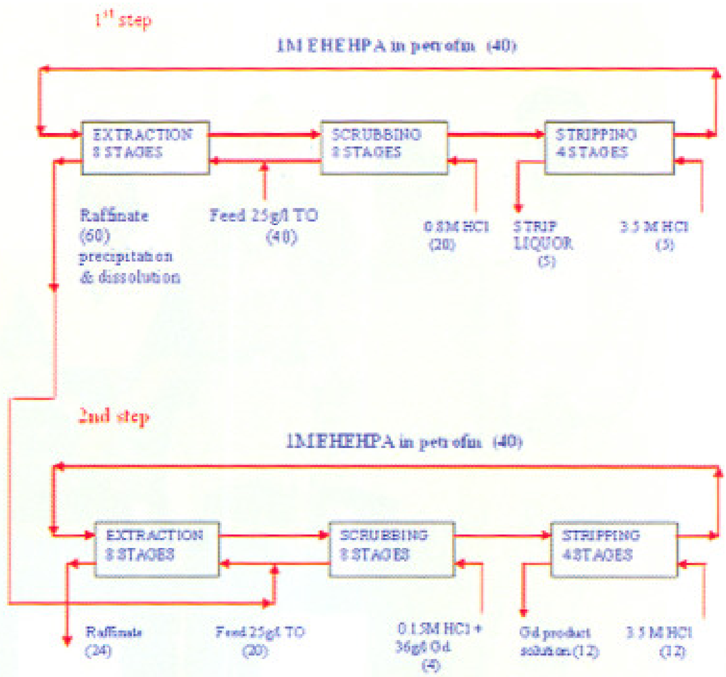

Fig.1 : Flow-sheet for purification of Gadolinium from its concentrate

step, Gd is extracted, leaving Sm and other lighter rare-earths in the raffinate. Gd is stripped with HCl and the solvent is recirculated (Fig.1). The composition of different fractions obtained in the process (Table 1), indicates that, >99% Gd_2O_3 has been obtained. To get higher purity >99.8% of Gd_2O_3, a total reflux system of solvent extraction process was adopted. A 16 stage counter-current process using glass separating funnels was carried out, in which Gd concentrate, bearing chloride solution was fed at the 8th stage, scrubbing liquid at the 13th stage and stripping acid at the 16th stage (Fig. 2). In this system, the aqueous phase of stage nos. 5, 6 and 7 and organic phase of stage nos. 9 and 10 contained Gd_2O_3 (>99.9%). The liquid from these phases was drained and Gd was recovered as Gd_2O_3. Laboratory scale work was followed by mixer-settler runs too.

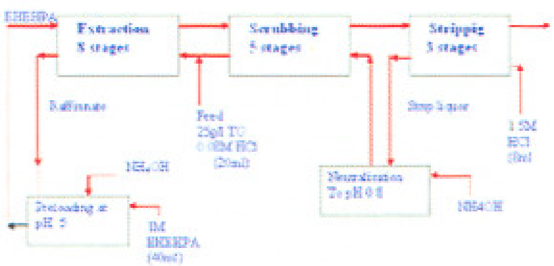

Fig. 2 : Total reflux system for purification of Gadolinium from its concentrate

Production of Gadolinium nitrate

Pure gadolinium oxide was dissolved in 12M HNO_3 (GR grade), to get a solution of pH 3. Clear solution was evaporated, till the temperature of the solution reached 130° C. It was cooled while stirring, to form fine crystals. The analysis of the crystals was carried out in the Radiochemistry Division as well as in REDS. Product contained (µg/g) Na<1, Mg<1, Cl<6.5, Cu<1, Y:4, Nd<1, Sm<1, Eu :7, Dy<1, Tb<1. More than 150kg gadolinium nitrate crystals were produced, packed in two drums and sent to Tarapur.

ANNOUNCEMENT

FORTHCOMING CONFERENCE

InPAC-2006

The Indian Particle Accelerator Conference (InPAC-2006), will be held at BARC/TIFR, Mumbai, during November 1-4, 2006. The conference will deliberate on all aspects of the physics, technology and applications of accelerators and related areas.

Topics include: Proton and heavy ion accelerators; electron accelerators; synchrotron radiation sources and FEL; Beam dynamics and optics; sources for stable and unstable ion beams; Beam diagnostics, control and instrumentation; Magnet design and technology; Power supplies; Radiofrequency and microwave technology; Vaccum technology; cryogenic technology; Radiation safety issues in accelerators; Target development and secondary particle production; Medical, industrial and other applications of accelerators and Novel acceleration techniques.

Contributions in the form of a two-page abstract (as per JACoW template) should be sent through e-mail only to inpac06@barc.gov.in

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ACCELERATED LIFE TEST OF GAMMA UNCOMPENSATED NEUTRON ION-CHAMBER FOR TAPS 3 & 4

K.S. Ulhas, C.G. Karhadkar, A.P. Srivastava, P. Ganeshan and S.K. Marik
Reactor Group

The Reactor Group, BARC, carried out an accelerated life test on a gamma uncompensated Neutron Ion-chamber, supplied by ECIL for the Dhruva radial beam hole, up to a fluence level of 9.46×10^{18} nvt. These test results will be used by ECIL, to validate the design of neutron ion-chambers for TAPS 3 and 4. These neutron ion chambers will be used, to measure the neutron flux in the range of 1×10^{3} to 3×10^{10} nv for regulation and protection system, in 235 MWe as well as 540 MWe Indian PHWRs. Maximum neutron flux for which the detector is designed, considering off normal operations, such as loss of coolant accident (LOCA) and consequent power surge, is 6×10^{10} nv.

Objective

The objectives of the accelerated life test were :

- Irradiation of the neutron detector up to a fluence of 9.46×10^{18} nvt (Corresponding to a flux of approx. 1×10^{10} n/cm^2/sec for a period of 30 years)
- Checking saturation characteristics and ensuring that the 90% saturation current is attained, at a voltage less than 300 V.
- Ensuring that the change in the sensitivity of the neutron detector due to irradiation, is less than 10% of the original value.
- Ensure that the residual current i.e. the current due to the self activation of the detector within 40 hr of reactor shut down, is less than 10^{-11} A.

Brief Design Description of the Neutron Detector

i. *Type of detector:* The detector was a Gamma Uncompensated Neutron Ion-chamber, with integrated mineral insulated cable and consisted of an aluminium chamber, housing the electrodes in a gas-filled atmosphere. The detector assembly weighed 3 kgs.

ii. *Principle of detection:* The electrodes of the detector are coated with a thin film of 95% enriched Boron, which undergoes a $B^{10}(n,a)_3\ Li^7$ reaction, with the incident neutrons. The charged particles in the above reaction spend their energy in the fill gas to produce ionisation and these ions are collected, to give an output direct current.

iii. *Specification of the detector:*

iv. *Various tests carried out on detector by ECIL and BARC before the commencement of the accelerated life test:*

- Insulation resistance test
- Gamma sensitivity test
- Neutron Sensitivity test (neutron flux testing from 1×10^{5} to 1×10^{8} n/cm^2/sec at Apsara and further range of flux testing at Dhruva)
- Dry heat test (at a temperature of 100°C, for 2 hours duration)

SNo	Parameter	Specification
1	Neutron Sensitivity	1.2×10^{-14} Amp/nv
2	Gamma Sensitivity	1.3×10^{-12} Amp/R/Hr
3	Insulation resistance at 25°C *(between signal and inner sheath)*	10^{12} Ohms
4	Insulation resistance at 25°C *(between HT and inner sheath)*	10^{11} Ohms
5	Insulation resistance at 25°C *(between inner and outer sheath for ground isolation)*	10^{9} Ohms
6	Operating Voltage	600 V
7	Detector chamber material	Aluminium 1S i.e.99.6% Aluminium
8	Sensitive Material	Boron Enriched to 95%
9	Fill gas	Hydrogen at atmospheric pressure

- Damp Heat Test (at a temperature of 25°C, RH-80% and a temperature of 40°C RH-95% for 2 No. of cycles)
- Seismic test (acceleration of 3.5 g motion and sinusoidal frequency of 1 to 50 Hz)
- LOCA test (detector subjected to steam at 145°C for half an hour, for two cycles).

Design of the Carrier Assembly for the Detector, for Accelerated Life Test

A two-stage telescopic arrangement was designed and developed for the carrier assembly, which helped in minimising radiation exposure during installation and removal of the Ion-chamber. The telescopic arrangement also helped in negotiating the inner gate cavity of beam hole, during installation and removal.

The carrier assembly consisted of the following sub assemblies.

1. A wooden shielding plug 1260 mm long, 366 mm diameter, with aluminium guide pipe (Fig.1).

2. An adjustable Ion-chamber carrier assembly with roller mechanism, to adjust the Ion-chamber position with respect to the reactor core (Fig. 2).

At the inner end of the wooden shielding plug, a 50 mm thick lead and 2 mm thick cadmium plates were provided and encased in MS cover of 6.5mm thickness, to reduce the radiation field. The wooden shielding plug was provided with a central hole of 45 mm diameter, through which a 32 NB, schedule-80, Al pipe was passed. Through this pipe, three mineral insulated SS cables of 5 mm diameter, connected to the detector were made to pass. The annular gap between the 32 NB Al pipe ID and the three SS cables of 5 mm diameter were packed with lead wool, for minimizing streaming of radiation. A 125NB schedule 40 Al guide pipe of length 1092 mm

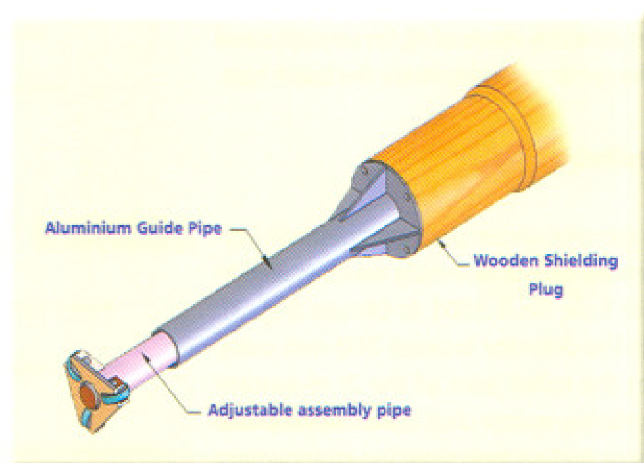

Fig. 1 : One of the subassemblies of the carrier assembly

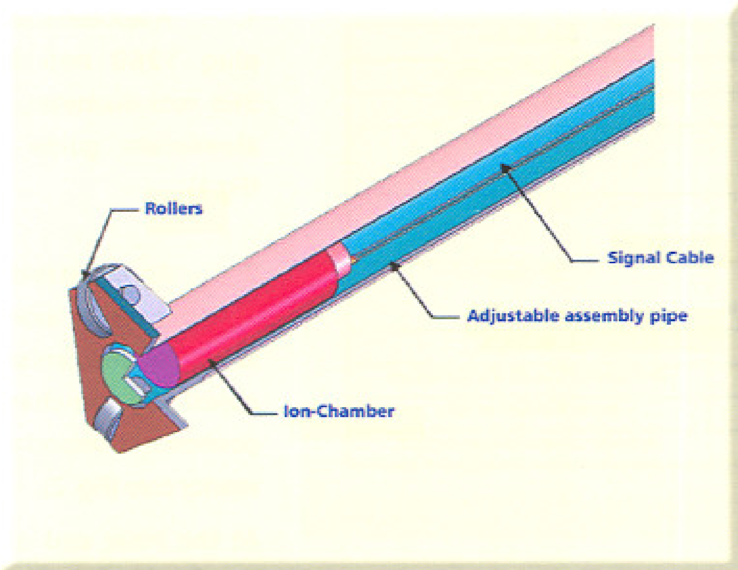

Fig. 2 : The adjustable ion-chamber carrier subassembly

was attached to the wooden shielding plug, by using a Al flange connection and Al 57-S bolts. This 125 NB Al guide pipe, provided support to the 100 NB sch. 40 adjustable assembly Al pipe, which housed the detector.

The detector was placed in this 100 NB Al pipe at the far end and 32 NB schedule 80 Al pipe was hinged to 100 NB Al pipe at the other end. The far end of the 100 NB Al pipe was provided with a roller assembly made of Al, for smooth travel of the carrier assembly inside the beam hole.

Installation

The gamma uncompensated neutron ion-chamber carrier assembly was installed in beam hole No.R-3001 in Dhruva (Fig. 3), with the detector located 200 mm away from the rolled joint of the Zr re-entrant can, during reactor shut-down. Appropriate gamma and neutron shielding arrangement was designed, fabricated and provided around the beam hole R-3001 and approved procedures were followed, for the installation of the detector assembly, so as to minimize radiation exposure.

Disposal of Irradiated Detector Assembly after Accelerated Life Test

The above ion-chamber completed its scheduled irradiation upto a fluence of 9.46×10^{18}, in the month of March 2006. Residual current test, for the qualification of the ion-chamber, was one of the important requirements. This test specified the limit on the maximum current, due to self-activation of the ion chamber and its associated cables, following its irradiation at a rated fluence. This test was to be carried out by ECIL within 40 hrs. of reactor shut-down. After satisfactory residual current test, the detector carrier assembly was remotely cut under special ventilation arrangement, so as to minimise air borne activity. The cut radioactive waste was disposed off to the waste treatment plant.

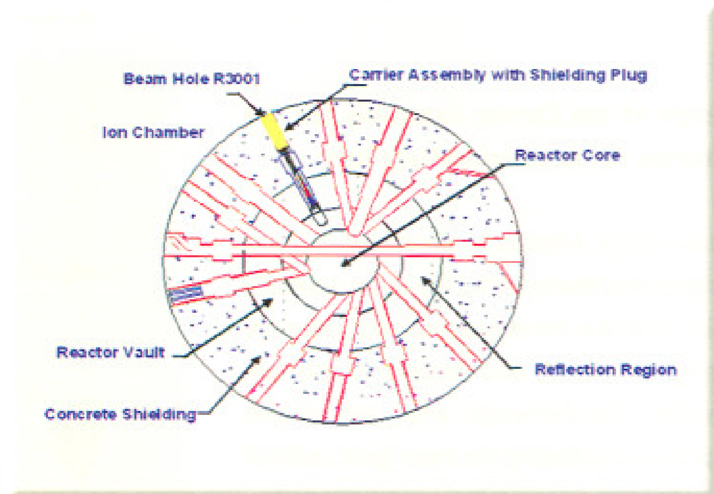

Fig. 3 : Crossectional view of the Dhruva Reactor

बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

QUALITY ASSURANCE OF FUEL AND REACTOR CORE COMPONENTS FOR TAPS 3 & 4

N.G. Dutta, Ram Chandram and B.K. Shah
Quality Assurance Division

The Tarapur Atomic Power Station has two 540 MWe Pressurised Heavy Water Reactors. Fuel and Reactor core components are fabricated at the Nuclear Fuel Complex, Hyderabad. Quality Assurance work on all fuel and Zirconium alloy components is being done by the Quality Assurance Division, BARC.

The various components for each of the 540 MWe PHWRs that are subjected to Quality Surveillance, are as follows:

I. Pressure tubes: 392; seamless, hot-extruded and cold-worked tubes of Zr-2.5% Nb alloy

II. Calandria tubes: 392; seamless, hot-extruded and cold-worked tubes of Zircaloy-4.

III. Garter spring spacers: 1568; (392×4) - Zr-2.5 % Nb - 0.5 % Cu alloy

IV. Fuel Bundles: 5096 (392×13, 37 element fuel bundles)

V. Reactivity Control Mechanism Assem blies: 94.

i.	LZC (Liquid Zone Control, 3 compartments)	2
ii.	LZC (Liquid Zone Control, 2 compartments) -	4
iii.	HFU (Horizontal flux unit) -	7
iv.	VFU (Vertical flux unit) -	26

ANNOUNCEMENT

FORTHCOMING CONFERENCE DAE-BRNS SYMPOSIUM ON NUCLEAR PHYSICS

In the Golden Jubilee year of BARC, the Board of Research in Nuclear Sciences, DAE, is organizing a national symposium between Dec. 11-15, 2006, at the Maharaja Sayajirao University of Baroda, Vadodra.

The symposium topics include:
(a) Nuclear Structure (b) Low and Medium Energy Nuclear Reactions (c) Physics with radioactive Ion Beams (d) Intermediate Energy Nuclear Physics (e) Physics of Hradrons and QCD (f) Relativistic nuclear collisions and QGP (g) Nuclear Astrophysics and nuclear matter and (h) Accelerators and instrumentation for Nuclear Physics. The Symposium format consists of invited talks, contributory papers and thesis presentations. The two-page contributions should be submitted in PDF format and through e-mail only to sympnp@barc.gov.in .

An orientation programme has also been arranged for students, on the topic "Physics with Radioactive Ion Beams". This will be conducted on Sunday, the 10th of December, 2006.

Important dates :

Last date of Submitting contributions	: 31st Aug., 2006
Acceptance status available on website from	: 30th Sep., 2006,
Pre-registration & accommodation request	: 31st Oct., 2006
Accommodation list on website from	: 30th Nov., 2006

For further details one may contact :

Dr Suresh Kumar	Prof. N.L.Singh
Symposium Convener	Local Convener
Tel. : 91 - 22 - 25593712	Tel. : 91–265–2795339
E-mail: sureshk@barc.gov.in	E-mail: singhnandlal@gmail.com
Home page: www.barc.gov.in/symposium/snp2006	

v.	LPI (Liquid Poison Injection)	6
vi.	Guide tubes	
	a) Shut-off Rods-	28
	b) Control Rods-	4
	c) Adjuster Rods-	17
	Total -	94

The Quality Surveillance activities involve:

i. Review of Technical Specifications and Drawing

ii. Review of Manufacturing Engineering Instructions (MEIs) and Quality Control Instructions (QCIs)

iii. Qualification of welding and manufacturing process

iv. Checking calibration of measuring and test equipments

v. Quality audit of fuel and reactor core components to ensure compliance with specifications

vi. Review of deviations or Design Concession Requests (DCRs) in consultation with the designers

vii. Issuing shipping release for the despatch of acceptable fuel and reactor core components, to the reactor site.

By periodic and effective quality surveillance, timely supply of fuel and reactor core components to TAPS 3 & 4 has been achieved, meeting all quality requirements. This has helped the project to be completed on schedule.

ANNOUNCEMENT

FORTHCOMING SYMPOSIUM
51st DAE Solid State Physics Symposium
(DAE –SSPS 2006)

Dec. 26-30, 2006

The DAE –SSPS 2006 will be held at Barkatullah University, Bhopal. Scientific deliberations will include invited talks, Seminars, Ph.D. thesis presentations and contributed papers. The scope of the symposium covers a. Phase transitions to soft condensed matter including biological systems b. Liquid crystals c. Nano- materials d. Experimental technologies e. Liquids glasses & amorphous systems f. Surfaces, Interfaces & their films g. Electronic structure & phonons h.Superconductivity i. Transport properties j. Semiconductor Physics k. migration including Spintronics and l. Novel materials.

Important Dates:

Paper submission 18th Sept. 2006
Acceptance intimation 15th Oct. 2006
Registration & accommodation 15th Nov. 2006

For further details contact:

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E-mail: ssps@barc.gov.in
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बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

SHIELDING ANALYSIS OF TAPS 3 & 4 END-SHIELD

M. V. Gathibandhe, R. A. Agrawal and C. G. Utge
Reactor Projects Division
and
A.S. Pradhan
Nuclear Power Corporation of India Limited

540 MWe TAPS 3 and 4 reactor core is shielded by vault water and heavy concrete shields in radial, top and bottom directions. In the axial direction the circular ends of calandria are closed by End-Shields, containing steel ball and water mixture in the central region and water jacket on the periphery. The end-shields are grouted in the heavy concrete wall. They form part of the calandria vault enclosure and provide shielding to limit the dose rate in the fuelling machine vaults to acceptable limits during shutdown. In reactor operating conditions they along with concrete walls, reduce the neutron and gamma dose rates in accessible areas below acceptable values. The material composition inside lattice tube is entirely different neutronically as compared with the steel ball and water in end-shield. Air gap between end fitting and lattice tube also give rise to radiation streaming. The peripheral water jacket surrounding the end shield, is weaker for gamma radiation compared to central portion. Due to variation of material composition in radial and axial direction, radiation streaming paths and complex geometry, it is necessary to carry out 3-D analysis for reasonable prediction of neutron flux and gamma dose rates. The method of analysis is validated against the measured values for 220 MWe PHWR (KGS-2). Detailed radiation shielding analysis of end-shield for 540 MWe PHWR using Monte-Carlo code MCNP is carried out.

Description of End-Shield of 540 MWe PHWR:

The ball filled end-shields are cylindrical boxes of O.D. 706.4 cm with ends closed by 5.5 cm thick calandria side tube sheet (CSTS) and 8.0 cm thick fuelling machine side tube sheet (FSTS) of stainless steel as shown in Fig.1. The box is pierced by 392 lattice tubes arranged on 28.6 cm square lattice. The

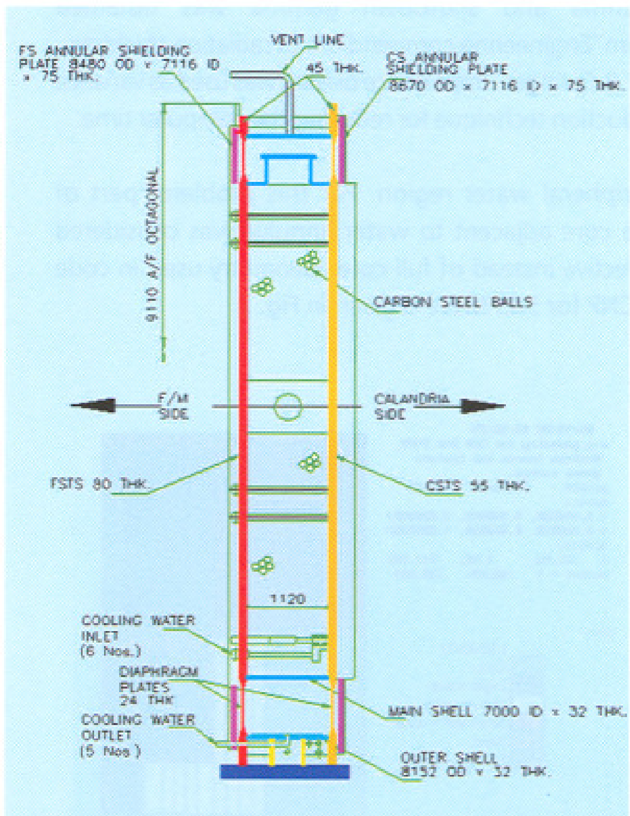


Fig. 1: 540 MWe ball filled End-Shield

compartment of 98.5 cm thickness formed by CSTS and FSTS is filled with 1.0 cm diameter carbon steel balls and water mixture in 57:43 volume ratio same as in 220 MWe end-shields. The end-shield of 706.4 cm O.D. is surrounded by a water jacket up to O.D. 806.4 cm. Annular shield plates of 7.5 cm thick steel are provided on calandria and fuelling machine side to reduce gamma dose rates through the 50 cm wide water jacket.

Method of analysis: The analysis was carried out separately for peripheral water region and for central portion of end-shield simulating the actual 3D geometry. Radial and axial power distribution in the core is taken into account. For calculation of neutron flux and capture gamma dose rate, code MCNP was used in n,p-mode. For core gamma and shutdown gamma dose rates, the code was used in p-mode. Gamma spectrum for core gamma and shutdown gamma was obtained from"Engineering compendium of radiation shielding , Ed: R.G. Jalger" Importance biasing was used as variance reduction technique for reducing the computer time.

Peripheral water region: For this problem, part of the core adjacent to water annulus was considered effective instead of full core. Geometry used in code MCNP for 540 MWe is given in Fig. 2.

Central portion: Part of the core adjacent to the central portion of the end-shield was considered. Lattice tubes passing through the steel ball and water mixture and End fittings protruding out up to 1m from FSTS were modeled. The lattice tubes and end-fittings along with air gap were modeled for MCNP using LAT, FILL and U cards. The geometry plots used in code MCNP are given in Figs. 3 and 4. Dose rate calculation opposite to central portion of end-shield was carried out at the surface and also at 1m away from the end-shield.

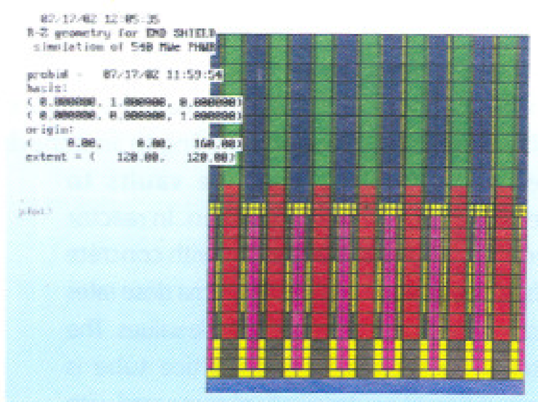


Fig. 3: MCNP plot of End-Shield with lattice tubes in central portion of 540 MWe PHWR

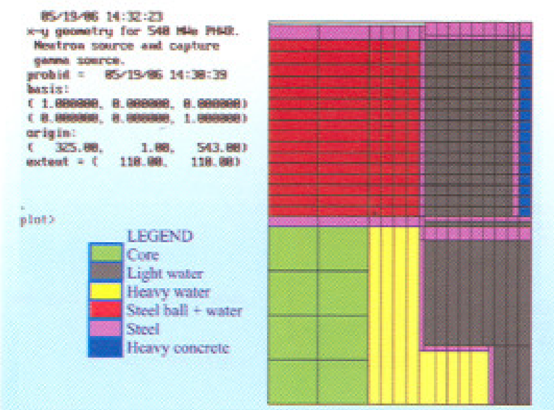


Fig. 2: MCNP plot of core and water annulus region for 540 MWe PHWR

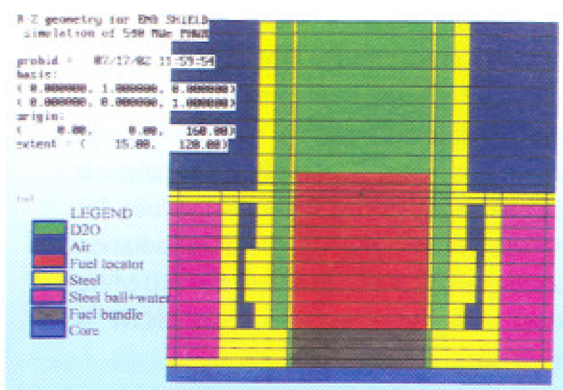


Fig. 4: MCNP plot of lattice tubes for 540 MWe PHWR (magnified 8 times in R-axis)

Results and Discussion (200 MWe PHWR)

Latest measured values for KGS 2 operating at a power of 200 MWe are available. These values have been measured 1.3m away from the FSTS surface and are in Table 1.

Table 1

Location	Neutron flux (n/cm 2-sec) at200MWe			Location	Gamma dose rate (R/hr)
C-10	thermal (<0.4eV)	2.0E+4		Central portion	8-10
	fast (>0.4eV)	6.0E+5		Peripheral portion	200-300

Fast neutron flux calculated at corresponding location at 200 MWe power is 1.36E+6 neutron/cm^2-sec which is ~ 2 times higher compared to measured values (KGS 2). Similarly calculated thermal neutron flux is 3.89E+4 neutron/cm^{2}sec compared to 2.0E+4 neutron/cm^2-sec.

Gamma dose rate in central portion at corresponding location at 200 MWe power due to core is 5.8 R/hr. In addition to this there will be 5-10 R/hr gamma dose due to coolant from feeders and headers. This compares well with the measured value of 8 – 10 R/hr.

Maximum gamma dose rate in peripheral portion at 200 MWe power due to core is 270 R/hr compared to measured value of 200-300 R/hr due to core and coolant.

From the above it can be concluded that, calculated values differ at the most by a factor of 2 from measured values. This can be considered quite satisfactory seeing the complexity of the calculation.

Results and Discussion [540 MWe PHWR]

Total average neutron flux on the surface is 8.82E+6 n/cm^2-sec on surface and 1.56E+6 n/cm^2-sec, 1m away from surface(Table 2A). Neutron flux in central portion is dominated mainly by neutron streaming through lattice tubes. Shielding provided within lattice tubes is

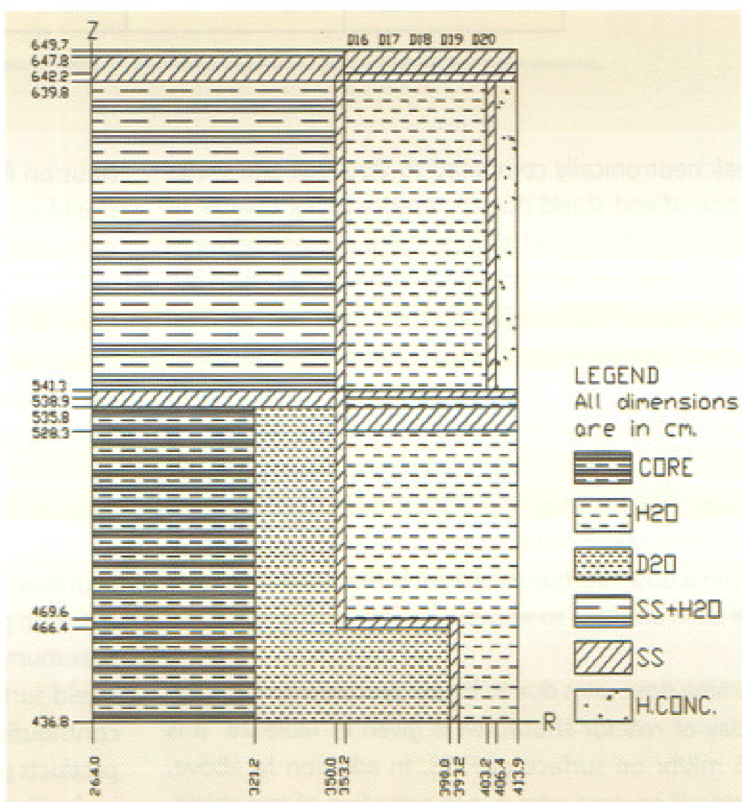


Fig. 5: R-Z geometry of End-Shield of 540 MWe PHWR as Modeled in computer *code* MCNP

On End-Shield (through central steel ball + water region)

Table 2A : Neutron flux on during reactor operating condition				
Location	Neutron flux (n/cm²-sec)			
	Thermal	Intermediate	Fast	Total
On FSTS surface	1.54E+05	8.37E+06	2.98E+05	8.82E+06
1m form FSTS surface	3.56E+04	1.43E+06	9.68E+04	1.56E+06

Table 2B : Gamma dose rate (mR/hr)				
	Reactor operating condition			1 day after shutdown
Location	Capture	Core	Total	
On FSTS surface	1.41E+4	7.64E+2	1.49E+4	1.50E+00
1m form FSTS surface	4.44E+3	1.91E+2	4.64E+3	1.88E-01

weak neutronically compared to steel ball and water potion of end-shield due to absence of light water. In addition to this, small air gap is also present between lattice tubes and end fitting. Separate study carried out using Monte Carlo code indicates that out of total neutron flux, 95% contribution is from lattice tubes.

Gamma dose rates in central portion due to core in reactor operating condition are given in Table 2B. It can be seen that 1m away from FSTS surface, total dose rate is 4.6 R/hr. Major contribution is from capture gamma. Gamma dose rate due to coolant in feeders and headers will be in addition to above.

Gamma dose rates due to fission products in core after 1 day of reactor shutdown is given in Table 2B. It is 1.5 mR/hr on surface of FSTS. In addition to above, there will be dose rates due to activation of end-shield. This is estimated to be about 5–10 mR/hr at 50 cm away from FSTS surface. Dose rates due to corrosion products deposited in headers, feeders, etc. will be in addition to the above.

Neutron flux in peripheral region, on surface of end-shield is given in Table 3A. They are much lower compared to central portion.

Capture and core gamma dose rates on surface of peripheral region is given in Table 3B. It can be seen that, maximum gamma dose is 165 R/hr on contact. Corresponding dose rate in 220 MWe PHWR is 1000 R/hr. This is due the additional 7.5 cm steel plate towards FSTS.

Shutdown dose rates 1 day after reactor shutdown due to fission product gamma in core are given in Table 3B. Maximum dose rate is 2.7 lmR/hr on contact of end-shield surface. In addition to this, there will be major contribution from the fission products and corrosion products present in the coolant of feeders and headers in fuelling machine vault. Activation of FSTS and components near FSTS will also contribute to shutdown dose rate.

On surface of 540 MWe-end-shield (opposite water annulus)

Table 3A : Neutron flux during reactor operating condition

Location* (Refer figure 5)	Neutron flux (n/cm^2-sec)			
	Thermal	Intermediate	Fast	Total
D16	-	7.83E+2	3.45E+2	1.13E+3
D17	7.39E+0	5.77E+2	4.84E+2	1.07E+3
D18	7.39E+0	6.20E+2	4.58E+2	1.09E+3
D19	-	4.64E+2	3.92E+2	8.56E+2
D20	-	3.48E+2	2.62E+2	6.11E+2

Table 3B : Gamma dose rate (mR/hr)

Reactor operating condition				1 day after shutdown
Location (Refer figure 5)	Capture	Core	Total	
D16	7.10E+4	1.95E+3	7.30E+4	7.39E-01
D17	1.25E+5	4.45E+3	1.30E+5	1.21E+00
D18	1.41E+5	1.11E+4	1.52E+5	2.02E+00
D19	1.40E+5	2.43E+4	1.64E+5	2.71E+00
D20	1.16E+5	2.75E+4	1.44E+5	2.04E+00

Conclusion

Neutron flux in fuelling machine vault after end-shield is dominated mainly by contribution from the central portion. This is mainly due to neutrons streaming through lattice tubes. Neutron flux in this area is about 10^6 n/cm^2-sec for both 220 MWe and 540 MWe PHWR. Most of the neutrons are of intermediate energy. Neutron flux in peripheral region is much less.

Gamma dose rate on contact in the peripheral region in case of 220 MWe end-shield is 1000 R/hr. Gamma dose rate in the corresponding location for 540 MWe is 6 times lower. This is due the presence of additional 7.5 cm thick annular shield plate towards FSTS in the latter case. The main contribution is from the capture gammas. Dose rates in the central portion due to core are less.

Shutdown gamma dose rate in fuelling machine vault due to core, through central portion is small in both the cases. The main contribution will come from the fission products and corrosion products present in the coolant of feeders and headers in fuelling machine vault. In addition to above, activation of FSTS and components near FSTS also contribute to shutdown dose rate. Gamma dose rate due to fission products present in the core, give 58 mR/hr on contact of peripheral region for 220 MWe PHWR and 2.7 mR/hr for 540 MWe PHWR, after 1 day of shutdown.

FUEL HANDLING CONTROL SYSTEMS FOR PHWRs

Fuel Handling Control Section, A & M Group

On power refueling is an essential requirement for nuclear power plants (NPPs) based on Pressurised Heavy Water Reactors (PHWRs). Two fueling machines work in synchronism, one at each end of the selected reactor channel for carrying out regular on power refueling. The system handles radioactive spent fuel bundles from reactor and all operations are carried out by qualified operators. Execution of large number of steps in a predefined sequential order and the interlocks based on continuous monitoring of various signals characterise on power refueling operation on a selected reactor channel. In addition to the on power sequences associated with unison operation of the fuelling machines (FM) on the two sides of a reactor channel, there are off reactor sequences associated with receipt of new fuel and discharge of spent fuel through fuel transfer system (FT) by fuelling machines. On reactor sequences and off reactor sequences are executed simultaneously to the extent possible.

The fuel handling system (FHS) is an engineered combination of mechanisms, hydraulics and computerised controls. The complex mechanisms, hydraulics and computer based control systems have to ensure the accuracy of positioning and the required operating forces for fuel handling system devices during repeated operations involved in opening and closing of the pressure boundary of reactor. The system design and operating procedures ensure safety during refueling operations.

For the two 540 MWe PHWR based NPPs at Tarapur, the control system involved utilization of state of the art computer hardware and software technologies. There are many innovative features in the design of mechanical assemblies of fuelling machine. Ram assembly is a very critical component of fuelling machine head and for testing and validation of mechanical design of the first ram assembly, 'ram assembly test facility (RATF)' is setup. To enable testing and validation of various mechanical design concepts of first fuelling machine head, 'fuelling machine test facility (FMTF)' is setup.

RATF and FMTF control systems have been commissioned at BARC. In these control systems, the hardware has been kept to minimum and control requirements and operator interface have been implemented in software, taking advantages of the latest technologies and the interlock logic has been kept in firmware. This has achieved high reliability and low maintenance for the control system, since the software performance is unaffected by ageing. As the failures in software would be due to design deficiencies, considerable effort have been made for thorough testing of control system using real time dynamic simulators as replacement for passive simulator hardware used earlier. These dynamic simulators are PC based, having software models of fuelling machine components and are specifically developed for the purpose. Subsequently, full-fledged control systems were developed and commissioned at Tarapur. The control system software is developed at FHCS, BARC.

These control systems have PC based 'operator interface system (OIS)' replacing the large operators console consisting of digital panel meters (DPMs), indicating alarm meters (IAMs), PID controllers, lamps, analog meters and operate buttons that have been the characteristic features of earlier control systems. OIS provides fuelling machine sequence execution status, text messages indicating operation in progress and diagnostic messages, analog signal plotting, and status of field devices through implementation of meters and lamps. OIS accepts operate commands through keyboard. PID controller algorithms have been implemented in software. OIS is window based providing on line mimics of critical fuel handling system components.

Even though the operations of FHS are fully automated, these are sequential machine operations and require continuous attention and involvement of trained operators. The refueling operation is highly operation intensive whenever full auto mode of operation is held up because of sluggishness of field devices and the need to satisfy all interlocks. Skilled and trained O&M crew therefore becomes necessary to overcome the off-normal situations by correct interpretation of the status of the system and provide uninterrupted regular on power refueling. Proper operations by trained crew can prevent occurrence of off-normal and emergency situations. The simulator training and retraining are uniquely capable of dealing with many of the main recurring problems in the development, qualification and evaluation of personnel.

Work on the first fuel handling training simulator (FHTS) was initiated in 1997 and a training simulator system was commissioned at Nuclear Training Center (NTC) Kaiga in 2002. The simulator in its present form is being used as very effective tool for imparting training to fresh graduate engineers at KGS. However, since then, considerable progress has taken place in the field of computer hardware and software. It has now become

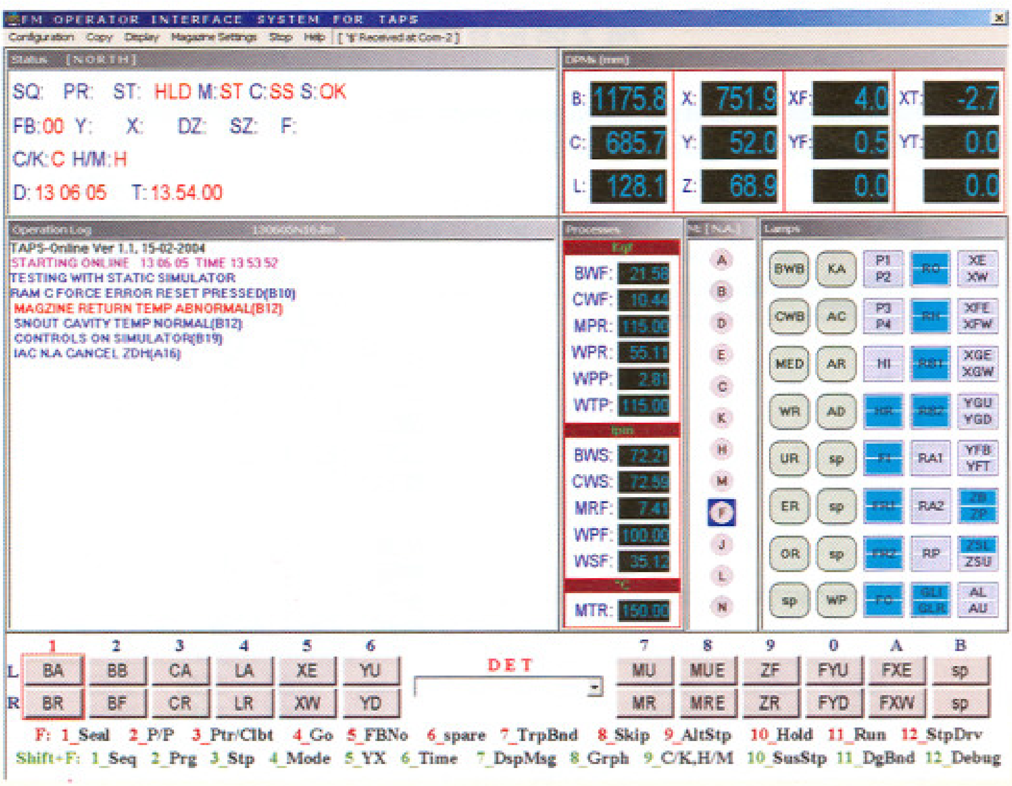


Fig. 1: OIS for FM (TAPS 4)

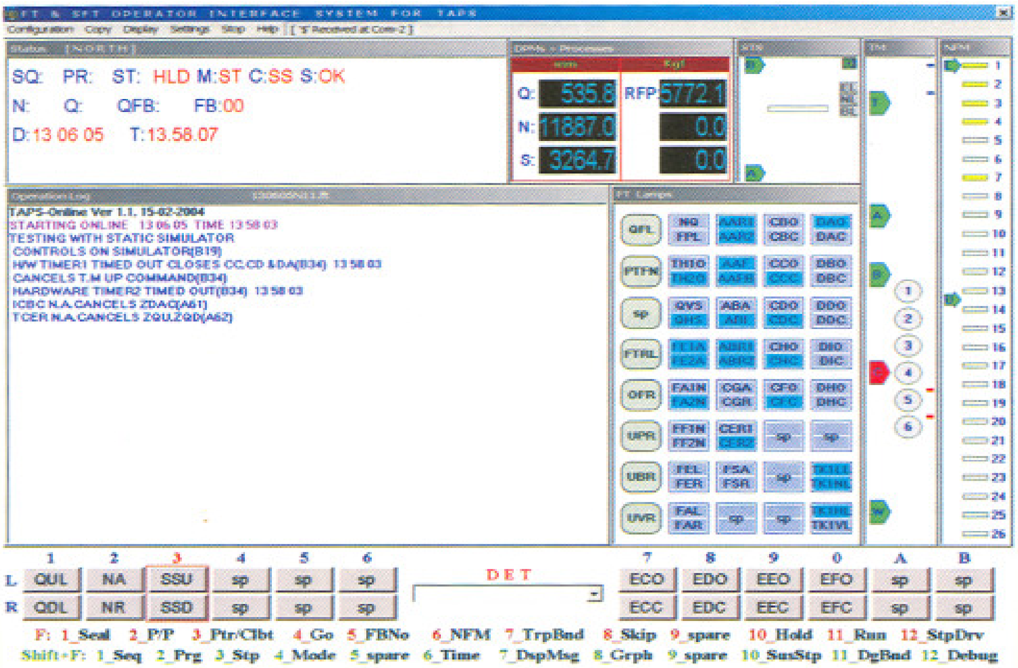


Fig. 2: OIS for FT (TAPS 4)

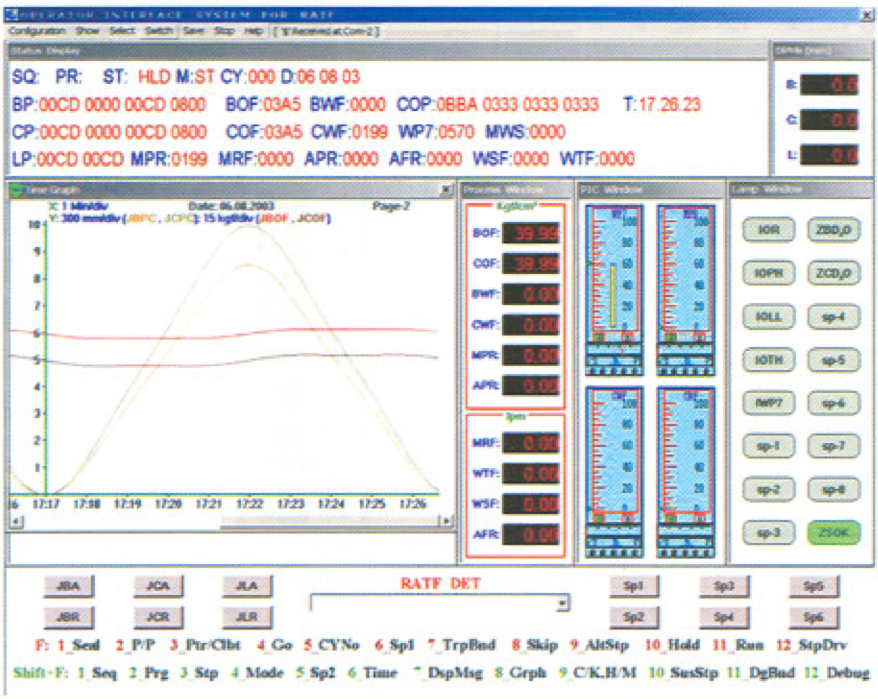


Fig. 3: OIS for RATF

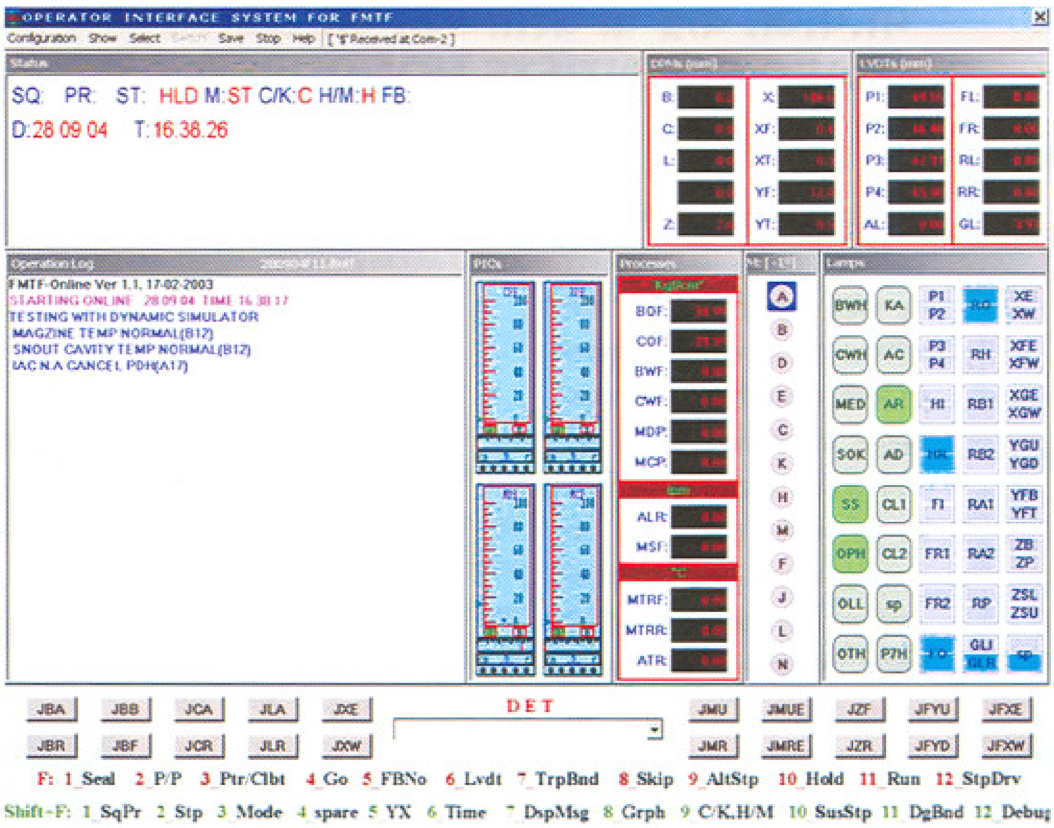


Fig. 4: OIS for FMTF

feasible to provide advanced 3D graphics as part of the simulator and implement the complete system using network of PCs, thereby, reduceing the cost and make it easy to maintain. Also, the simulator at NTC-Kaiga could not fulfill the training needs of fuel handling operators and main plant operators associated with 540 MWe PHWRs at TAPS 3 and 4. In view of this, two MoUs have now been signed on 30[th] January 2006 between BARC and NPCIL. One for the development of 220 MWe PHWR fuel handling training simulator for installation at NTC, Kota. The other MoU is for development of 540 MWe PHWR fuel handling training simulator for installation at NTC, Tarapur.

भा.प.अ. केंद्र के वैज्ञानिकों को सम्मान
BARC SCIENTISTS HONOURED

Dheeraj Jain

C. G. S. Pillai

B. S. Rao

R. V. Kulkarni

E. Ramdasan

धीरज जैन, रसायन प्रभाग, भाभा परमाणु अनुसंधान केंद्र को थर्मल डिफ्यूजिविटी एंड थर्मल कंडक्टिविटी स्टडीज़ आफ थोरिया - लंटाना सोलिड सोल्यूशन्स इन दि लोअर कम्पोज़िशनल रेंज नामक शोध-पत्र की मौखिक प्रस्तुति के लिए प्रथम पुरस्कार दिया गया। इस शोध-पत्र को फरवरी 6-10, 2006 के दौरान राजस्थान विश्वविद्यालय में आयोजित थर्मल विश्लेषण की 15 वीं राष्ट्रीय परिचर्चा तथा कार्यशाला में सर्वश्रेष्ठ पुरस्कार भी प्रदान किया गया। सी.जी.एस.पिल्लै, बी.एस.राव, आर.वी.कुलकर्णी, ई.रामदासन तथा के.सी.साहू इस शोध-पत्र के सहलेखक हैं।

Mr Dheeraj Jain of Chemistry Division, BARC, was awarded the first prize for the oral presentation of a paper entitled "Thermal diffusivity and thermal conductivity studies of thoria- lantana solid solutions in the lower compositional range. This paper also received the best paper award at the 15th National Symposium and Workshop on Thermal Analysis, held at the University of Rajasthan, during Feb. 6-10, 2006. Co-authors of this paper were C.G.S. Pillai, B.S. Rao, R.V. Kulkarni, E.Ramadasan and K.C. Sahoo.

रोहिणी वांदेकर (मुंबई विश्वविद्यालय की पीएच.डी.छात्रा), रसायन प्रभाग, भाभा परमाणु अनुसंधान केंद्र, ने कम्पेरेटिव स्टडी ऑफ 20 मोल % Gd_2O_3 डोप्ड CeO_2 प्रिपियर्ड बाई डिफरंट सिंथटिक रुटस् नामक शोध-पत्र के लिए द्वितीय पुरस्कार प्राप्त किया। यह शोध-पत्र फरवरी 6-10, 2006 के दौरान राजस्थान विश्वविद्यालय में आयोजित थर्मल विश्लेषण की 15वीं राष्ट्रीय परिचर्चा तथा कार्यशाला में प्रस्तुत किया गया। बी.एन.वानी एवं एस.डी.भारद्वाज इसके सहलेखक हैं।

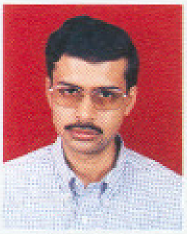
Ms Rohini Wandekar (a Ph.D. student of Mumbai University) from Chemistry Division, BARC, received the second prize for her paper "A comparative study of 20 mole % Gd_2O_2 doped $Ce O_2$ prepared by different synthetic routes. " This paper, co-authored by B.N. Wani and S.D. Bharadwaj was presented at the 15th National Symposium and Workshop on Thermal Analysis, held at the University of Rajasthan, during Feb. 6-10, 2006.

K. C. Sahoo

R. Wandekar

B. N. Wani

S. D. Bhardwaj


Tapan Das

डॉ. तपन दास, रेडियोभेशज प्रभाग, भाभा परमाणु अनुसंधान केंद्र को मार्च 17,2006 को मुंबई में वर्ष 2005 का प्रोफसार एच.जे.अरनिकर-उच्चतम शोध ग्रंथ पुरस्कार से सम्मानित किया गया। नाभिकीय रसायन एवं संयुक्त वैज्ञानिकों (IANCAS) के भारतीय समुदाय द्वारा संस्थापित यह पुरस्कार नाभिकीय विज्ञान क्षेत्र के उस सर्वश्रेष्ठ पीएच.डी शोध ग्रंथ को दिया जाता है जो कि किसी भी भारतीय विश्वविद्यालय में पिछले दो साल के अंदर प्रस्तुत किया गया है। प्रपरेशन एंड करेक्टरैजेशन ऑफ लाइगेंडस् लेबल्ड विद रियक्टर प्रोड्यूसड आइसोटोपस् फॉर यूज़ एज़ पोटेंशल एंड रेडियोफार्मासेटिकल्ज़ नामक यह शोध-ग्रंथ आर. पीएच.डी, भाभा परमाणु अनुसंधान केंद्र के डॉ. एम.आर.ए. पिल्लै के निर्देशन में संपूर्ण हुआ।

Dr Tapan Das of Radiopharmaceuticals Division, BARC, was awarded the "Prof. H.J. Arnikar-Best Thesis Award", for the year 2005, on the 17th of March, 2006, at Mumbai. The award instituted by the Indian Association of Nuclear Chemistry & Allied Scientists (IANCAS) is given to the best Ph.D. thesis in the field of nuclear sciences, submitted to any Indian University, within the last two years preceding the award. His Ph.D. work "Preparation and characterization of ligands labeled with reactor produced isotopes for use as potential radiopharmaceuticals "was completed under the guidance of Dr M.R.A. Pillai, from RPhD, BARC.


H. D. Sharma

डॉ. एच.डी.शर्मा को पशु-चिकित्सा विज्ञान की प्रगति में योगदान को मान्यता देने के लिए भारतीय चिकित्सा विज्ञान की राष्ट्रीय अकादमी की सदस्यता प्रदान की गई। आजकल डॉ. शर्मा पशु-सुविधा एवं रेडियो आईसोटोप लेबोरेटरी के अध्यक्ष तथा भाभा परमाणु अनुसंधान केंद्र की पशु-नीतिशास्त्र समिति के महासचिव हैं। राष्ट्रीय एवं अंतर्राष्ट्रीय पत्रिकाओं में 50 से अधिक शोध-पत्र इनके श्रेय में हैं। पशु मॉडलस् में चिकित्सा तथा निदान में अभूतपूर्व रेडियो भेषज कैंसर पैदा करने वाले विकिरण, इसका प्रयोग एवं मूल्यांकन, इनकी अनुसंधान रुचि में शामिल है।

Dr H.D. Sarma has been inducted as a Fellow, to the National Academy of Veterinary Sciences (India), in recognition of his contribution to the advancement of Veterinary Sciences. Dr Sarma is currently Head, Animal Facility and Radioisotope Laboratory and Member secretary, BARC Animal Ethics Committee. He has more than 50 papers (in national and international journals) to his credit. His research interests include radiation carcinogenesis and application and evaluation of newer radiopharmaceuticals in diagnosis and therapy in animal models.


S. S. Gandhi


J. Udhaya Kumar


A. Das

श्यामला एस.गांधी, जे.उदय कुमार, एवं ए.दाश, रेडियोभेषज प्रभाग, भाभा परमाणु अनुसंधान केंद्र को प्रपरेशन ऑफ ^{63}Ni इलेक्ट्रोडिपोज़िटिड स्पेशल कस्टममेड सोरसिज़नामक शोध-पत्र के लिए सांत्वना पुरस्कार दिया गया। यह शोध-पत्र नवंबर 10 एवं 11, 2005 के दौरान भाभा परमाणु अनुसंधान केंद्र, मुंबई में आयोजित एन ए सी-2005 (NAC-2005) के सम्मेलन में प्रस्तुत किया गया था।

Ms Shyamala S. Gandhi, J. Udhaya Kumar and A. Dash from Radiopharmaceuticals Division, BARC, were awarded the consolation prize for their paper "Preparation of ^{63}Ni electrodeposited special custom made sources." The was presented at the NAC-2005 Conference, held between Nov.10 & 11, 2005 at BARC, Mumbai.

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