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BARC NEWSLETTER

IN THIS ISSUE

BARC'S CONTRIBUTION TO 540 MWe PHWRs : TARAPUR ATOMIC POWER STATIONS 3 & 4

(ISSUE 2 OF 3 SPECIAL ISSUES)

भाभा परमाणु अनुसंधान केंद्र
Bhabha Atomic Research Centre

पंडित जवाहरलाल नेहरू द्वारा परमाणु ऊर्जा ंस्थान, ट्रॉम्बे (जिसका बाद में श्रीमती इंदिरा ाधी 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.

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DEVELOPMENT OF ION-EXCHANGE SCHEMES FOR SELECTIVE REMOVAL OF GADOLINIUM NITRATE IN THE PRESENCE OF BORON, FROM THE MODERATOR SYSTEM OF 540 MWe PHWRs - TAPS 3 & 4

S. Dutta, P. Suryanarayanan, A.R. Kandalgaonkar, R.S. Sharma, H. Bose and V.K.P. Unny

Research Reactor Services Division

In Indian Pressurized Heavy Water Reactors (PHWRs), soluble neutron poison is employed, in the heavy water moderator system, to provide an independent and alternate shut-down capability as well as a mechanism, for reactivity control. Boron, as boric acid/lithium pentaborate has traditionally been used for this purpose, in the 220 MWe Indian PHWRs. A noteworthy design change in the 540 MWe Indian PHWRs, TAPS 3 & 4, is the use of gadolinium as neutron poison in the fast acting Secondary Shut-down System (SDS 2) and also in the reactivity shim control. This involves, direct injection of a concentrated solution of gadolinium nitrate hexahydrate into the bulk moderator in calandria, to effect a reactor shut-down. For the first time, gadolinium has been used in an Indian reactor for the above purpose. The use of gadolinium, results in significant changes, in the operational chemistry of the moderator system. For a pre-equilibrium core, both gadolinium and boron are maintained in the moderator, to keep the reactor in a sub-critical state. The approach to first criticality and subsequent reactor start-ups, require a reliable and fast acting method, for selective removal of gadolinium nitrate, leaving behind boron in the moderator. For this purpose, an ion exchange scheme was developed, optimized and qualified in the Reactor Group laboratory, for plant scale operation. This scheme has since been successfully employed in both TAPS 3 & 4.

Advantage of gadolinium over boron

Advantage of using gadolinium (as gadolinium nitrate) is, due to its high thermal neutron absorption cross section, quick burnout (matching Xe-135 build-up rate), greater solubility in the desirable pH range applicable to moderator (and for storage of concentrated poison solution) and more efficient removal by ion-exchange system, as compared to boron (as boric acid). The considerably smaller concentration requirement of gadolinium implies, much shorter removal time, which results in a considerable reduction in the run-up time to reactor start-up. This time-period is further reduced, by the quick burnout of gadolinium (matching with Xe-135 build-up rate), which obviates the need for complete removal of gadolinium, before reactor start-up. The smaller concentration and more efficient ion exchange for gadolinium ion removal, result in ion-exchange resin requirements, that are about 15 times lesser, as compared to that for boron removal. This results in considerable cost savings on ion-exchange resins and deuterization of resins, as well as a reduction in man-rem and radioactive waste generation. Notwithstanding its many advantages, the tendency of gadolinium salts to undergo hydrolytic precipitation in neutral and alkaline water, requires that, due caution be exercised in ensuring strict water chemistry control, where the pH of gadolinium solution is always maintained at $\leq$ 5.5 with free acidity, commensurate with gadolinium concentration. This is to prevent any deposition of gadolinium as gadolinium hydroxide in the core, which can have adverse implications. Gadolinium nitrate also enhances the radiolysis of the moderator, which needs to be taken care of, by suitable design and operational measures.

Considerations in selective removal of $Gd(NO_3)_3$ in the presence of boric acid :

The mixed resin beds commonly employed in moderator purification systems of other PHWRs, consist of strong acid-gel type cation resin (SAC), with sulphonic acid functional group and strong base gel type anion resin (SBA), with quaternary amine functional group. Such mixed beds will remove both Gd $(NO_3)_3$ and boric acid and hence cannot be used for the stated purpose. The selective removal is, however, possible, if a weak base anion resin (WBMA), with a tertiary amine functional group ($R\text{-}NMe_2$), available in macroporous matrix form, is used along with strong acid cation resin (SAC), to constitute the mixed resin bed. This is because, unlike SBA resin, a WBMA resin cannot remove a weak acid like boric acid but can efficiently remove a strong acid like nitric acid, which is added to provide the necessary free acidity. The removal of gadolinium and the free acidity by resins SAC denoted by $[R\text{-}SO_3^-H^+]$ and WBMA, denoted by $[RNMe_2]$ may be represented as follows:

$$Gd(NO_3)_3 \qquad Gd^{3+} + 3\,NO_3^-(aq) \qquad (1)$$

$$3[R\text{-}SO_3^-\,D^+] + Gd^{3+} \longrightarrow [R\text{-}SO_3^-]_3\,Gd^{3+} + 3D^+ \qquad (2)$$

$$3[R\text{-}NMe_2] + 3D^+ + 3NO_3^- \longrightarrow 3[R\text{-}NMe_2.DNO_3] \qquad (3)$$

Another important aspect to be considered in the ion-exchange removal of gadolinium by mixed bed, is the tendency of precipitation of gadolinium hydroxide, on anion exchange resin beds, in the ion-exchange column. This precipitation is significant on SBA resins, as a result of which, such resins cannot be recommended for the removal of $Gd(NO_3)_3$ even when boron is not present; as will be the case in post-equilibrium core. However, even in the case of mixed bed constituted of SAC resin and weak base macroporous resin (WBMA) , trace precipitation in the column was observed in the laboratory, for higher influent gadolinium concentration (> 10 mg/L), possibly, due to the presence of a trace residual strong base function (~2%). Precipitation of gadolinium hydroxide, inside the ion-exchange bed must be avoided, since it could lead to a slippage of the precipitate into the moderator, resulting in unwanted negative reactivity and/or flux tilts in the core, which could even lead to forced outages and delayed reactor start-ups.

Laboratory Studies:

Plant simulated laboratory studies on ion-exchange removal were carried out, to arrive at optimized configurations/schemes, for removal of gadolinium nitrate in the presence of boron. In the scaled down experimental set up (Fig. 1), a mixed solution of gadolinium nitrate and boric acid, with pH suitably adjusted with nitric acid, was circulated in a closed loop through an ion-exchange column, using a peristaltic pump. Samples were collected at intervals, at the inlet and outlet of the resin column

Fig. 1: Removal of Gadolinium Nitrate : experimental set-up

and analysed for pH, specific conductivity, gadolinium and boron. An evaluation of WBMA resin quality characteristics, including thermal and radiation stabilities were also carried out on several resin samples, based on

Table 1
Analysis carried out for WBMA resin Duolite A-369

Characteristic properties	Duolite A-369
Form	Free base
Total Ion-Exchange Capacity(meq/ml) Weak base capacity Strong base capacity	1.8 1.76 0.04
Moisture Content (%w/w)	45
Effective size (mm)	0.5
Uniformity Co-efficient	1.6
Fines content passing through 300 micron sieve	<0.5% V/V
Water soluble organics (mg $KMnO_4$/ml resin)	0.2
Chloride content (meq/ml)	0.03
Impurities (mg/l of resin) a) Iron b) copper	 <10.0 <10.0
Thermal stability: Capacity loss at 60°C, in 24 hours (%) Organic leach at 27°C (mg $KMnO_4$/ml resin) Organic leach at 60°C (mg $KMnO_4$/ml resin)	 Nil < 0.2 0.3
Radiation stability: Capacity loss at 70 Mrad dose (%) Organic leach at 27°C (mg $KMnO_4$/ml resin)	 4 0.6

which, Duolite A-369 was selected. The results of analyses carried out on Duolite A-369 (Table 1) showed the resin, to be at par with nuclear grade resins.

***Operational scenarios with addition and removal of gadolinium*:**

The various operational situations involving addition of gadolinium to moderator and its subsequent removal are : (i) during approach to first criticality (ii) during reactor start-up after a poison shutdown (iii) for routine reactivity shim control and (iv) upon actuation of Secondary Shut Down System (SDS 2). From the point of view of ion-exchange removal, the above scenarios can be grouped under the following two classes:

- Scenarios (i), (ii) & (iii), requiring addition of gadolinium upto a concentration of about 2mg/L in the moderator.
- Actuation of SDS 2 (scenario (iv)), requiring addition of gadolinium at a concentration of about 15 mg/L in the moderator.

Ion-exchange Purification Schemes :

Given the propensity of gadolinium for precipitation on anion resin in mixed beds, especially at higher concentrations, an ideal configuration would be, a cation bed followed by a mixed bed in series. However, since only parallel bed arrangement exists in the moderator purification system of TAPS 3 & 4, two types of ion-exchange processes, viz., a single-stage scheme and a two-stage scheme were developed in the Reactor Chemistry Section of RRSD, for the selective removal of gadolinium nitrate, in the presence of boron, for the two classes of operational scenarios mentioned earlier. Scaled-up plant versions of the laboratory experiments denoting the two purification schemes, presently deployed at TAPS 3 & 4 are described below:

Single stage purification scheme:

During first approach to criticality or during reactor start-up after a prolonged shut down or during routine reactivity shim control, gadolinium nitrate which is present in the moderator at concentrations of 2mg/L or less, was removed from the moderator system, by employing a mixed bed comprising 50 litres of strong acid cation resin (SAC) and 70 litres of weak base macro-porous anion resin (WBMA), topped with a layer of 80 litres of SAC. The schematic diagram of the single stage purification scheme is shown in Fig. 2.

The cation resin topping is provided in the mixed bed as a precautionary measure to prevent precipitation of gadolinium hydroxide in the mixed bed. Removal characteristic of $Gd(NO_3)_3$ and variation of pH and specific conductance of bulk solution are shown in Figs.3 & 4.

Two stage purification schemes:

After the actuation of SDS 2 for reactor shut-down, 15mg/L of Gd is present in the moderator system along with boron. The single stage procedure described above

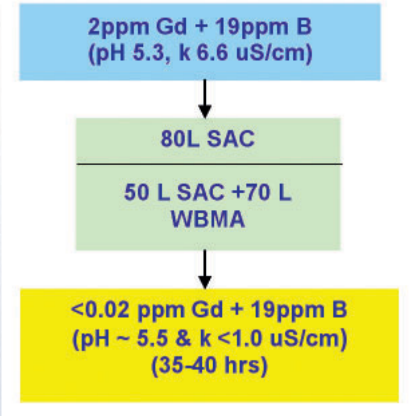


Fig. 2 : Single stage purification : schematic

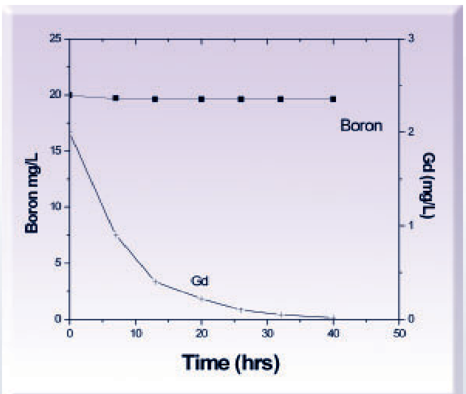


Fig. 3 : Removal characteristic of $Gd(NO_3)_3$

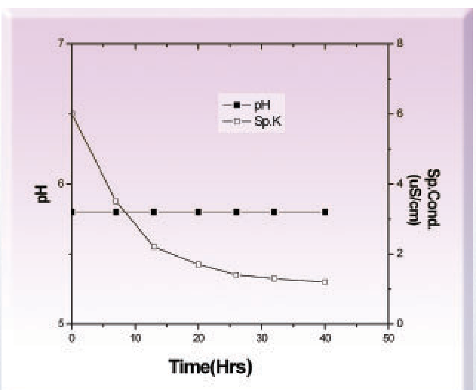


Fig. 4 : Variation of pH and specific conductance of the bulk solution

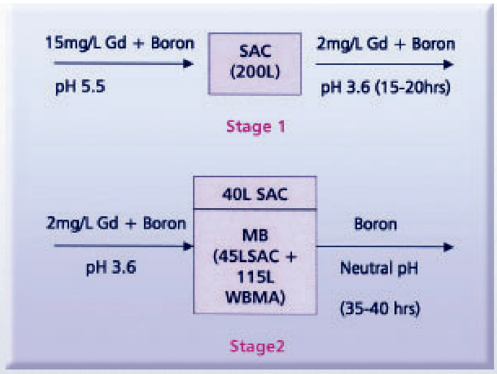


Fig. 5 : Two-stage purification scheme for complete removal of gadolinium

was found to be unsuitable for higher gadolinium concentration, which was observed to lead to trace precipitation in the resin column. A two-stage scheme was therefore employed. In the first stage, a cation bed containing 200L SAC was employed, to reduce gadolinium concentration from 15mg/L to $\leq$ 2.0mg/L. In the second stage, the acidic solution containing $\leq$ 2 mg/L gadolinium from the first stage, was passed through a mixed bed, comprising 45 litres of SAC and 115 litres of WBMA resins, topped with 40 litres of SAC resin, for complete removal of gadolinium and free acidity.

The above mentioned ion-exchange schemes developed by the Reactor Group, BARC have been successfully employed in the operating TAPS 4 unit and have also been installed at TAPS 3, which is approaching criticality.

Ongoing developmental work for improvisation of the ion-exchange scheme:

The time taken for gadolinium removal after actuation of SDS 2 can be significantly reduced, if the two-stage process is replaced with a single-stage process. Developmental work continues in the Reactor Chemistry Section, RRSD using different types and combinations of resins and layered arrangements in column, to overcome the problem of gadolinium hydroxide precipitation, in mixed bed.

DESIGN AND DEVELOPMENT OF FUEL LOCATOR FOR 540 MWe PHWR

Refuelling Technology Division

In a 540 MWe PHWR, the coolant channel is longer as compared to that of a 220 MWe. To keep the length of the Fuelling Machine shorter, shield plugs are located out board. This increases the gap between the fuel bundles and the shield plug. To fill this gap, a fuel locator has been used, one each on the upstream and down stream side of the coolant channel assembly. In addition to being compatible with the fuel bundle, the out board end of the fuel locator, has a mechanism to be picked up by the fuelling machine. It offers less pressure drop (maximum 2 Kg/cm^2) and it provides shielding against radiation.

To satisfy the above requirements, three alternative designs of the fuel locator and the corresponding liner tubes were proposed to NPCIL, by the Refueling Technology Division. Based on the result of minimum pressure drop across the locator and fulfilling other intended requirements, the following design was finalized :

Objective of the work

- To conceptualise alternative designs of the fuel locator,
- To design the corresponding liner tube,
- To measure the pressure losses experimentally, in simulated condition for all designs,
- To measure the vibration level caused by coolant flow in fuel locators,
- To select the best design of the fuel locator based on experimental and theoretical results,
- To refine the selected fuel locator for reactor use and if possible, to minimize pressure losses and vibration.

Design evolution of Fuel Locator

Three designs of fuel locators were conceptualized, each with flow entry/exit from front, middle and end

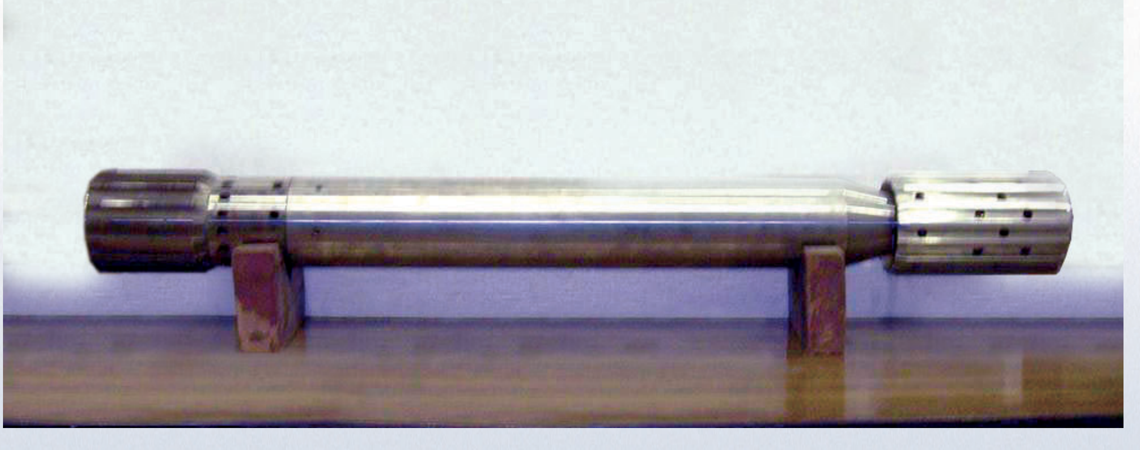


Fig. 1 : Fuel Locator

respectively. Corresponding liner tube designs, were also worked out. Functioning of the locator in crept channel and its effect on liner tube design were considered. Three designs of locators with respective liner tubes were fabricated (one was supplied by NPCIL) for performance evaluation.

Description of Fuel Locator Test Set-up

Since an actual coolant channel was not available, a fuel locator test set-up was designed, fabricated and installed at BARC. Until recently, when an actual 540 MWe coolant channel was installed, this set-up was proven to be useful and handy for qualifying other channel components too. The set simulated the coolant channel internal geometry by using pipes. The set-up was connected with Integral Thermal facility (ITF), to tap water at high temperature and pressure. The test set-up was designed in such a way, that liner tubes could be easily changed to enable the experiment to be carried out with different combinations of fuel locator and liner tubes. Seventeen pressure tapings are provided on the set-up. To understand various pressure drops like entry loss, liner tube annulus and fuel locator, many differential pressure transmitters were installed, to get each and every component.

Fig. 2 : Fuel Locator Test Facility

Design Conditions

- Design pressure : 104.7 Kg/cm^2
- Operating pressure max.: 92 Kg/cm^2.
- Minimum operating temperature : 270°C

In this facility, pressure drop of all the three alternatives were measured, at various flows, at room temperature. From these readings, pressure drop of all the three fuel locators was estimated at reactor condition. Accelerometers were mounted on the outer diameter of the pipe, where fuel locator is positioned inside the channel, to measure flow-induced vibrations.

Experimental programme

The locator with a coolant entry/exit at the end, was found to have maximum pressure drop and a relatively higher magnitude of vibration. Hence it was not pursued. Maximum combined pressure drop, upstream and downstream conditions, for the locator with coolant entry/exit at the front was 1.677 Kg/cm^2. The same value for the locator with coolant entry at the middle, was 2.248 Kg/cm^2. This was based on 97 holes in the liner tube. By increasing the number of holes in the liner tube to 247, pressure drop was reduced by about 35% to 1.5 Kg/cm^2 well within prescribed limits of 2 Kg/cm^2. Based on this, the locator with a coolant entry /exit at the middle, was selected for the reactor. This design was further subjected to endurance test and accidental entry simulation in the fuel locator test facility, at near-simulated condition.

DEVELOPMENT OF FLUID POWER COMPONENTS AND CIRCUITS FOR 540 MWe PHWR FUEL HANDLING SYSTEM

Refuelling Technology Division

The fuelling machine (FM) of Indian PHWRs operates remotely in an inaccessible, highly radioactive area. It becomes part of the Primary Heat Transport (PHT) System of the nuclear reactor during on-power refueling. As such, any equipment of the fuelling machine should be light, compact, radiation resistant and extremely reliable.

Oil and water hydraulic controls are extensively used in FM since these are compact, generate constant stalled torque, speed and torque can be continuously controlled independently and easily, high radiation resistance and highly reliable. In 540 MWe PHWR's fuel handling system, the following major components and system have been developed / selected.

A) Design and Development of Differential Pressure Reducing Valve (DPRV).
B) Selection of High to Rque Low Speed (HTLS) hydraulic motors and testing of special oil hydraulic components.
C) Development of leak detector circuit for FM.

A) Design and Development of Water Hydraulic Differential Pressure Reducing Valve (DPRV)

In fuelling machine, there are some actuators where the generated force needs to be constant. The force generated by a hydraulic actuator depends on the differential pressure across the actuator. In the fuelling machine, return lines of all the actuators open in fuelling machine magazine cavity. The pressure of this magazine cavity varies, depending on the operation of the fuelling machine. This change of pressure will lead to change in actuator force. To avoid this change in force by the actuators inside the pressure boundary with fluctuating pressure, DPRV is used. Thus, DPRV is a valve, designed to maintain a constant force generated by an actuator located inside a pressure boundary with fluctuating pressure.

The valve has three ports, viz. inlet port, outlet port and reference port. DPRV always maintains a constant differential pressure between the outlet and the reference port. Changing the spring setting can change the magnitude of this difference.

Working Principle of DPRV:

Fig. 1 shows the cross section of a DPRV. The valve consists of a poppet, seat and a balance piston and spring enclosed in a valve body. Arrangement for compressing spring by a screw has been provided. A light spring, maintains contact of the poppet with the balance piston.

The balance piston in the valve is subjected to the outlet pressure of the valve on one side. The other side is

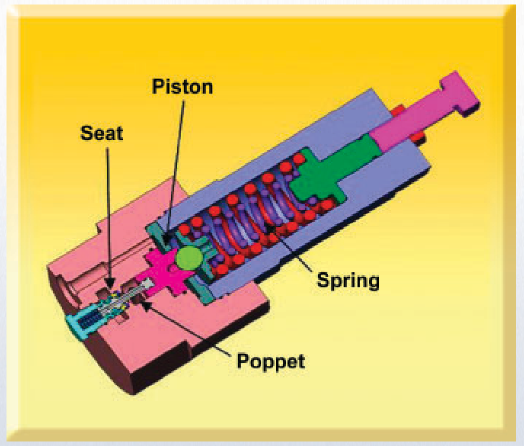


Fig. 1: Single stage DPRV

subjected to reference pressure and a force generated by the spring. Hence, for the balance piston to be in equilibrium, force generated by outlet pressure equals to the gm of force generated by reference pressure and the spring force.

Therefore, the outlet pressure of the valve will always be maintained above the reference pressure, by an amount equivalent to the spring force.

Fig. 2: High flow single stage DPRV

In the DPRV used in Fuelling Machines (FM), the supply pressure is 147 Kg/cm^2 while the reference pressure is 92-100 Kg/cm^2 and the differential pressure of 14 or 35 Kg/cm^2 is required to be maintained. So the pressure drop across DPRV is of the order of maximum 43 Kg/cm^2.

While the DPRV for Fuel Transfer (FT) system, the supply pressure is 147 Kg/cm^2 and reference pressure is 0-2 Kg/cm^2 and a differential pressure of 14 Kg/cm^2 is to be maintained. So the pressure drop across the DPRV is of order of 133 Kg/cm^2.

When the same valve was tested for working condition in FT system, choked flow was observed. Mathematically it was found that choked flow will occur in DPRV at a pressure drop of 116 Kg/cm^2 or above. Therefore, a new valve was developed for this specific application to avoid choke flow & cavitations. Fig. 3 shows the DPRV with anti-cavitation design. In this DPRV the main orifice has two stages, the total pressure drop is divided into two parts so the pressure drop at each stage is less than 100 bar.

Fig. 3: Low flow anti-cavitation DPRV

Specifications

RTD has designed various DPRVs for 220 MWe as well as 540 MWe PHWRs. The specifications of the valves are shown in Table 1.

Testing of DPRV

The performance of DPRVs developed at RTD has been evaluated in DPRV test facility. Following tests have been carried out on the DPRVs.

- Spring setting vs differential pressure
- Effect of supply pressure on differential pressure
- Effect of Flow on differential pressure
- Step change in load
- Testing to observe

The test results show that the performance of the valves is satisfactory.

Fig. 4 shows a plot displaying the effect of flow on differential pressure of the valve, as observed in the testing of the valve.

Valve Type		Low Flow Single stage	High Flow Single stage	Low Flow Anti-cavitation	High Flow Anti-cavitation
Design Pressure (Kg/cm^{-2})		200	200	200	200
Number of stages		1	1	2	2
Inlet Pressure	Low Pressure Mode (Kg/cm^{-2})	63 – 68	63 – 68	63 – 68	63 – 68
	High Pressure Mode (Kg/cm^{-2})	147	112-147	147	147
Reference Pressure (Kg/cm^{-2})		92-100	92-100	0-1	0-1
Outlet Pressure above Ref. Pressure (Kg/cm^{-2})		14 and 35	14 and 35	14	10- 40
Normal Flow (lpm)		2-20	7-70	2-20	7-70
Accuracy		± 3%	± 3%	± 3%	± 3%
Operating Temperature (°C max)		94	94	94	94

Table 1: Specifications of the valves of DPRVs for 220 MWe and 540 MWe PHWRs

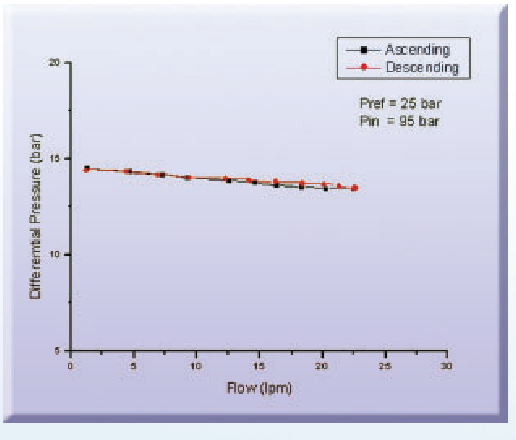


Fig. 4: Effect of flow on differential pressure (Low flow anti-cavitation DPRV)

B) Selection of oil hydraulic components:

1. HTLS motors:

In a 500 MWe PHWR FM, particularly in rams drive mechanism, drastic change has been carried out. In this design, no mechanical advantage takes place inside the water boundary, since water is less lubricative and highly corrosive in nature and the coefficient of friction becomes highly erratic at low speed operations. Because of this, the actuators required to drive the rams and other mechanism should operate at high torque and low speed.

The High Torque Low Speed Motors (HTLS) were commercially available in the market in the early 1980s. A major advantage of the HTLS oil hydraulic motor is that, they are quite compact and generate consistent high torque at very low speed operation (0.1 rpm or below), therefore, gearbox or the speed reducer mechanism is eliminated by these motors. Various designs of the HTLS motors are available in the market. Most of them are radial piston motors of different designs like Eccentric motors, Cam-Curved Motors, Inverted eccentric motors etc.

In early eighties, during the design of new fuelling machine for 540 MWe PHWRs, the HTLS motors were one of the most appropriate drives for the rams and other mechanisms. In Fuelling Machine application, the operating range is quite narrow and far away from the motors' most efficient operating zone. Most of the motors have been designed for heavy earth moving equipment where operating torque and speed are much higher than the requirements for fuelling machine actuators. The data/ characteristics for these kind of motors are not available for the required range. Apart from this, the motors are to be operated in simulated operating condition and checked for pressure, flow, torque and speed relations.

In order to gain confidence in these motors and for evaluation of various types and makes of motors, it was essential to get a first hand operating experience of these new kind of motors, by testing them under their simulated operating conditions and check their performance like running losses, leakage, stalled torque, mechanical and volumetric efficiencies, consistency in torque generation in its full operating range, etc. It was also felt necessary to develop a test code/test procedure for their performance evaluation for comparison of various make of the motors. Various test codes/standards are available as International Standards but these mainly deal with high-speed oil hydraulic motors and are not fully applicable to HTLS motors. Based on the actual application requirements, the test procedures and test set-up details were prepared (Table 2).

Various methods were studied, for evaluating the performance or generating the required information. Following are the areas in which substantial work was carried out, for finding the goal, i.e., acceptance

A new full fledged Motor Test Facility, set-up at the Fluid Power Lab

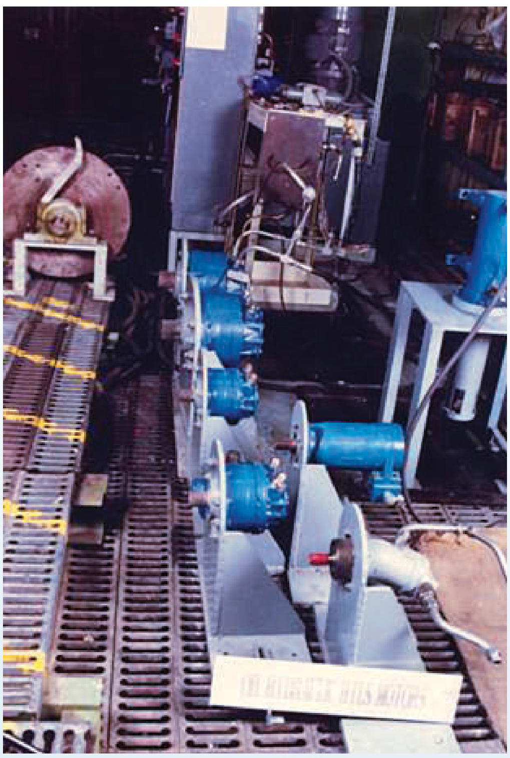
Motor Type	Displacement cc/rev	Torque per 100 bar in m-daN	Maximum power KW	Type	Maximum operating bar	Filtration required
Make-1 model-1	1048	166 m-DaN	50	Radial piston scam curved type	400	20micron absolute
Make-1 model-2	213	34 m-daN	17	Radial piston scam curved type	400	20 micron absolute
Make-2	154	45 m-daN	50	Radial piston eccentric motor	250	20 micron absolute
Make-3	86.9	1.38 Nm/ bar	30	radial piston eccentric Swept volume	250	20 micron absolute
Make-4	80		30	Axial piston Motor German	250	20 micron absolute
Make-5	15.73	15 in-lb./ 100 psi	25	Bent axis axial piston motor	207	20 micron absolute
Make-6	28.1	0.4463 Nm/ bar	25	Fixed displacement bent axis axial piston	320	20 micron absolute

Table 2: Technical Specifications of various motors

of the motors for our specific applications.

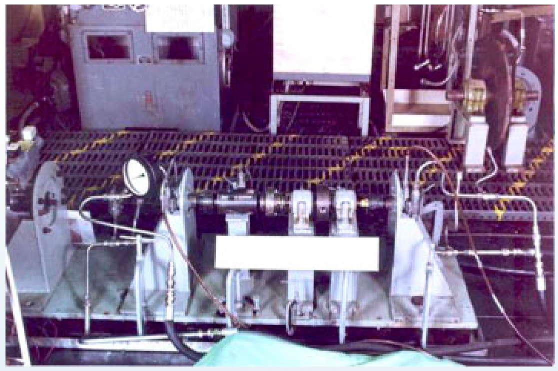
- Study of technical requirements for our application.
- Converting the above requirements in terms of torque, speed, pressure, flow, inertia etc. parameters for the motors, which are working as actuators for the various mechanisms of the Fuelling Machine.
- Development of Testing Scheme for obtaining necessary information under simulated operating conditions.
- Development of the Testing Facility.
- Development of testing procedures.

A full-fledged test facility was setup at the Fluid Power Lab, for testing any kind of motor and linear actuator. The dynamometer for loading various types and size of motors, was indigenously developed, using hydraulic motor working as pump and its hydraulic control circuit. By using this technique, a very accurate, versatile and low cost arrangement was developed which can attain a consistent loading torque of up to 400 Kgf-m, even at 0.1 rpm to full speed i.e. 1000 rpm.



Various types of motors under testing

Brief testing results are indicated in Table 3 and general observations are as follows.



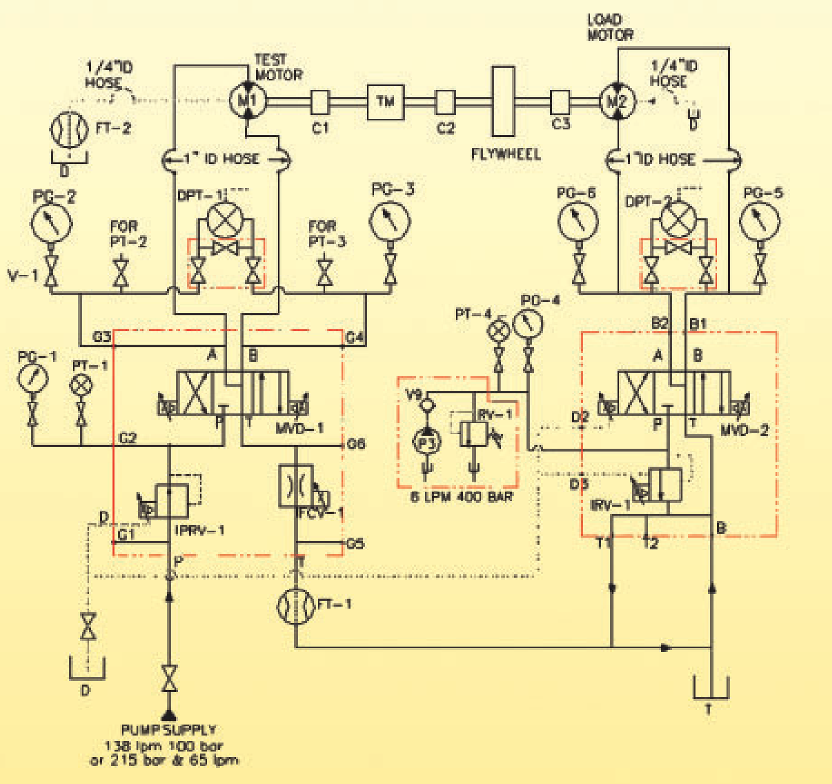
HTLS motor test set-up

1. Normally, the stalled torque generated by the motors is proportional to differential pressure across the motor. But during testing it was observed, that the torque was also affected by a) connected load, b) initial speed of the motor, c) directional valve jogging etc. d) change in the differential pressure with already stalled motor, etc. Therefore the stalled torque characteristics were different for different operating conditions and the variation in mechanical efficiency changed from 20 to 100 %. The stalled torque characteristic with directional valve jogging was very consistent and the fluctuation was within the acceptable range, for some of the motors.

2. Every motor has its most efficient operating zone i.e. a particular speed and a particular load. Near this zone, the motor behavior is very consistent. As the operating point moves away from the efficient operating zone, the performance of these motors becomes erratic.It is a fact, that motors available in the market are designed for some applications, where these motors

No load Test						
Motor Type	Starting pr (Bar)	Drain Leakage (cc/min)	Min Speed (rpm)	Max. variation in running torque % of set value	Stalled torque variation % of set value	Mech. Efficiency %
Make-1 model-1	21bar	150-200	0.11	±2to5%	30to100	20to100%
Make-3	15	10to60	6to9	15%	30%	20to99%
Make-4	15	10	12to15	3%	15to30%	50to95%
Make-2	15	10	0.5	5to20%	15to30%	20to100
Make-5		40	5	2to3	35%	80to100%
Make-6	18	40	4	2to3	35%	80to100%

Table 3: General Testing Results



Motor test set-up

are extensively used e.g. earth moving applications, etc. Where as for handling nuclear fuel, our operating points are far away from these zones, e.g. our operating speed and torque range were only 1% to 30% of rated speed and torque of the motor. In such cases, even manufacturers have not tested their motors and the characteristic data were not available. If a motor is selected just by the catalogue information, its behavior will not meet its intended characteristics at such a low operating point and it would have only during the commissioning of the equipment (i.e. fuelling machine). These test results highlight this phenomenon very strongly and the testing gave confidence in using these motors, for our nuclear fuel handling applications, with proper operating procedures.

2) Testing of Special Oil Hydraulic Components

Electrically modulated oil hydraulic relief valve, flow control valves etc. and linear actuators with position feed back transducers were developed by various internationally well reputed oil hydraulic manufacturers in the early 1980s. No first hand experience was available to the designers about these items. It was decided to test these components thoroughly under simulated operating conditions of fuel handling hydraulic systems. For this purpose, NPCIL purchased some of these electrically modulated valves. RTD, BARC developed test procedures, test facilities and tested these special components for checking their steady state and dynamic characteristics. All the components were successfully tested for their full operating range and all characteristics data were reported to NPCIL in early 1990s.

C) Design and Development of Leak Detector Circuit of 540 MWe Fuelling Machine Using Gas Accumulators

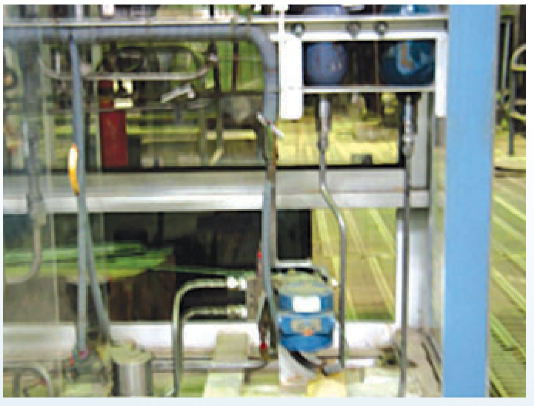
After clamping fuelling machine on to a channel, skinner seal leak test is required to be carried out, before removing the snout plug and seal plugs from the channel, in order to ensure that D_2O leakage from snout cavity to outside (through skinner seal) is within acceptable limits. Similarly, seal plug leak test is carried out after refueling and installation of seal plug into the channel. This is required to ensure that D_20 leak from channel to outside through seal plug is within acceptable limits, before unclamping the fuelling machine.

The leakage is either from snout cavity to outside in the case of skinner seals or from channel to snout cavity in the case of seal plug leak. In the first case, the snout cavity pressure falls where as in the second case, the snout cavity pressure rises.

In 220 MWe FM, the leakage testing is carried out with the help of a special Differential Pressure Transmitter (DPT). It consists of a calibrated bellow. The bellow's volume is calibrated with the differential pressure and the differential pressure range is very small. The increase or decrease in the volume of snout cavity due to leakage from seal plug or leakage from skinner seal respectively, causes the bellows to expand or contract. This movement is calibrated in terms of electrical output and hence the special DPT is calibrated in terms of leakage rate. However, this system requires zero leakage high pressure water hydraulic solenoid valves in magazine and snout cavity lines for filling the snout cavity and across the special DPT for initial equalizing. Due to leaky valves, the differential pressure unit DPU of this DPT may be subjected to high differential pressure so as to cause failure to the DPU of the special DPT.

Earlier in original scheme, the DPT and its isolation valves were located on Fuelling Machine. Failure of the DPU of the DPT brings the Fuelling Machine from the Fuelling Machine vault, to an accessible area for its repair. Installing the DPT and its isolation valves in an accessible area solved this problem.

The special DPT is an imported costly item. For simplifying the leak detector scheme, it was proposed



Leak Detector Circuit installed
at FMTF of Hall 7

to develop a leakage detection circuit using bladder type accumulators.

The principle behind this proposed system is to monitor the change in snout cavity pressure, due to leakage either to the outside atmosphere or from the seal plug to the snout cavity. Accumulators are connected to the snout cavity. These are required to reduce the rate of change of pressure by filling the snout cavity in case of skinner seal leakage or by absorbing the extra flow in case of seal plug leakage. Slow change of pressure will help in accurate monitoring of the leakage rate since it will be easy to calibrate leakage in terms of change in the pressure. The rate of change in the pressure can be co-related to rate of leakage flow in incoming or outgoing to the accumulator. A test facility was developed and the rate of change of pressure was calibrated to the leakage in the cavity. Theoretical estimations were verified by experiments and appropriate size of accumulators and their gas charging pressure selected and a complete leak detector circuit was developed and implemented in the fuelling machine of 540 MWe PHWR.

ANNOUNCEMENT

Forthcoming conference
Role of Analytical Chemistry
in Nuclear Technology
(RACNT-2007)

As part of the Golden Jubilee celebrations of BARC, DAE-BRNS has organized a symposium on the above topic, at BARC Mumbai, between Jan. 4-6, 2007. In this symposium, the role of analytical chemistry in the following areas of current and future programmes of DAE would be covered.

(1) Nuclear Materials
(2) Reactor systems
(3) Thorium Technology
(4) Alternate Energy Sources
(5) Biology, Agriculture and Environment
(6) Water Technology
(7) Isotope, Radiation and Laser Technology
(8) Materials Development and Processes
(9) Development of Analytical Instruments
(10) Reference Methods and Inter-comparison exercises.

The programme would comprise both invited talks and contributed papers (in the form of posters). Authors may send their manuscripts to the Secretary, National Organizing Committee, RACNT-2007 either by e-mail or through a floppy containing the full text of the paper. In addition to the above, two hard copies of the paper should also be sent to the Secretary, RACNT-2007.

Address for correspondence:

Dr. Rakesh Verma
Secretary, National Organizing Committee, RACNT-2007
Analytical Chemistry Division,
Bhabha Atomic Research Centre ,
Trombay, Mumbai – 400 085, India
Tel.:91-22-25592225/2226/5087
Fax: 91-22-2550 5151 / 2551 9613
E-mail: rverma@barc.gov.in
Home page:www.barc.gov.in/symposium/acdsymp.

PRE-SERVICE INSPECTION OF COOLANT CHANNELS OF TAPS 3 & 4

Arbind Kumar, U.N. Tripathi & B.K. Shah

Quality Assurance Division

The TAPS 3&4 are 540 MWe pressurized heavy water reactors containing 392 coolant channels. Each coolant channel consists of a pressure tube (Zr-2.5 Nb), a seamless calandria tube (Zr - 4) concentric to the pressure tube and four tight- fit garter springs (Zr - 2.5 Nb - 0.5 Cu) on the outer surface of the pressure tube, which maintain the required gap between the pressure tube (PT) and the calandria tube (CT). Detection and measurement of garter spring position and CT-PT gap are important, due to the possibility of hydride blister formation and cracking of the pressure tube. Pre-Service Inspection provides the base line data for periodic In-Service Inspection. Canadian Standard CSA N 285.4 and NPCIL Procedure govern the plan for coolant channel inspection. As per these documents, eddy current testing should be employed, for CT-PT gap measurement and garter spring detection. Ultrasonic Testing should be used for wall thickness measurement and volumetric flaw detection in the pressure tube.

Eddy current test procedures for garter spring detection, CT-PT gap measurement and PT ID flaw detection for TAPS 3 & 4 were standardized at AFD, BARC. Eddy current test coils for garter spring detection and CT-PT gap measurement were designed, fabricated and tested. A fixture containing pressure tube and concentric calandria tube was designed and fabricated for calibration of CT-PT gap measurement. Using these coils and CT-PT gap fixture, 20 coolant channels of the two PHWRs: TAPS 3 & 4 were inspected, during pre-service inspection.

Detection of Garter Springs

Differential bobbin coil was designed, fabricated, tested and used for detection of garter springs. The frequency used was 9 kHz, at which the standard depth of penetration is 4.1 mm, equivalent to pressure tube wall thickness. The magnetic field emerges out of the pressure tube and interacts with the garter spring. The phase was adjusted to get the coil wobbling signal in horizontal direction and gain was adjusted, to get the appreciable 3 volt signal from the garter spring. During retraction of

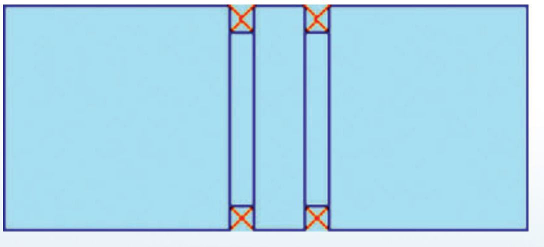


Fig.1a : Differential bobbin coil for garter spring detection

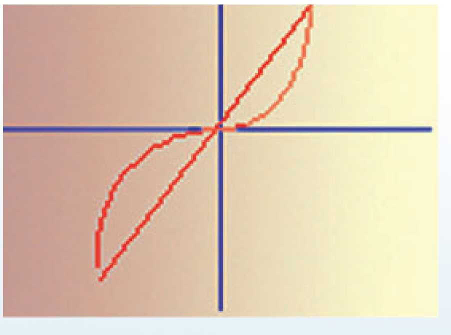


Fig.1b : ECT signal observed from garter spring

the test coils inside the pressure tubes from one end face to another, a lissajuous signal was observed at the garter spring location. The garter spring location was marked by measuring the distance from one end face to the centre of the test coil. The details of the test coil and test signal are depicted in Figs. 1a and 1b respectively.

Calandria Tube (CT) – Pressure Tube (PT) Gap Measurement

Differential plate-type surface probes were designed, fabricated, tested and used for CT-PT gap measurement. There were two probes fitted opposite to each other, on the circumference of the inspection head. The frequency used was 7 kHz, at which the standard depth of penetration is 4.5 mm, more than the pressure tube wall thickness (3.2 mm). The magnetic field emerges out of the pressure tube and interacts with the calandria tube. The phase was adjusted, to get the coil wobbling signal in horizontal direction and gain was adjusted to get the appreciable 3.6 volt signal from the 8 mm CT-PT gap. The test coil was inserted inside the pressure tube in the fixture in such a way, that one coil faced the 3 O'Clock position and another coil the 9 O'Clock position and the calandria tube was moved horizontally to vary the gap. The eddy current signal amplitude corresponding to the varying gap at a step of 1mm was observed and a calibration curve was plotted between gap and signal

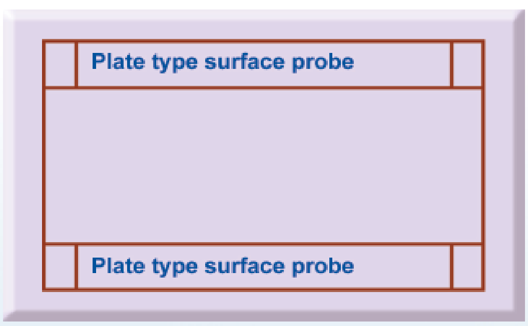


Fig. 2a : Inspection head incorporating ECT plate type surface probe

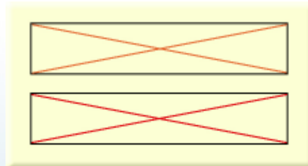


Fig. 2b : Plate type surface probe

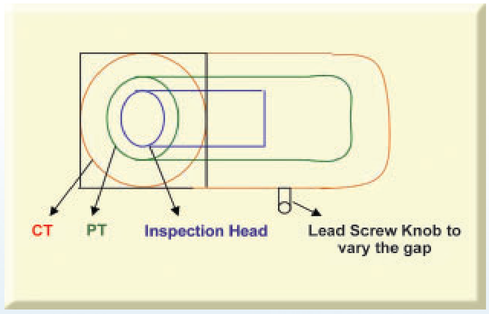


Fig. 2c : CT-PT gap fixture

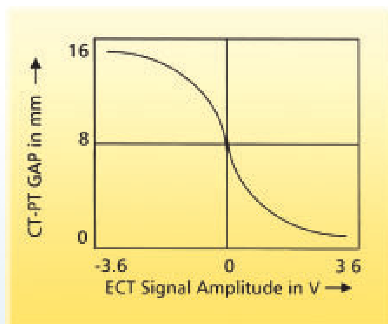


Fig. 2d : Calibration curve for CT-PT gap measurement

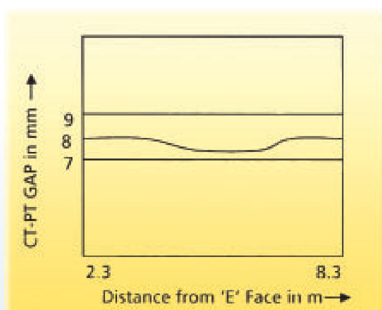


Fig. 2e : CT-PT gap profile for an installed channel amplitude. During retraction of the test coils inside the pressure tubes from one end face to an (another) other, signal amplitude was recorded continuously. The gap was calculated from the calibration curve. The details of the inspection head, test coil, gap fixture, calibration curve and gap curve obtained from the installed channel are depicted in Figs. 2a, 2b, 2c, 2d and 2e respectively.

Pressure Tube ID Flaw Detection

Differential surface probe was designed, fabricated and tested for ID surface flaw detection, in the pressure tube. The frequency used was 300 kHz, at which the standard depth of penetration is 0.5 mm. The reference standard contains circumferential as well as axial flaws (6 mm long X 2 mm wide X 0.066 mm deep). The phase was

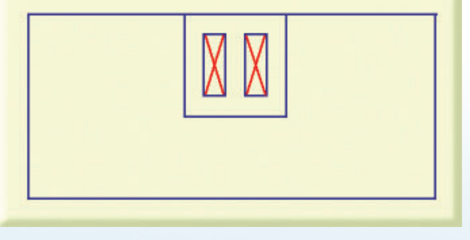


Fig. 3a : Surface probe

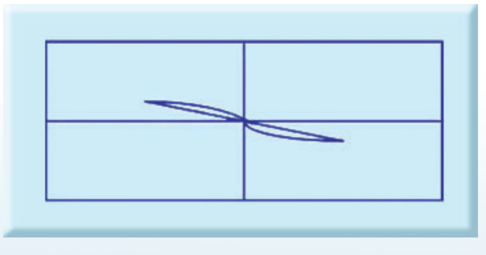


Fig. 3b : ECT signal of surface flaw

adjusted to get the coil wobbling signal in horizontal direction and gain was adjusted, to get the appreciable signal from the surface flaw. The details of the probe and the flaw signal are depicted in Figs. 3a & 3b respectively. This has not been used during PSI, but will be utilized for PT ID flaw detection during periodic ISI.

INTERNATIONAL BLIND PROBLEM EXERCISE ON MODELLING OF PHWR FUEL

A blind problem exercise is being organised on PHWR fuel as part of the forthcoming IAEA technical meeting on 'PHWR fuel modelling" to be held in Mumbai during December 5–8, 2006. The aim of this exercise is to investigate the predictive capability of existing fuel performance analysis codes with respect to fission gas release in PHWR fuel at extended burnup. The current discharge burnup of PHWR bundles is about 7,000 MWd/TU. It is proposed to extend the discharge burnup beyond 15,000 MWd/TU. Computer codes are needed to analyze the adequacy of current PHWR fuel design for extended burnup and to investigate the effect of possible design changes for achieving this burnup. The blind problem for this exercise is based on a PHWR fuel bundle, which was irradiated in KAPS-1 up to about 15,000 MWd/TU and subjected to detailed post-irradiation examination in PIED hot cells facility. The participants of this exercise will be provided with a data package containing all relevant information required for the simulation of irradiation behaviour of this fuel bundle.The package will consist of (i) design, fabrication and pre-characterisation data of fuel, cladding and fuel pins of the bundle; (ii) irradiation power history of the fuel pins and (iii) coolant temperature and fast neutron flux data for irradiation period. Participants of this exercise are required to provide blind predictions of the following parameters using their fuel performance codes and submit the results in the form of plots and Excel data sheets

1. Fuel centre temperature vs Burnup
2. Internal gas pressure vs Burnup
3. % Fission gas release vs Burnup
4. Volume of gas release vs Burnup and at EOL
5. Internal gas pressure at EOL
6. Fuel-clad cold gap at EOL
7. Fuel grain size at centre vs Burnup
8. Radial porosity profile at EOL
9. Cladding oxidation on external surface at EOL

The last date for submission of the results of blind prediction is November 20, 2006. The experimental data of post-irradiation examination on the fuel pins of the bundle will be provided to the participants after the submission of results by them. After the release of experimental results the participants will be required to carry out a revised calculation to match the predictions with PIE results by modifying their codes suitably. The result of this exercise will be presented and discussed in a special session of the IAEA Technical Meeting on PHWR Fuel Modelling to be held in Mumbai during December 5 – 8, 2006. This exercise will help in identifying areas of improvement in the codes for reliable prediction of PHWR fuel behaviour at high burnup.

बीएआरसी न्यूज़लेटर
BARC NEWSLETTER

QUALIFICATION AND CHARACTERISATION OF IN-CHANNEL COMPONENTS OF 540 MWe PHWR

Refuelling Technology Division,

Thirty seven element fuel bundles have been used in 540 MWe PHWR. Since this is a new design, various tests had to be carried out to qualify the fuel bundle. Following tests were carried out at BARC.

1. Pressure drop test
2. Acceleration and impact test
3. Endurance test.

Pressure Drop Measurement across Fuel Bundle and Fuel Locator

A phased experimental programme was proposed and carried out to generate data on pressure drop across the 37-element Fuel Bundle and Fuel Locator of 540 MWe PHWR. Under phase-I, pressure drop across the junction between two fuel bundles was measured for various radial alignments using a transport test section. The effect of diametric creep of fuel channel on the pressure drop was investigated using fuel channels of inside diameters 105.0 and 107.0 mm. In addition to Phase-I experiments, pressure drop across the different designs of fuel locators was measured over the temperature range of 40ºC to 270ºC at low flows in Integral Thermal Facility (ITF) (Fig. 1).

Under phase-II, pressure drop tests at high flows and high temperatures were carried out, using a 103.4 mm ID fuel Channel with 7 fuel bundles in ITF. Pressure drop measurements across the fuel locators at inlet and outlet end fittings with liner tube, having 247 holes, were also carried out along with 9 fuel bundles. In addition to this, the effect of longitudinal creep of fuel channel on fuel locator pressure drop was also investigated experimentally.

Fig. 1: Integral Thermal Facility (ITF)

Test set-up:

A test set-up that simulates the actual reactor condition was designed and fabricated. The test section containing the fuel bundles is connected between two flanges in the facility (Fig. 2).

Test conducted

The experiment was carried out with prototype 37-element fuel bundles, to be used in the reactor, housed in the test section. The pressure drop measurement was made first by increasing the flow rate from the minimum to maximum in steps and then decreasing it back to minimum flow rate in steps and then again by increasing to maximum flow rates in steps. The repeatability of the experimental data was checked. The initial experiments with 7 fuel bundles were carried out at 43°C. From the pressure drop data obtained at this temperature, friction factors were estimated and compared with those obtained during the earlier experiments. After that, the experiment was carried out at high temperature. Using the data of pressure drops at various temperatures and flows, equation for pressure drop of the fuel locator and the fuel bundle, were arrived at by curve fitting.

Conclusion

The total pressure drop at reactor operating conditions across the two fuel locators has been estimated to be equal to 1.85 Bar and 1.99 Bar for the cold and as installed and hot and crept conditions respectively. The pressure drop across 13 fuel bundles at reactor operating conditions has been estimated to be 6.71 Bar.

Acceleration and impact test

During refueling 540 MWe PHWR reactor, when first pair of fuel bundle is loaded, impact is expected to occur, especially in high flow channel. Since flow and acceleration distance in 540 MWe are higher as compared to 220 MWe PHWR, a test to demonstrate that fuel bundle can withstand this impact without significant damage, had to be carried out. The severity of impact test increases with bundle velocity, which depends on the acceleration distance and coolant flow. Hence, it was necessary to carry out this test to specify the design and fabrication methods for mass production.

Test set-up

A test set-up that simulates the actual reactor condition was designed and fabricated (Fig. 3). Impact test set up

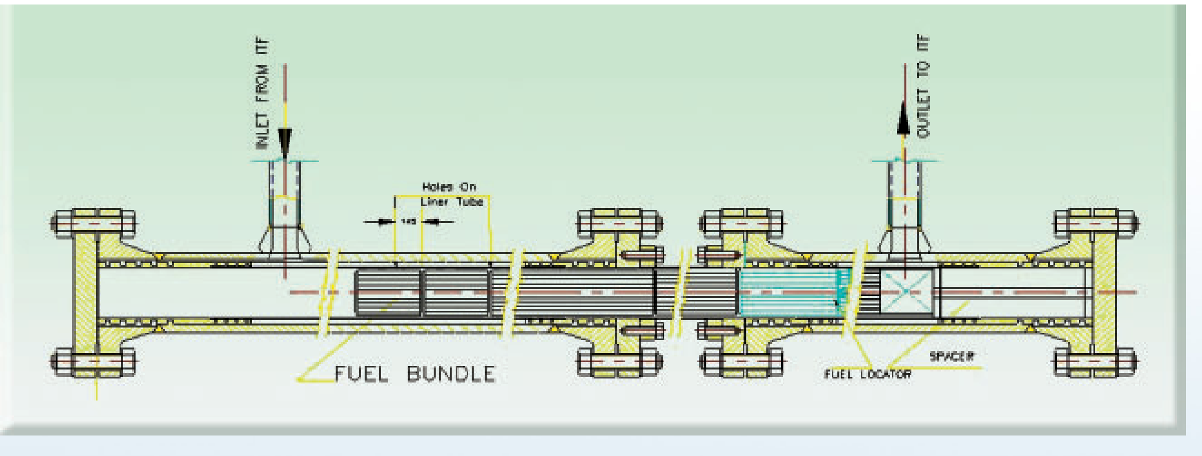
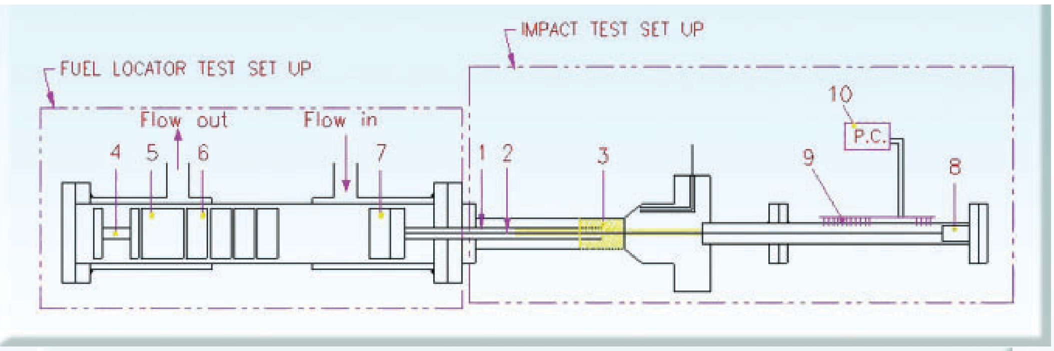


Fig. 2: 540 MWe fuel locator test facility



Part No.	Description	Part No.	Description
1	Ram shaft	6	Fuel bundle
2	Wire cable	7	Moving bundle
3	Ram support ring	8	Magnet
4	Spacer	9	Glass Switch
5	Fuel locator	10	Computer

Fig. 3: Integrated impact test set-up (schematic)

is connected to fuel locator test set-up. The ram shaft is operated by water hydraulic pressure and it pushes the impacting fuel bundle into the cross flow region.

Test conducted

The impact test was conducted at cold condition (temperature:40°C and pressure 90 Kg/cm^2) simulating the momentum of the coolant similar to that of reactor condition. The test flow rate was set at 29 Kg/sec to allow for conservatism, because of pressure tube radial creep, during the lifetime of the reactor.

Conclusion

During the test, terminal velocity achieved by fuel bundle was 2.7 m/sec. However the bundle was seen to be decelerating when it was approaching the stationary bundle and the velocity measured at the time of impact, was 2.45 m/sec. Theoretical estimation of bundle terminal velocity at reactor creep condition, is 2.5 m/sec. Actual impact velocity in the reactor, will be much less than the experimental value as fuel bundle is not likely to accelerate to the full value of terminal velocity. In the experiment, no damage was noticed on the fuel bundle. Based on this result, it is concluded that the Fuel Bundle will be able to take the refueling impact.

Endurance test

Since the flow is more in 540 MWe when compared to 220 MWe PHWR, it was necessary to assess the effect of flow induced vibration.

Test set-up

Endurance test was carried out in Fuel Locator Test Facility. The first bundle of the fuel string will see maximum amount of vibration, due to higher turbulence at the entrance region of the fuel channel. Relatively, the first fuel bundle experiences higher flow excitation in its lowest modes, since it is near the fuel locator and the inlet.

Test conducted

Endurance tests up to 10,000 hours have been conducted simulating the same pressure, temperature conditions as inside the reactor and flow rates in excess of 10% of maximum rated channel flow.
Inspection of the bundle was carried out after every 1000 hrs. The following were done.

1. Visual Inspection
2. Kink Tube
3. Go guage and No go guage.

Till 6000 hrs., there was no visible wear in spacer or bearing pads. After the 7000 Hrs testing, no wear on the bearing pads was observed. Wear on spacer pad was seen on few bundles. But even after wear, more than one mm of spacer was available which is the minimum separations distance, required between pins, as per specifications. Endurance test was continued till 10,000 hrs. and wear did not increase significantly. However, to characterise the vibration, extensive measurement of vibration was carried out, in a test set up with optical window.

Conclusion

The bundle vibrates in twisting/rocking mode. During endurance testing in FLTF, due to the additional degree of freedom offered by free-free liner tube, the first fuel bundle, by the virtue of its position might have had higher vibration.

Therefore, endurance testing at reactor operating conditions and under simulated boundary conditions for the channel components is required, to demonstrate satisfactory fretting performance of the fuel bundle.

After the fixation of liner tube in test set-up vibration was found to be reduced and thus lesser amount of spacer pad wear, is expected.

ANNOUNCEMENT

Nuclear Data for Advanced Nuclear Systems, Nuclear Databases and Applications
NWND-2006

The Department of Atomic Energy (DAE) & the Board of Research in Nuclear Sciences (BRNS) have organized a national symposium from Nov. 8-11, 2006, at the Microtron Centre, Mangalore University, Mangalagangotri, Mangalore. The worshop will comprise of invited talks and contributed papers and also PhD theses (submitted and awarded during the last one year, upto 31st Aug 2006) on the following topics

- Experimental facilities for nuclear data and experiments.
- Surrogate nuclear reactions to generate neutron data
- Nuclear model based predictions of neutron, proton and proton cross sections.
- Atomic masses, measurement and evaluations
- Nuclear model input parameters; RIPL-2 parameters
- All stages of thorium fuel cycle and nuclear data needs.
- Nuclear data for non-nuclear power applications……
- Computerized nuclear data bases for nuclear energy and non-energy applications
- Nuclear data for astrophysics applications and advanced space programmes
- Fission yield data for actinides, measurements and evaluations
- Selected studies of Accelerator Division Sub critical Systems (ADSS)
- Operational requirements in 21st century for a nuclear data centre
- Indian efforts to code Indian experimental data into XFOR format

All correspondence should be addressed to

Dr S. Ganesan, (Convener-NWND-2006)
Reactor Physics Design Division,
Bhabha Atomic Research Centre,
Trombay, Mumbai – 400 085
Tel. : +91-22-2559 5002
Fax: +91-22-2550 5151 / 25519613
E-mail: nwnd2006@gmail.com / ganesan@barc.gov.in

Prof. N. Govinda Nayak, (Local Convener-NWND-2006)
Head, Neutron Centre & Chairman,
Dept. of Studies in Physics
Mangalore University, Mangalagangotri - 574 199
Tel. (O) : +91 8242287363 / (R) : +91 8242288050
E-mail: nerolunayak@yahoo.com

बीएआरसी न्यूज़लेटर
BARC NEWSLETTER

DEVELOPMENT AND TESTING OF FUEL HANDLING COMPONENTS FOR A 540 MWe PHWR

Refuelling Technology Division

Leakage Testing of 540 MWe Seal Plug

The 540 MWe Pressurised Heavy Water Reactor (PHWR) is a horizontal pressure tube type reactor, with Heavy Water as its coolant and moderator. Sealing plug (Figs. 1 & 2) is required, to close the pressure boundary of main heat transport system of the reactor, by preventing escape of heavy water from the coolant channel. There are 392 coolant channels in the reactor,

Fig. 1: 540 MWe seal plug

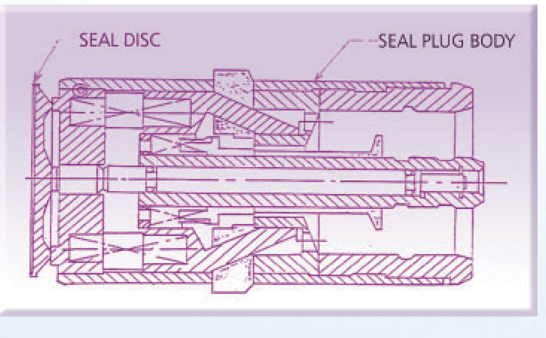


Fig. 2: Cross section of 540 MWe Seal Plug

located in square lattice pitch. Sealing plug closes both the ends of the coolant channel. End fitting has a stepped bore, to create a sealing face. Sealing plug is held through its expanded jaws, in a specially provided groove of the end fitting.

The plug was designed, fabricated and used in TAPS 3 & 4. The plug was designed considering its functional importance, intricate design and precision machining requirement. A sealing plug consists of about 20 components, mostly made of precipitation hardening stainless steel, suitable for water environment and also meets other requirements of strength and resistance to wear and galling. Seal disc is a critical component of the sealing plug, as it is a pressure-retaining component. It is a circular disc with protruded stem. The sealing face of the disc is nickel plated in the peripheral area, that creates the sealing, by abutting against the sealing face, provided in the end fitting. The typical shape and profile of the seal disc provides flexibility and allows elastic deformation to assist in the locking of the sealing plug and creating adequate seating force to effective sealing. It was necessary to carry out actual functioning of the plug and leak test, prior to use at the site. In a test conducted at simulated reactor condition (Fig. 3), the performance of the plug was found to be satisfactory. This report is about leakage testing of the 540 MWe PHWR sealing plug (Fig. 4), performed at simulated reactor conditions.

Testing of Water Lubricated Bearing

Newly sourced Water Lubricated Bearing (WLB) (Fig. 5) for a 540 MWe Fuelling Machine, required qualification testing. A test set-up was designed and fabricated and

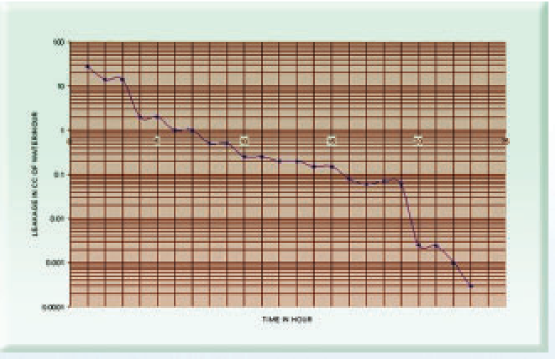


Fig. 3: Leakage monitoring of seal plug

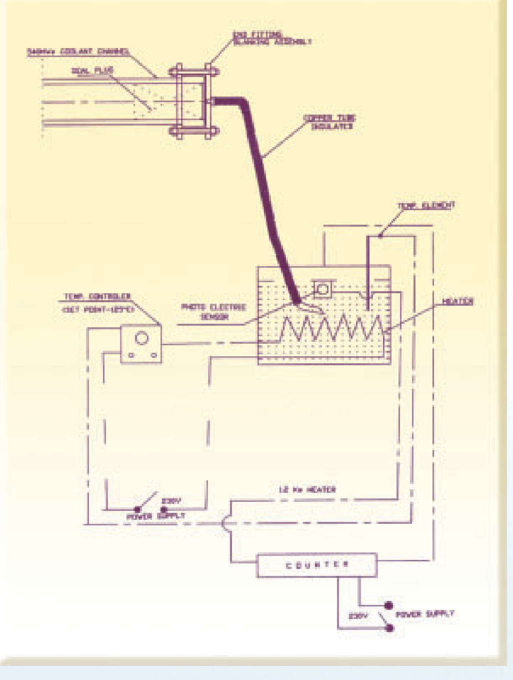


Fig. 4: Sealing plug leakage monitoring by bubble counting method

two bearings were tested for failure. The test was monitored using a Shock Pulse meter and an accelerometer. In one case, the shock pulse meter readings indicated bearing failure. However, the other bearing was found to have failed, when the rig was opened to solve water leakage. In both cases, the bearings had failed within 30 % of their expected life.

Fig. 5: Water Lubricated Bearing (WLB) 2

Both the bearings were found to have through circumferential cracks on both races as shown in the photograph (Fig. 5). The test bearing was subjected to metallurgical analysis, where the microstructure revealed non-uniform distribution of undissolved carbide. Thus the failure was attributed to the improper microstructure of the bearing race material.

Performance evaluation of special extension

The upstream fuelling machine needs a ram extension to push the bundles in 220 MWe PHWR design. Owing to much longer channel lengths, the 540 MWe design uses a special extension instead of more numbers of normal extensions. The special extension tool generates enough defuelling force using pressure drop across itself.

It was required to establish hydraulic characteristics of special extension to ensure successful performance in lowest flow channel and the highest flow channel.

The test was carried out using locator test set-up (Fig. 6). The Pressure drop v/s flow characteristics (Fig. 7) was developed for cold and hot conditions. The performance of the special extension was found to be satisfactory.

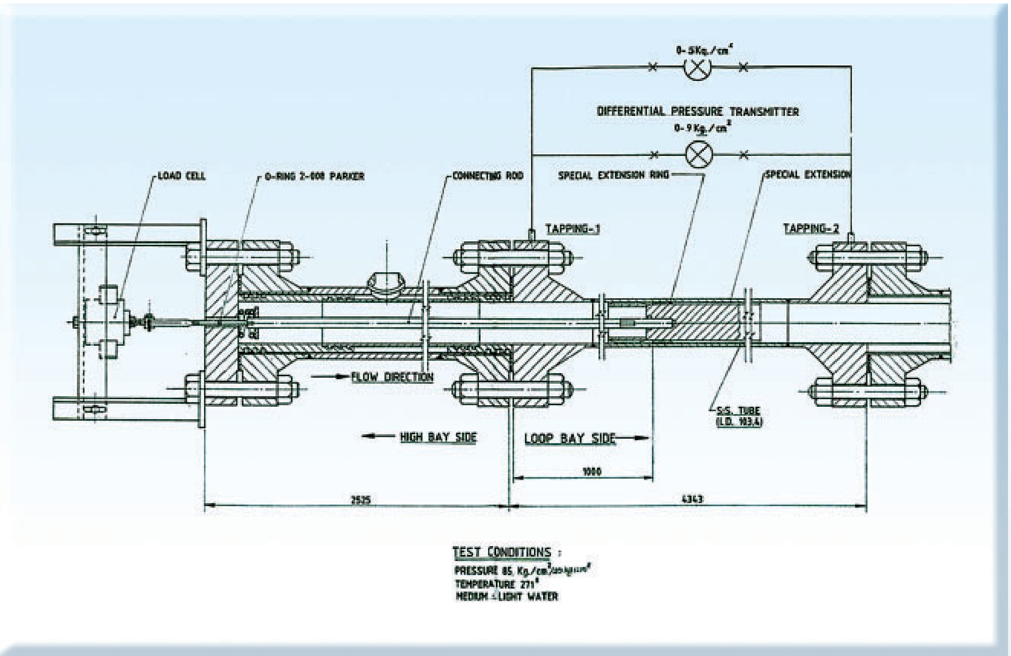


Fig. 6: Schematic diagram Special Extension Testing Phase -1

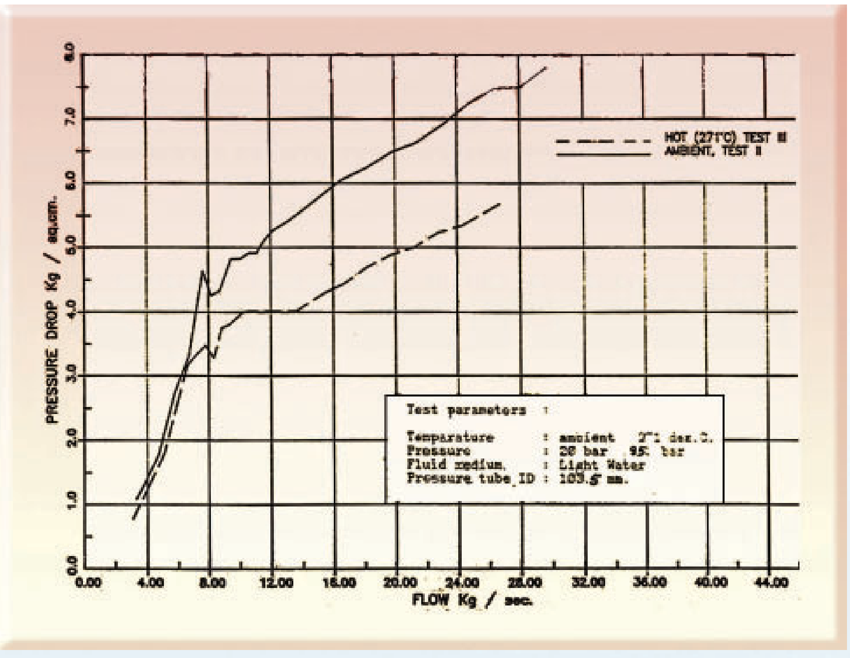


Fig. 7: Pressure drop versus flow characteristics

Development and testing of vlier plunger for fuelling machine

A vlier plunger (Fig. 8) is required to hold the components against drifting. These are used in FM. A test set-up (Fig. 9) was fabricated and commissioned, to test the vlier plungers. An alternative design was worked out and tested for giving higher force.

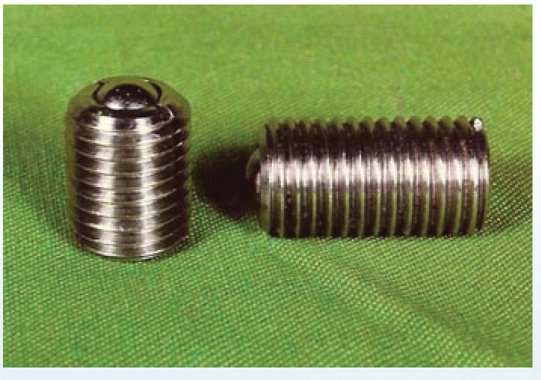


Fig. 8: Vlier plunger

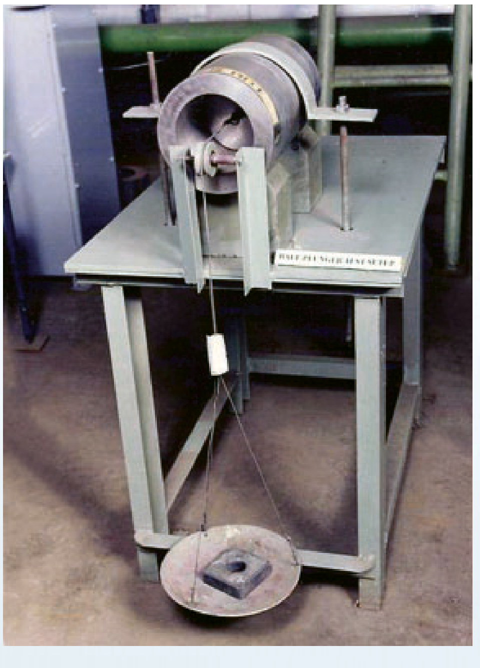


Fig. 9: Test set-up for vlier plunger

Design Verification of Fuel stop and Magazine Fuel tube

In 540 MWe fuelling machine, a fuel tube is used in the magazine. These fuel tubes, four in number, are located in the magazine tubes. There are moved during the refueling operation. Fuel stops are used, to support the fuel column in the fuelling machine. To simulate these components, a fuel stop rig (Fig. 10) was fabricated and commissioned. The following tests were conducted for freezing the design:

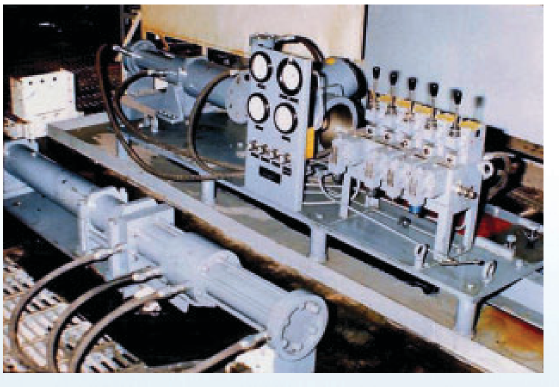


Fig. 10: Fuel stop rig

- Side stop compatibility test to decide radius of curvature, for having proper contact with 37-element fuel bundle. The combination was tested up to 2800 kg load. Based on the experiment, it was concluded that a single fuel stop cannot withstand the fuel column load and hence such use was prohibited.

- Gap jumping test of fuel bundle, to find out, the maximum gap which a 37-element fuel bundle can jump. This was used to decide the gap to be provided in the design.

- Slippage test of the fuel bundle.

DESIGN AND DEVELOPMENT OF SPENT FUEL TRANSFER SYSTEM FOR 540 MWe PHWR

Refuelling Technology Division

On-power fuelling is a special feature of Pressurized Heavy Water Reactors (PHWRs). Fuelling Machines (FMs) are deployed to charge the fresh fuel into the reactor and to discharge spent (irradiated) fuel from the reactor. Spent Fuel Transfer System facilitates the transport of spent fuel from reactor building to the spent fuel receiving bay, located outside the reactor building. NPCIL entrusted the job of design and development of the Spent Fuel Transfer System (SFTS) to the Refuelling Technology Division. The work was completed and system design was handed over to NPCIL for manufacturing. The system was manufactured and commissioned by NPCIL at TAPS 3 and 4. Support was provided to NPCIL during fabrication and commissioning. Up to the end of May 2006, about 200 pairs of spent fuel bundles have been transported to the bay in TAPS 4. The performance of the system was found to be satisfactory. The SFTS for TAPS 3 has also been successfully commissioned.

System Description

A typical PHWR has two identical Fuel Transfer (FT) systems, each dedicated for “North” and “South” side of the reactor. Major FT equipment are located in the FT room, adjacent to the reactor. Transfer Magazine (TM) receives the spent fuel from the FM. The TM has heavy water as medium, as it has to interact with the FM. The fuel is received at 97,500 elevation in the TM and needs to be lowered down to 91,450 elevation. This elevation makes it possible to transport the spent fuel under the shielded trench below the working floor of 93,000 elevation. Spent Fuel Receiving Bays (SFRB) and Spent Fuel Storage Bay (SFSB) are located away from the reactor building Figs. 1 and 2. The spent fuel is transported to SFSB for storage. The bays are light-water filled. Hence an additional equipment called Shuttle Transfer Station (STS) is used, down the

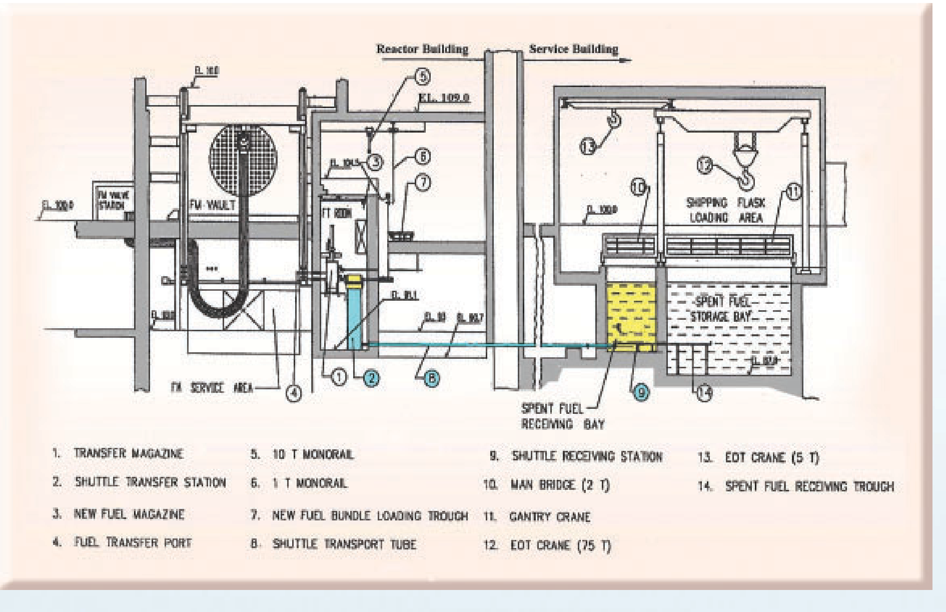


Fig. 1: Spent Fuel Transfer System (schematic)

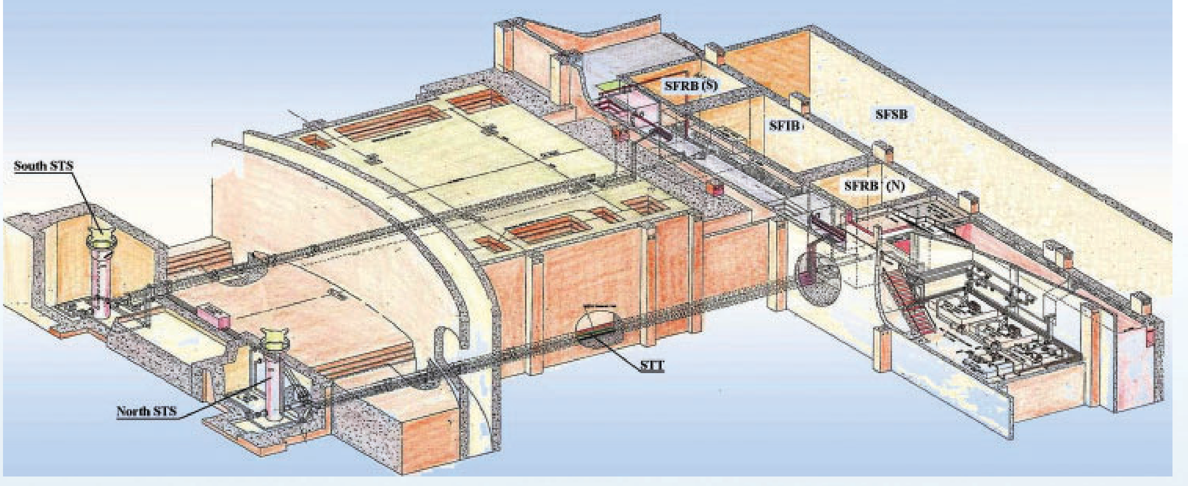


Fig. 2: Spent Fuel Transfer System (SFTS) 540 MWe

line of Transfer Magazine (TM), which has light water as medium. STS receives fuel at 97000 elevation from TM. A shuttle transport system connects STS on one end and storage bay receiving equipment on the other end. It facilitates the transport of spent fuel by virtue of light water flow, through a shuttle transport tube. The SFRB equipment receives the spent fuel and facilitates their further transport to SFIB occasionally for fuel inspection or to adjacent SFSB for storage. The fuel is received in the SFSB on a trough, from which fuel is loaded manually and kept in the fuel tray position adjacent to trough. These trays are transferred to the storage area once it is full. SFTS comprises of STS, Shuttle Transport System, SFRB equipment and spent fuel bays.

Concept Evolution

Lay out

Compared to 220MWe PHWR, the lay out of SFTS was improvised. Table 1 gives the major changes:

Concept Evolution of STS

Shuttle Transfer Station is a cylindrical vessel Figs. 3 and 4. It is 1.3 m. in diameter and 6.3-meter in height. Its carriage receives the empty shuttle from the bay against the buffer. Position sensors confirm the shuttle receipt and its correct position. It is hoisted up to align with the TM to receive the fuel. Hydraulic rams are mounted on STS to facilitate the receipt of fuel in the shuttle. Subsequently it aligns with shuttle transport tube for sending the shuttle to SFRB.

Apart from the changes dictated by layout changes, the feedback of previous designs was also taken into account. Some of the major problems faced in 220 MWe design were the possibility of mixing of STS light water with TM heavy water, unreliable underwater switches and difficulty in maintenance, ball screw maintenance etc.

The STS for 540 MWe has been divided into two parts called top housing and bottom housing. This has made STS an open tank instead of a pressure vessel. This helps in absorbing the thermal expansion and in avoiding mixing of light and heavy water. It uses a simple rope hoisting mechanism to make fabrication easier and cheaper as compared to a ball screw option.

A hydraulic cushioned sleeve ram makes it possible to take out all position sensors outside the water environment,for quick replacement and better reliability. It facilitates the positioning of shuttle accurately. It also enables STS to be an open tank concept.

Item	220 MWe PHWR	540 MWe PHWR	Remarks
Shuttle Transfer Station	4m height x 1.14m dia	6.3 m height x 1.3 m dia	Long STS needs to be divided in to two parts
Shuttle Transport Tube	70-100 meter tube with bends	45 meter long tube without any bends	Straight tube avoids possibility of shuttle stuck up
SFRB equipment	• Common receiving station for "north" and "south" fuel transfer system • Equipment in different planes	• Separate receiving stations for "north" and "south" fuel transfer system • All equipment in same horizontal plane	It increases the reliability and availability of equipment
Spent Fuel Receiving Bay	Common receiving bay for one unit	Separate receiving bays for north and south fuel transfer system	Individual bay can be isolated
Spent Fuel Inspection Bay	Inspection activity done in receiving bay	Separate spent fuel inspection bay in between north and south receiving bays	Better inspection facility for fuel.

Table 1 : Layout of SFTS

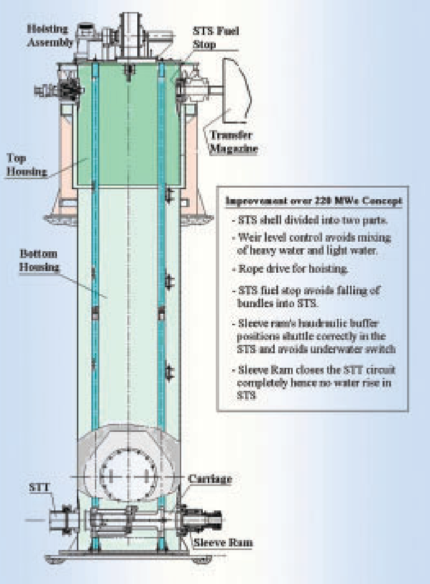


Fig. 3: Shuttle Transfer Station (STS)

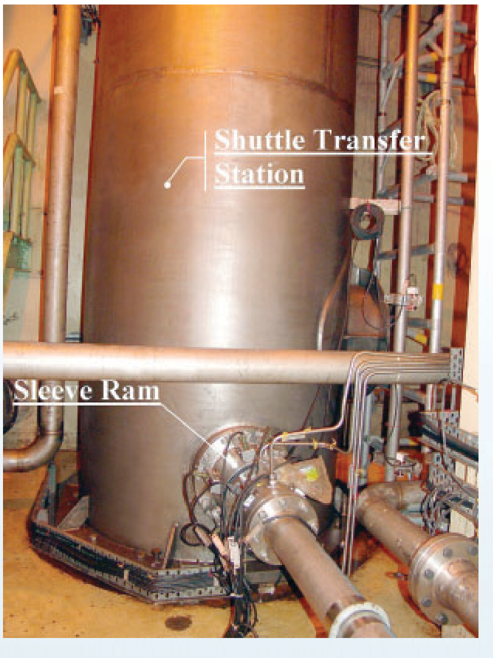


Fig. 4: Shuttle Transfer Station (STS)

In 220 MWe STS, incidents of dropping the fuel inside STS, creating an accidental situation, have been reported. Hence, the need for a mechanical interlock was felt, to receive fuel only in presence of shuttle. Hence STS fuel stop has been so designed, that it opens the fuel passage only when shuttle is physically present at right location.

Fig. 6: Water hydraulic valve station

Concept Evolution of Shuttle Transport System

Shuttle Transport System comprises a shuttle, shuttle transport tube and associated light water process system to provide flow for shuttle transport system and actuator system. Shuttle transport system provides the flow for moving the shuttle inside the STT using shuttle transport pump (Fig. 5) and gang operated butterfly valves. Actuator supply system has been provided for actuation of rams in SFRB/SFIB/FT room. A water hydraulic valve station (Fig. 6) operates actuators in SFRB/SFIB/FT room.

Fig. 5: Shuttle transport pump

The shuttle is a cylindrical container, which houses a pair of spent fuel bundles. Shuttle is supported on two bearing rings (Fig. 7), which are guided in the tube. Shuttle is hollow, to provide flow through the spent fuel. Projected area of bearing rings gives the drag effect. Pair of fuel is locked inside the shuttle by six pawls on the open end of the shuttle. Shuttle can also be used to transport either special ram extension or fuel locator to SFRB. Due to bigger size of 540 MWe PHWR fuel, the shuttle diameter is increased. In addition to this, several design features were improvised based on 220 MWe PHWR experience viz. use of a fuel pusher to provide full face support to fuel and to eject out disintegrated pencil, to strengthen nose piece and pawl mechanism, to provide positive locking of all fasteners and features to facilitate rotation of the shuttle.

Detail design of shuttle was worked out accommodating design features mentioned previously. Critical calculations were

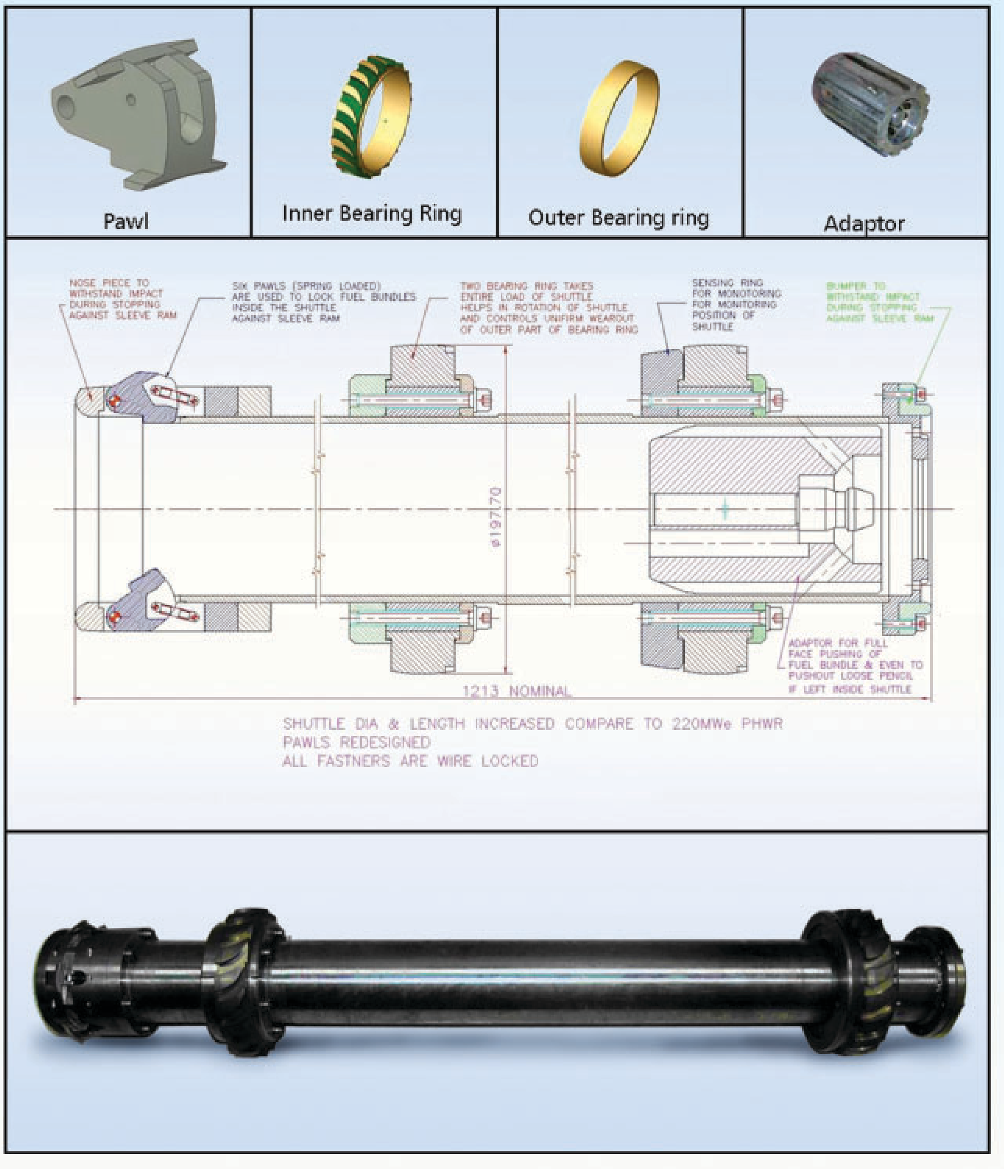


Fig. 7: Shuttle assembly

performed to know the flow requirement to move the shuttle. It was felt that calculations are complex due to various parallel flow pass across the shuttle. Hence, a in-house computer programme was developed to generate hydraulic data of shuttle, like pressure drop, drag force, minimum flow required etc. It was seen that design similar to a 220 MWe PHWR could not result in shuttle rotation. Shuttle rotation is desirable to have uniform wear on the bearing pads.

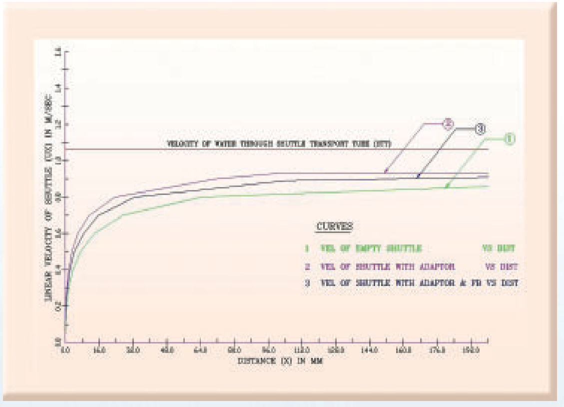


Fig. 8: Linear velocity distribution of shuttle for 2000 L.P.M. flow

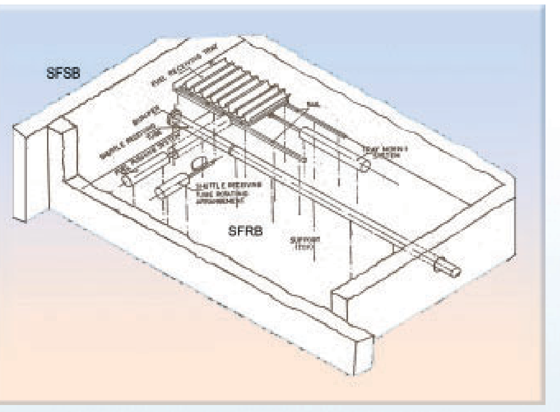


Fig. 9: SFRB layout with indexed tray mechanism

Shuttle Transport Tube is straight having a number of Victaulic clamps to observe thermal movement. Design features are provided for containment isolation of the shuttle transport tube and other fuel handling pipes passing through the reactor building.

Concept Evolution of SFRB equipment

Shuttle transport tube ends in the spent fuel-receiving bay. The shuttle is decelerated and received in Shuttle Receiving Tube (SRT). After opening the pawl of shuttle, the fuel is ejected by hydraulic ram on a trough and pushed further for transferring to SFSB or SFIB. The complete spent fuel transfer operations in SFRB are done remotely (Figs. 9 and 10).

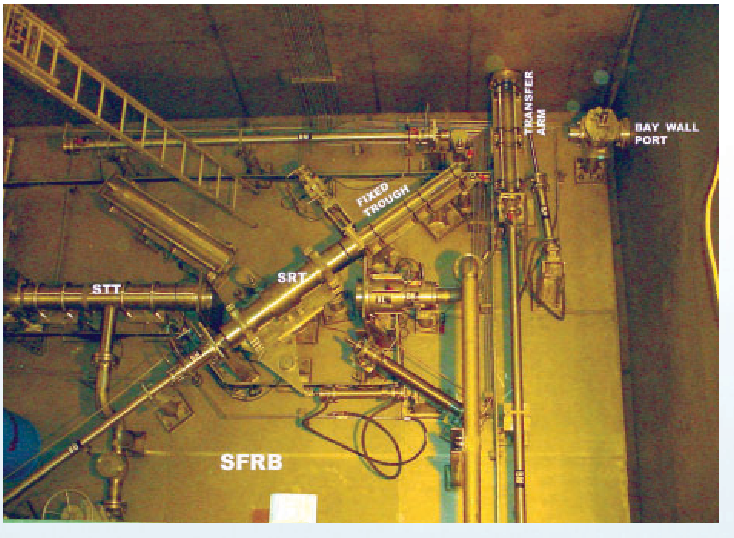


Fig. 10: SFRB equipment TAPS-4

The bay equipment is designed, such that, all the fuel transfer operations in SFRB take place in one horizontal plane. The whole spent fuel movement operation in SFRB is visible from the top of the bay.

A concept of indexed tray was proposed, to receive eight spent fuel bundles in the tray in SFRB and then transfer the whole tray to the SFSB, through a rectangular opening between SFRB and SFSB. Two gates were provided to close the opening during draining of SFRB. The whole tray with eight spent fuel bundles is lifted and stored in SFSB. This concept was not considered by NPCIL because of large opening between bays and problems in handling of gates. Subsequently, the bay equipment was redesigned considering receiving of one pair of spent fuel bundles in SFRB and transfer it to SFSB through bay wall port.

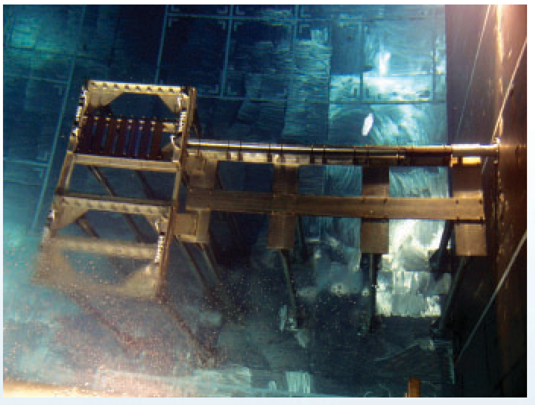


Fig. 11: Open trough and bundle receiving tray in SFSB

Fig. 12: Shuttle Test setup

Shuttle removal is made easy from the system; a ball valve is used for isolating the SFRB from SFSB/SFIB during the draining of the SFRB to get leak tight barrier, these ball valves are located in SFRB for easy accessibility for maintenance, hydraulic cylinder are made single-stage in place of multi-stage for ease of fabrication and reliability.

Design and Development Activities

Design and Development of Shuttle

An experimental shuttle test set-up (Fig. 12) was designed and fabricated. It simulates the full-scale size of the shuttle transport tube with Victaulic joints and by- pass loops at either end to decelerate the shuttle. There is a provision in the set-up to monitor the flow, position, pressure drop, time etc. related to shuttle movement. By using the test set-up, design of shuttle was validated. Specifications of shuttle transport pump, butterfly valves were finalized, based on this pressure drop and calculation was done for the actual piping layout of the reactor.

A dummy shuttle was fabricated and used in the shuttle test set-up to increase the life of the shuttle bearing rings. Various alternative means were tried for increasing the life. Use of lead buttons and lead-filled slots of bearing rings showed increased life. Ultimately a new design bearing ring having curved slots were evolved which can give rotary effect on the shuttle (Table 2). This facilitates the shuttle rotation at the start of the movement.

Sr No.	Bearing ring configuration	Bearing ring wearout rate for 100 reactor channel of operation		Remarks
		Witout lead (mm)	With lead (mm)	
1.	Bearing ring with 22.5 deg inclined slots	2.16	0.30	No rotation of shuttle was observed
2.	Alternative Bearing ring (curved slots)	0.29	0.14	shuttle rotates at the starting of the movement

Table 2: Comparison of Wear out rate

Other development activities

To prove the design full-scale models were fabricated and tested for performance evaluation and endurance tests. The following components (Figs. 13-16) were tested:

1. Weld set-up
2. Full scale Model of Transfer Arm
3. Shuttle Transfer Station Fuel Stop
4. Test set-up for Pawl Opening Ring

Fig.13: Welding mock-up

Fig. 14: Pawl assembly of shuttle

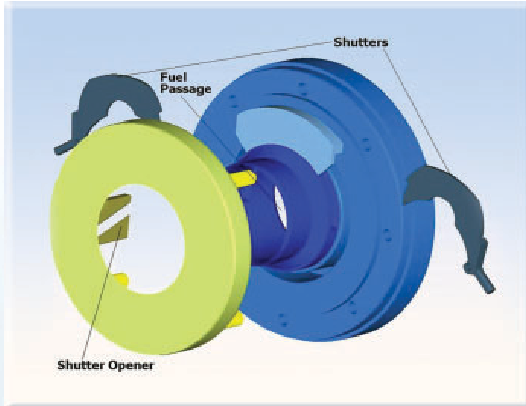


Fig. 15: STS fuel stop (exploded view)

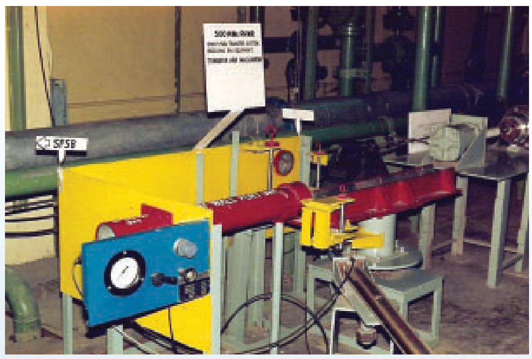


Fig. 16: Full scale model of Transfer Arm

Detail Design

Detail design of the SFTS was taken up. Challenges in the design of a SFTS are that, spent fuel is highly radioactive, hence it has to be handled inside shielding. The mechanisms need to be completely remotely-operated. Design must consider defense in depth. Each likely abnormal situation needs to be visualized in detail, to provide built-in remedies. All safety principles like equipment redundancy, avoiding common mode failure, controlled activity release, multiple barriers to avoid probable activity release etc. had to be considered.

Initially as the design evolved, many updates of the drawings were required. Continuous discussion with

NPCIL was done. Drawings formats were finalized in view of standard practice and relevant codes. Drawing preparation began on drawings boards, However, AutoCAD was up coming as a tool for drafting. The tool was adopted, customized to suit the mechanical drawing preparation, draftsman was trained and drawing preparation was taken up, using this facility. As this system was quite complex in nature, wherever required a three-dimensional models were prepared for fuel stop, carriage assembly, sleeve ram etc. This helped in identifying many constraints, which ultimately facilitated manufacturing, without significant deviation.

Following deliverables were handed over to NPCIL:

- Design Drawings
  - Shuttle transfer station: 160 Drawings
  - Shuttle transport tube: 32 Drawings
  - Receiving bay equipment and bay wall ports: 200 Drawings
  - Shuttle assembly: 16 Drawings
- Flow sheets of light water process system: 2 Nos.
- Piping and tubing layout: 10 Nos.
- Sizing of EP and drawings: 10 Nos.
- Design basis report for SFTS
- Design basis report for containment isolation
- Design basis report for inspection and canning facility
- Seismic analysis of SFTS with the help of RSD
- Development of shuttle technology
- Layout finalisation and comments on civil drawings.

Conclusion

The Spent Fuel Transfer System design has been proven to be satisfactory at TAPS 4. Precise detail design, exploration of auto cad and 3D packages, design verification by experimentation and continuous interaction with other groups has made it possible for almost deviation-free manufacturing and commissioning of the system.

ANNOUNCEMENT

Forthcoming Symposium
Life Sciences Symposium
LSS - 2006

DAE/BRNS has organized a three day symposium, from Dec. 18-20, 2006, at the BARC Training School Hostel, Anushaktinagar, Mumbai. The theme of the symposium would be 'Trends in Research and Technologies in Agriculture and Food Sciences. " The programme includes invited talks, poster and paper presentations. The forum for discussion would focus on:

- Trends in conventional and molecular plant breeding to tackle new challenges
- Enhancement of crop quality, particularly nutritional and nutraceutical factors and processing quality
- Trends in use of bioregulators to enhance crop performance
- Biochemistry, physiology and genetics of major processes influencing crop productivity
- Food preservation for domestic consumption and export
- Trends in processing and packaging
- Handle technologies
- Biochemical basis of shelf-life extension of fruits and vegetables
- Scientist-grower-technologist-processor-user interactions.

For further details, please contact:
Dr Narendra Jawali
Convener, LSS - 2006,
Molecular Biology Division,
Bhabha Atomic Research Centre,
Trombay, Mumbai – 400085.

बीएआरसी न्यूज़लैटर
BARC NEWSLETTER

THE ROLE OF CDM IN THE POWER PROJECTS TAPS 3 & 4 540 MWe (PHWR)

M.D.P. Gupta, T.L. Govindankutty, H.B. Panse, S.B. Jawale and R.L. Suthar

Centre for Design and Manufacture

During the prototype development of various 540 MWe PHWRs and the manufacture of reactor components and assemblies, the CDM has been involved, right from the design and drawing stage. In the initial period (from 1987 onwards), many prototype components were manufactured to validate conceptual design. Manufacturing feasibility and modifications for maintaining quality and ease of manufacturing, was provided to 540 MWe design groups.

CDM successfully manufactured the following assemblies and components and delivered them to NPCIL. CDM's contribution to R&D activities in nuclear power reactors are many and varied and this will continue for various other projects.

(1) End Fitting bodies – 1700
(2) Liner tubes – 1700
for coolant channel assemblies.
(3) Ram assemblies – 5
4) Pressure Housing – 5
(5) End covers – 5
for fuelling machine assemblies.
(6) Electrolysing of Sealing Plug Jaws - 1750
(7) Fuel Transfer Ports – 5

End Fitting Bodies

This is a critical, intricately shaped component of the coolant channel assembly of a reactor. This component was finish-machined out of raw material of 220 Ø solid forging of SS 403 (ESR modified quality, heat treated to 240 BHN) to the final size O.D. 188.5 mm x I.D. 137.1 mm bottle bore x 2516 mm long. Twenty main machining operations are involved from the beginning to the end of manufacture. U.T. flaw detection, thermal stress relieving, wet magnetic particle testing and hydrostatic testing are carried out as per applicable codes for nuclear components. Bottle boring head (adaptable to deep hole boring machine to machine the bottle bore in one pass, using pull boring technique), Trepanning head, Boring head, a special high range Honing head, burnishing tool etc. were developed indigenously at CDM, for successful manufacture and completion of the component. The sealing surfaces were burnished to improve the surface mechanical properties, viz., surface finish 0.06/0.08, micron CLA, surface hardness increased upto 15%. The manufacturing processes developed for manufacture of End Fitting bodies of 220 MWe PHWRs were transferred to a vendor, to carry out the manufacture of End Fitting bodies of 220 MWe PHWRs.

End fitting bodies

Linear Tubes

These are manufactured out of seamless stainless steel (SS-410 grade), to the final size of O.D. = 111.86 mm, I.D. = 104 mm and 2135 mm long, achieving IT7/IT8 grade tolerances, with stringent geometrical features, like 0.01mm/100 mm straightness etc. For a thin section cylindrical component of such a large length, maintaining dimensional and geometrical features was a difficult task. This was maintained by developing suitable toolings, fixtures and processes.

Linear tubes

RAM Assemblies for Fuelling M/C Heads

The Ram assembly, the main assembly of F/M Head, consists of various large size, precision and critical components like, Front Housing, Rear Housing, Ram drive body, B'Ram-Pinion Drive Housing, co-axial (telescopic) and horizontally mounted rams, i.e. B-ram, A latch and C-ram. A large number of toolings, jigs, fixtures, processes etc. were developed, to manufacture the major components. Front Housings of size 330 I.D x 3300 mm long with IT-6 grade and surface finish 0.4 micron was achieved, with a specially developed Honing head. This component was machined out of forged SS 403 cylinder with integral end flanges.

There were 135 components to be manufactured at CDM, numbering a total of about 500 parts for each assembly. Most of the components were with IT-7 grade tolerances and stringent geometrical requirements.

Ram assembly

Elaborate manufacturing processes, quality control systems, follow-up, planning and development was done, to get the desired quality of components manufactured. The development of manufacturing processes for this assembly, was a challenge and CDM satisfactorily completed its manufacture, in all respects. Five Ram assemblies were delivered to NPCIL.

Pressure Housing and End Covers

These large-size heavy components, form the pressure boundary components of Magazine assembly joint, with gray-lock clamps and are very critical in nature. The accuracy called for, is in the range of IT-6 grade. The raw materials supplied, were in the form of SS 403 forgings, weighing approx. 1700 kg. and 2800 kg. respectively. The finished product weighs approx. 1000 kg. and 1400 kg. respectively. The first stage of semi-finish machining, involves heavy machining operations, milling, turning, contour machining and CNC operations. Further final machining operations are carried out, on CNC-Horizontal Boring machine, CNC Vertical Turret Lathe, CNC-Universal Milling m/c and Jig Boring m/c. Quality control procedures for dimensional and geometrical features were also meticulously planned and executed. A large number of jigs and fixtures were designed, developed, fabricated and used successfully for setting of jobs, machining precision features. All the five sets of components were completed and handed over to NPCIL for assembly of fuelling machines as per schedule.

Pressure housing and end cover

Electrolysing of Sealing Plug Jaws

CDM has been carrying out electrolyzing of Sealing plug jaws for NPCIL, for various power projects and has successfully completed, 1750 Sealing plug jaw sets for TAPS 3 and 4. Elaborate electrolyzing procedures were planned and developed. Inspection after electrolyzing demonstrated uniformity and form accuracies on coating thickness 0.008 mm.

Sealing plug jaws set

Fuel Transport Port

These were similar to End Fitting bodies with some added critical features. These were also manufactured and handed over to NPCIL, for installation and commissioning.

ANNOUNCEMENT

Seminar on

New Dimensions of Health Care and Protection

November, 2006

The Hindi Sahitya Vigyan Parishad, BARC and the Indian Nuclear Society, Mumbai, have organized a two day seminar, at the Rani Durgavati University, Jabalpur, Madhya Pradesh. The following topics will be deliberated upon, by experts, during the seminar:

- Use of radioisotopes for medical diagnosis and treatment and the development of the radioimmunoassay method.
- Generation of radioisotopes in nuclear reactors and accelerators
- R&D on radiation treatment, brachytherapy, teletherapy and telepathology
- Labeling of radioisotopes
- Radio-imaging: cyclotron induced PET
- Laser surgery
- Genes and stem cells for therapeutic purposes
- Sterilization of biomedical equipment and material through irradiation
- Radiation management and human health
- Gynecologic disorders in the context of the 21st century
- Tobacco and cancer
- Mental health
- Rapid strides towards treatment through ayurveda
- Yoga and health

For accommodation and further details, please contact:

Dr T.N. Shukla
Convener, Local Organizing Committee
Head, Hindi & Linguistics Dept.
Rani Durgavati University, Jabalpur
Tel. (O) : 0761-2600567 Ext. 259
(R) : 09425044685; 0761-2361640
Fax: 0761-2600632 / 2603752
E-mail: adovccl@rediffmail.com

Mr S.K. Agarwal
Head, Reactor Operations Division
Bhabha Atomic Research Centre
Trombay, Mumbai – 400 085
Tel. (O) : 022-2559 4610 / 25594618
(R) : 022-2551 4437
Fax: 91-22-2550 5311
E-mail: agarwal@barc.gov.in

PRE-SERVICE INSPECTION OF TAPS 3 & 4

Division of Remote Handling and Robotics

Visual inspection and sag measurement of coolant channels of TAPS 3 & 4 was carried out as part of Pre-Service Inspection (PSI) before startup of the reactors.

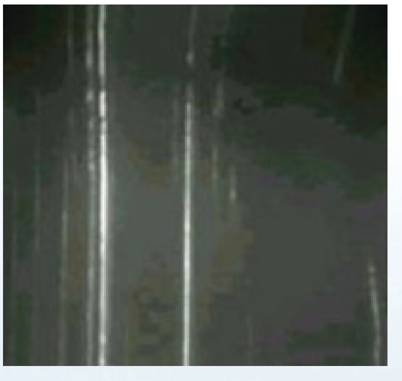
Visual Inspection of coolant tubes of TAPS 3 & 4

Visual inspection of 20 specified coolant channels was carried out, as part of pre-service inspection of TAPS 4, during January 17 and 18, 2005 and TAPS 3 during February 27 and 28, 2006. An indigenously developed RR CCTV camera, with radial viewing head was used, for this purpose.

The system was qualified on the mock-up channel, in the RC component shop of TAPS 3 & 4. In addition to qualifying the system on the mock up channel, the sensitivity of the system was checked, using a 0.8 mm diameter wire, with 10 x magnification. Image for this sensitivity calibration was recorded on a VHS cassette, which is shown below.

Inspection was carried out from fuelling machine vault, for the entire length of the channel, covering 3 '0' clock to 9 '0' clock circumferential position. Scanning was carried out along three circumferential positions viz: 4 'O' clock, 6 'O' clock and 8 'O' clock. Data from

Radiation resistant CCTV camera system



Scratch mark observed near rolled joint in R-15 coolant channel of TAPS 4

Calibration of CCTV camera using 0.8 mm diameter wire

Visual inspection being carried out in TAPS 4

A typical sag profile of coolant tube

Pressure tube sag measuring tool

Sag measurement being carried out in TAPS 4

each channel was recorded on separate VHS cassettes. No relevant crack like indications were observed, in any of the channels.

Sag measurement of coolant tubes of TAPS 3 & 4

Pre-service inspection of 20 coolant channels of TAPS-4 for sag measurement was carried out on January 17 and 18, 2005 and on February 27, 2006 for TAPS 3, using PTSMT (Pressure Tube Sag Measuring Tool) developed at the Division of Remote Handling & Robotics. PTSMT uses servo-inclinometer for fast and accurate sag measurement. Slope of pressure tube is measured, at a number of points along its length and sag is calculated using single integration. The measurements were made in the newly installed pressure tubes (in dry condition) from fuelling machine vault. The platform available for fuel loading was used, to access the channels to be inspected and the PTSMT was operated in manual mode.

AXIAL CREEP MEASUREMENT OF COOLANT CHANNELS FOR TAPS 3 & 4

Refuelling Technology Division

TAPS 3 & 4 are 540 MWe Pressurised Heavy Water Reactors. Each reactor has 392 coolant channels. These channels are subjected to axial as well as radial creep due to high temperature operation and radiation induced growth. It is required to monitor the axial creep of the coolant channels periodically. In line with previously developed remote creep measuring technique, creep measurement system for these reactors was developed at the Refuelling Technology Division, BARC. For future reference, base data reading for creep measurement was carried out. The technique enables to do creep measurement in just two-three shifts by using Fuelling Machines. A new technique also has been developed making use of non contact (ultrasonic) sensor. It enables to do creep measurement without touching the coolant channels. On experimental basis, in addition to TMAC technique, non-contact technique also was implemented in the TAPS 3 & 4. Performance was found to be satisfactory. On request from NPCIL, this technique was also used, to confirm the free thermal expansion of the channels in operating conditions.

Technique for Measuring Axial Creep (TMAC)

TMAC was developed for remote axial creep monitoring of the channels for 220 MWe. This technique is in use at all operating power stations of 220 MWe PHWRs. The same system cannot be used due to different numbers and different dimensions of the coolant channel, hence, the system was modified to suit 540 MWe PHWR and used at TAPS 3 & 4, for measurement of creep base data.

During the measurement of TMAC at TAPS 4, it was found that there is a systematic error of about 1.2 mm in 114 channel readings. A physical inspection of potentiometer and its coupling to Z advance cylinder was made, prior to the measurement of base data at TAPS 3. It was observed, that the nut connected to the hydraulic cylinder piston rod was loosely fitted, making a gap of around 2 – 3 mm. It was then recommended to provide check nut to prevent any looseness during operation. It is predicted that the same reason could be the cause of error of TMAC data, during measurement at TAPS 4.

Ultrasonic-Based Non-contact Technique

To measure creep data using TMAC the Fuelling Machines have to be aligned with each channel and both the machines should advance to touch the coolant channel to take the reading .This method takes around 3 shifts for completion of measurement including calibration and repeatability check.

To reduce the measurement time of the system and to make the system more user-friendly, along with enhanced graphical user interface and data management capabilities, a system based on non-contact type position sensor, has been developed. Sensor is carried to the reactor face by using a specially designed tool to be held by Fuelling Machines. Since the sensor used is non-contact type it need not be aligned with a particular channel, for taking the measurement. Instead, the channels of a particular row are scanned, while moving the machine in X direction, at a constant speed. Initially laser-based non-contact sensor was tried for measurement. It was observed that the laser-based sensor is highly sensitive to the reflection property of the

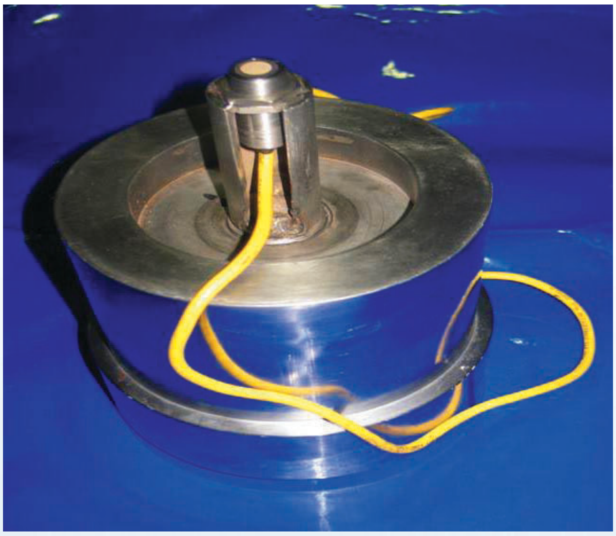


Fig. 1 : Ultrasonic sensor along with mounting tool used for creep measurement at TAPS 3

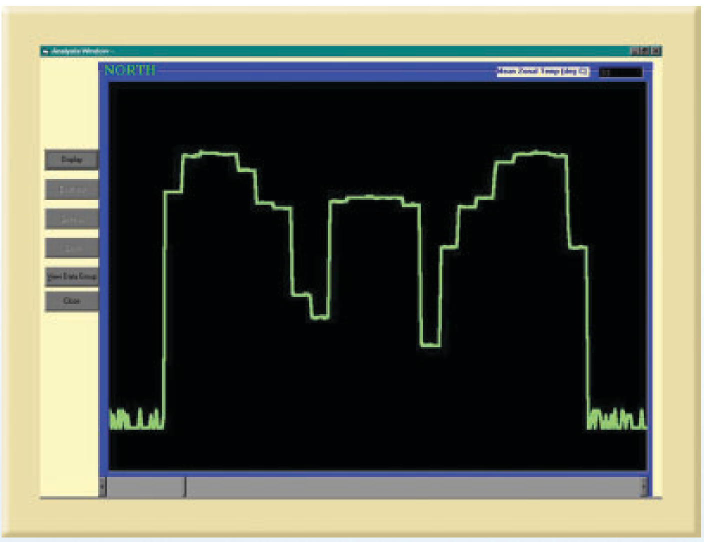


Fig. 2 : Data profile of A-14 (North side) channel taken during measurement at TAPS 3

target. This reflection property of the end fitting E-face (the measuring target) may vary from channel to channel and may change after a certain period. Hence it was not found suitable for this application. Alternately, a thermal compensated ultrasonic sensor was used, which is more independent of the target property.

System Description

Sensor

- Ultrasonic-based non-contact distance measurement sensor.
- In-built temperature compensation.
- Configurable range of the sensor.
- Measurement accuracy-0.3%.

Sensor Mounting Tool

Sensor mounting tool has been designed to take sensor at reactor face. This is compatible with fuelling machine. The tool is clamped in the fuelling machine after mounting the sensor. The tool has arrangement to adjust the gap between E-face of the channel. This ensures distance of the sensor from E-face within its operating range. The tool should be adjusted such that all the measuring channels should fall under the operating range of the sensor.

Data Acquisition System

A data acquisition card was developed and used for the acquisition of signal from the sensor. The card sends the signal in digital format, to the PC through a serial port.

Features of data acquisition card

- 8 single-ended analog input channels
- 2 single-ended analog output channels
- Input range : 0 to 2.5 V
- 12 bit resolution
- Sample Rate : 247 kSPS
- MCU : 8051 Based Core

Software

Software was developed using Visual Basic (ver 6.0) as front end and MS ACCESS (ver 7.0) as backend to calibrate, acquire, process and analyze the data taken during measurement. This software has enhanced data management capabilities and graphical user interface and is user-friendly.

Conclusion

Data measured by TMAC and non-contact creep measuring technique were compared and found to be matching satisfactorily for both units TAPS 3 & 4. As a check, they were also compared with physically measured values and found to be within accuracy range. Both the techniques could be successfully used for generating the base data. Time taken for measurement was insignificant and the technique was able to give print out in usable form. The technique was also useful, for measuring the thermal expansion of coolant channels in operating conditions to confirm the availability of intended gap, for thermal expansion of the coolant channel.

ANNOUNCEMENT

Forthcoming Conference

DAE-BRNS

Nuclear and Radiochemistry Symposium

NUCAR-2007

Feb. 14-17, 2007

This eighth biennial symposium is being organized in association with the Department of Chemistry, Maharaja Sayajirao University of Baroda, Vadodara. The scope of the symposium includes:

(a) Nuclear Chemistry and Nuclear probes
(b) Chemistry of actinides and reactor materials
(c) Spectroscopy of actinides
(d) Chemistry of fusion and activation products
(e) Radioanalytical Chemistry
(f) Radioisotopes
(g) Radioactivity in environment and
(h) Nuclear Instrumentation

A special half-day seminar on "40 years of $^{99m}T_C$ generator: its role in diagnostic imaging" is also being organized, as part of the symposium.

For further details please contact

Mr S. Jayakumar
Secretary, Symposium Organizing Committee,
NUCAR-2007
Radiochemistry Division,
Bhabha Atomic Research Centre
Mumbai-400 085, India
Tel.: 25594089 / 4088 / 0632
Fax: 91-22-2550 5151 / 2551 9613

Mr S.K. Pujari
Convener, Symposium Organizing Committee,
NUCAR-2007
Radiochemistry Division,
Bhabha Atomic Research Centre
Mumbai-400 085, India
Tel.: 2559 5326
Fax: 91-22-2550 5151 / 2551 9613
E-mail: pujari@barc.gov.in

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

एम.एल. शाह, ए.के. पुल्हानी, वास देव एवं बी.एम. सूरी द्वारा लिखित, दिसंबर 1-3, 2005 के दौरान डिपार्टमेंट ऑफ फिज़िक्स, कुमाऊँयुनिवर्सिटी, नैनीताल में आयोजित रीसेंट ट्रेंडस् इन फ्लोरिसेन्स स्पेक्ट्रोस्कोपी एन्ड इटस् एप्लिकेशन्स की राष्ट्रीय परिचर्चा में "मेजरमेंट ऑफ एटोमिक पेरामीटर्स ऑफ समेरियम यूजिंग टू कलर लेज़र इन्ड्यूस्ड फ्लोरिसेन्स" को सर्वश्रेष्ठ पोस्टर पुरस्कार दिया गया। श्री एम.एल.शाह, लेज़र एन्ड प्लाज़्मा प्रोद्यौगिकी प्रभाग ने इस शोध-पत्र को प्रस्तुत किया ।

A paper "Measurement of atomic parameters of samarium using two colour laser induced Fluorescence" (LIF) by M.L. Shah, A.K. Pulhani, Vas Dev and B.M. Suri, presented at the National Symposium on "Recent trends in Fluorescence Spectroscopy and its applications", held during December 1-3, 2005, at the Dept. of Physics, Kumaon University, Nainital, Received the Best Poster Award. Mr M.L. Shah of Laser & Plasma Technology Division, presented the paper.


M.L. Shah


A.K. Pulhani


Vas Dev


B.M. Suri


Chitra Seetharam

श्रीमती चित्रा सीथाराम, आणविक जैव प्रभाग को नवंबर 16-18, 2005 के दौरान भाभा परमाणु अनुसंधान केंद्र, मुंबई, में आयोजित भारतीय नाभिकीय सोसाइटी (INSAC-2005), की वार्षिक गोष्ठी में "फ ोसफे ट्स मीडियेटिड बयोरेमिडियेशन ऑफ कैडमियम" नामक शोध-पत्र के लिए सर्वश्रेष्ठ पोस्टर पुरस्कार से सम्मानित किया गया।

Ms Chitra Seetharam of Molecular Biology Division, was awarded the Best Poster Prize for her paper "Phosphatase mediated bioremediation of cadmium" at the 16th Annual Conference of the Indian Nuclear Society (INSAC-2005), held at BARC, Mumbai, from Nov. 16-18, 2005.


Swati Kota

श्रीमती स्वाती कोटा, आणविक जैव प्रभाग को दिसंबर 19-21, 2005 के दौरान भाभा परमाणु अनुसंधान केंद्र मुंबई, में मोलिक्यूलर बयोलोजी ऑफ स्ट्रस रेसपोंस एन्ड इट्स एप्लिकेशंस पर आयोजित LSS 2005 गोष्ठी में उनके शोधपत्र "पीपीआरए एन्. इम्पोर्टेंट प्रोटीन ऑफ रेडिएशन रसिस्टेंट इन ड्यूनोकोकस स्टिमुलेट्स कैटालेस एक्टिविटी इन एश्चरिचिया कोली" की प्रस्तुति पर सर्वश्रेष्ठ पोस्टर पुरस्कार दिया गया ।

Ms Swati Kota of Molecular Biology Division was awarded the Best Poster Prize for her paper "PprA: an important protein of radiation resistance in Deinococcus stimulates catalase activity in Escherichia coli –" at the DAE-BRNS Life Sciences Symposium 2005 (LSS-2005) on Molecular Biology of Stress Response and its applications, held at BARC, Mumbai, during Dec. 19-21, 2005.

G. Chakraborty

T.K. Basu

श्री जी. चक्रवर्ती, रिएक्टर संरक्षा प्रभाग एवं डॉ. टी.के.बासु, उच्च दाब भौतिकी प्रभाग, ने संयुक्त प्रयास से हिंदी में लिखित "नाभिकीय ऊ र्जा से विद्युत उत्पादन" नामक पुस्तक के लिए वर्ष 2003-2004 का मेघनाद साहा पुरस्कार प्राप्त किया। इन्होंने 25,000 रुपये की नकद राशि, एक स्मारिका एवं एक प्रशस्ति-पत्र भी प्राप्त किया। श्री चक्रवर्ती की दक्षता रियक्टर शक्ति संयंत्र संरक्षण, उष्मीय द्रव्यचालन विज्ञान एवं क्षणिक ऊर्जा संयंत्र विश्लेषण के क्षेत्र में है। इनके श्रेय में 35 से अधिक प्रकाशन हैं। डॉ. टी.के.बासु रियक्टर भैतिकी एवं न्यूट्रान भावावेग अनुकृति द्वारा उपसमीक्षात्मक प्रणाली (ADSS), पदार्थ विज्ञान में न्यूट्रान का प्रयोग आदि के विशेषज्ञ हैं। इनके श्रेय में राष्ट्रीय तथा अंतराष्ट्रीय पत्रिकाओं/परिचर्चाओं में 100 से अधिक प्रकाशन हैं।

Mr G. Chakraborty of Reactor Safety Division and Dr T.K. Basu of High Pressure Physics Division, have jointly received the Meghnad Saha Award for the year 2003-2004, for their book "Navikiya Urja Se Vidyut Utpaadan" (Electricity generation through nuclear energy) written in Hindi. They also received a cash award of Rs.25,000/-, a memento and a citation

Mr Chakraborty's areas of expertise include reactor power plant safety, thermal hydraulics and power plant transient analysis. He has more than 35 publications to his credit. Dr T.K. Basu is an expert in the fields of reactor physics, neutron transport simulation, accelerator driven sub-critical systems (ADSS), application of neutrons in material science etc. He has more than 100 publications in various national and international journals / symposia.

V.P. Venugopalan

डॉ.वी.पी.वेणुगोपालन, अध्यक्ष, जैव परिदूषण एवं जैव फिल्म प्रक्रियायें अनुभाग, भाभा परमाणु अनुसंधान केंद्र सुविधाएं, कल्पाकम को तीन साल की अवधि के लिए स्प्रिंगर, नेदरलेंडस् द्वारा प्रकाशित "एक्वटिक इकालाजी" नामक पत्रिका के संपादकीय मंडल का सदस्य नियुक्त किया गया है। डॉ.वेणुगोपालन एक मराइन् जीवविज्ञानी, ने एन आइ ओ, गौआ से पीएच.डी की उपाधि प्राप्त की है। समुद्री एवं ताजा जल का परिस्थिति विज्ञान, समुद्री जीव का शीत जल परिधि में क्षय इनके कार्यक्षेत्र में शामिल हैं।

Dr V.P. Venugopalan, Head, Biofouling and Biofilm Processes Section, BARC Facilities, Kalpakkam has been appointed as a member of the Editorial Board of the Journal "Aquatic Ecology" published by Springer, Netherlands, for a period of three years. Dr Venugopalan, a marine biologist, obtained his Ph.D. degree from NIO, Goa. His areas of work include marine and fresh water ecology, thermal ecology and marine bio deterioration in cooling water circuits.

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