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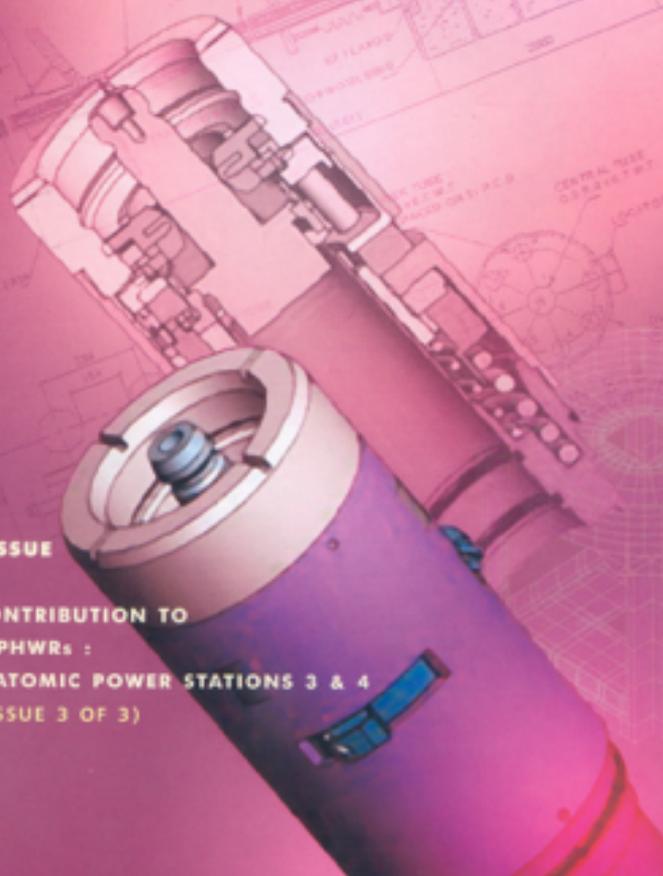
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न्यूज़लेटर

SEPTEMBER 2006

**BARC**  
NEWSLETTER

**IN THIS ISSUE**

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



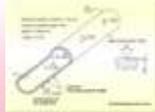
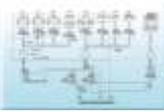
# भामा परमाणु अनुसंधान केंद्र Bhabha Atomic Research Centre



पंडित जवाहरलाल नेहरू द्वारा परमाणु ऊर्जा  
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गान्धी द्वारा दिनांक 12 जनवरी 1967 को भामा  
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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## TEST AND MONITORING SYSTEM 1 (TMS1) FOR SHUT DOWN SYSTEM 1 FOR TAPS 3 & 4

A. Lasitha, Amit Kumar, Manoj Kumar, M.K. Singh, Shruti Srivastava,  
R.M. Suresh Babu and U. Mahapatra

Control Instrumentation Division

TMS1 facilitates the following functions, which are important to the safety of the plant:

- Monitoring of Reactor Protection System 1 (RPS1) and SDS1
- Operator initiated online testing of trip circuits of RPS1
- Operator initiated online testing of clutches of shut off and control rods of SDS1, by de-energizing the clutch, for a predetermined time
- Performance evaluation of Shut Down System1, by measurement and analysis of the rod drop time and monitoring of RPS1, during reactor trip
- Generation of alarm and actuation of Liquid Poison Addition System, in case of failure of SDS1.

TMS1 tests the complete path of trip generation in one channel of RPS1 at a time, which includes PDCS-RPS1 Alarm Unit, Neutronic trip unit, Relay Module and 2/3 ladder contacts of the same channel. Problem with the sensors and field contacts (except process related analog signals), are detected by the spread/discordance checks of triplicated signals. Since TMS1 assures healthiness of RPS (RPS1) and Shut Down System1 (SDS1), which are safety systems, it has been categorized as a Class IB system (instrumentation system, class B).

Control Instrumentation Division, BARC designed and developed hardware and software for TMS1, using the Design Basis Report for TMS1, provided by NPCIL as input. Fabrication of the hardware and its testing was done at ECIL, Hyderabad.

TMS1 has been installed and commissioned in TAPS 3 & 4.

It has been successfully operating in TAPS 4 for more than a year.

### Principle of Operation

SDS 1 of TAPS 3 & 4, consists of 28 Shut off rods grouped in two banks, each consisting of 14 rods. The SDS1 instrumentation consists of the sub-systems: Reactor Protection System-1, Shut-Off Rod Instrumentation and Test and Monitoring System-1.

*Reactor Protection System-1* monitors the reactor trip parameter signals. When conditions calling for reactor trip are detected, the system generates command for actuation of SDS1. The trip parameter sensors/transmitters and associated circuitry are part of the respective process systems. The process (analog) trip parameter signals are monitored by Programmable Digital Comparator System for RPS1 and the neutronic trip parameters signals are monitored by Neutronic Trip Unit (NTU). Trip output generated by NTU and PDCS, and process parameters digital contacts from field are fed to a Relay module, which generates the channel trip by implementing necessary logic interlocks. The system follows triplicate channel philosophy with 2/3 voting logic for SDS1 actuation.

*The shut off rod instrumentation* incorporates electromagnetic clutch power supply and sensors for rod position monitoring. There are triplicate limit and reed switches mounted on the Shut-off rod drive mechanism assemblies for sensing fully OUT and 90% IN rod positions.

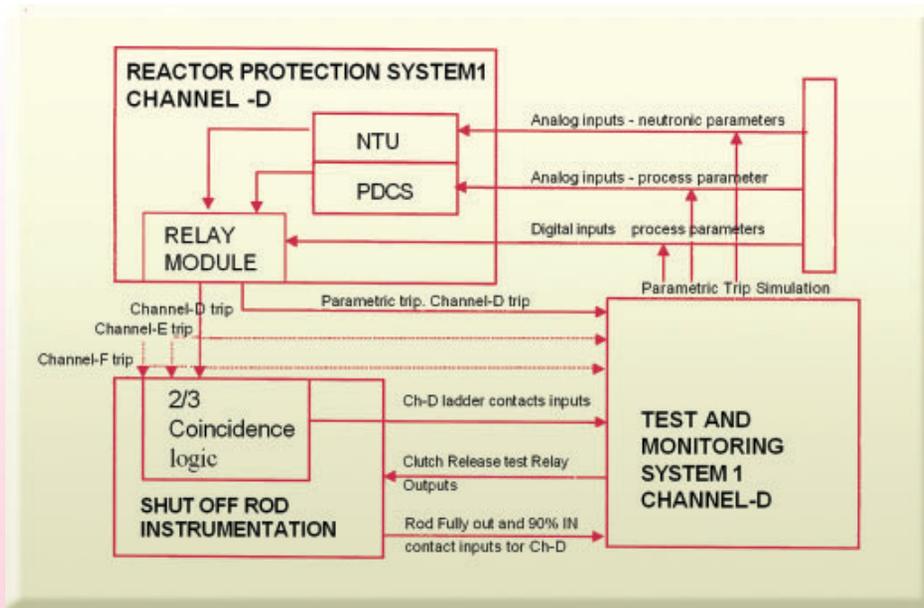


Fig. 1: Context of a single channel of TMS 1 in Shut Down System 1

Test and Monitoring System 1 (TMS1) is microprocessor-based system that facilitates complete monitoring and on-line testing of SDS1. Fig. 1 shows the context of one channel of TMS1 in SDS1. As RPS1 is a triplicated system, there is a corresponding channel of TMS1, for each channel of RPS1, to facilitate the testing of that channel. TMS1 forces the inputs to the RPS1, to simulate a trip condition and then it monitors the parametric trip and channel trip outputs from RPS1 and 2/3 ladder contacts of the corresponding channel.

Fig. 2 (a) shows the analog trip parameter test scheme. For analog trip parameter simulation, TMS1 energises corresponding test relays. The contacts of this relay are wired in such a way as to inject (contacts C3 and C4 close) the transmitter current,

for simulating trip on high setpoint or shunt (contacts C1 and C2 close) the transmitter current for simulating trip on low setpoint.

Fig. 2 (b) shows the logic switching test scheme. When TMS1 energises the test relay, C5 and C6 open to simulate the process trip parameter contact.

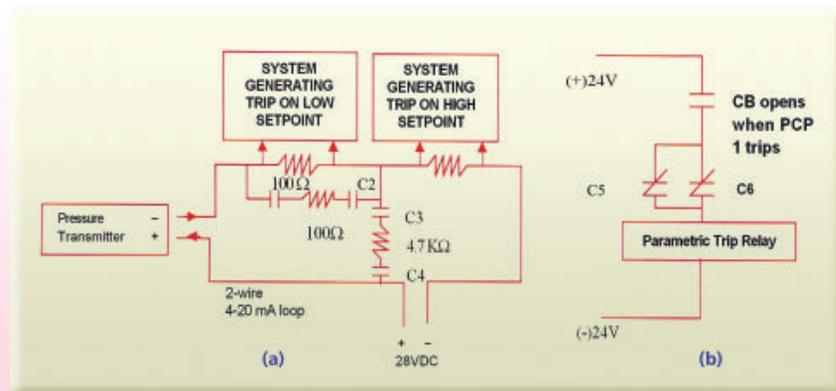


Fig. 2: Test scheme for (a) analog trip parameter (b) Logic switching trip parameter

TMS1 tests the electromagnetic clutches of 28 SRs and 4 CRs. On clutch test initiation for a selected rod, TMS1 energises the corresponding test relay for a specified period. Contact of this relay, is used to de-energise the corresponding clutch for the same period. This causes the rod to slip, while TMS1 monitors the Fully OUT contact. RRS is signalled about the clutch test in progress. After the clutch test is over, RRS drives the tested rod to fully OUT position.

### Description of the functions of TMS 1

TMS1 performs the following major functions :

#### Continuous monitoring of RPS 1 and SDS 1

TMS1 monitors inputs from RPS1 and SDS1 and displays their values continuously. It does spread check on its triplicate analog inputs and discordance check on all the digital inputs, in three channels and generates alarm, if the spread is high or discordance is detected. TMS 1 detects anomalies in SDS1 such as any rod on a bank slipping or 90% IN without corresponding SDS1 bank trip, any ladder contact open without channel trip and annunciates them.

#### Operator initiated online testing of trip circuits of RPS 1

TMS1 simulates all trip conditions of the plant. The operator can select a channel and the trip parameter to be tested, from the operator console. On command to perform trip test, permissive conditions, such as trip in any other channel, channel deselect, SDS1 bank 1 trip and SDS1 bank 2 trip are checked. If permissive conditions exist, corresponding test relays are energised, to simulate the trip condition. Simultaneously, parametric trip, channel trip, ladder contacts are monitored for a predetermined time. If any of these is sensed, the time elapsed is measured. The test is aborted, if any of the above mentioned permissive conditions disappear by de-energising all test relays. A test is successful, if all the

expected trip conditions corresponding to a trip parameter, are sensed in the predetermined time. Detailed test results are displayed on operator console and also logged.

#### Operator-initiated online testing of clutches of shut off and control rods of SDS 1

On-line test of clutches of Shutoff rods and control rods is being done for the first time being done in TAPS 3 & 4. From TMS1 operator console, operator selects the rod to be tested and gives command to test it. TMS1 checks permissive conditions such as all SRs are fully OUT prior to initiation of a test. TMS1 generates appropriate test output and monitor response of SDS1 on its fully OUT inputs, for a predetermined duration. If slipping of the rod being tested is sensed, within the duration in at least two channels of CPU, it measures the time elapsed and the test is declared successful. Detailed results are displayed on operator console and also logged.

#### Performance evaluation of SDS 1

Upon a reactor trip, TMS1 measures the drop time of each of the 28 Shut Off rods. Median value of drop times, measured in three channels for each rod, is checked for high as well as low values. High drop time value indicates some problem with the mechanism. Low drop time indicates that the Shutoff rod may not be at the top, at the time of reactor trip. TMS1 annunciates an alarm, if any of the rod drop time is outside acceptable band. If more than 2 rods are not fully in within 30 sec, an alarm is generated and the Liquid Position Addition System is actuated.

The Drop times measured in individual channel and their median and alarm status, are displayed on operator console and also logged.

#### Monitoring of trip parameters on Reactor trip event

TMS1 monitors parametric trip, channel trip and channel

reset during a set period (default 500ms), after a reactor trip is detected. It records the parametric trips that occurred before the tripping of 2/3 channels tripped and measures the time between parametric trip and channel trip, occurring after the reactor trip. This information is logged and displayed on operator request. This information is helpful to the operator, to detect a forced trip i.e channel trip initiated by other two channels trip rather than a parametric trip in the same channel. Forced trip is a condition, where two channels have tripped on a parameter, but the third channel has not tripped and is forced to trip by an external circuit.

### Information Logging

TMS1 logs the following in both the MPUs with time stamps with respect to the corresponding MPU's system time :

- Normal messages such as, configuration change messages, logging on and logging off from control access mode, etc
- Alarm messages such as failed tests, SDS1 malfunctions, SDS1 failures, system fault in any TMS1 nodes, networks faults, SDS1 anomalies, Analog input spread high, digital input discordance etc and the alarm clear messages
- Detailed test results
- Rod drop times
- Parametric trip monitoring report.

TMS1 automatically archive the logs, after they reach a predetermined size. It allows the operator to take the backup of logs, which have been archived. Operator can take a hard or soft copy of any type of log with filter such as logs between a start and end date, logs of test for a specified trip parameter, test summary log only, test details etc.

### System Description

As seen from Fig. 1, TMS consists of three Channel Processor Units (CPU-D, CPU-E and CPU-F) – one for

each channel of the RPS and two Main Processor Units (MPU1 and MPU2). The CPUs are located in the channel rooms. MPU1 is in the Main Control Room and MPU2 in the Computer Room. The five TMS nodes are networked through a dual Ethernet network.

The functions of TMS1 are distributed across these nodes. MPU1 and MPU2 are redundant identical units, which are operator consoles of TMS1. MPU provides GUI for the following:

- Test initiation with control access
- Display of TMS1 inputs/outputs, test results, alarm messages, etc
- Discordance check on inputs to three channels of TMS1
- Logging /printing of test results, alarm messages, etc.
- Sending information to a Centralized Operator Information System (COIS)
- Alarm annunciation for failure conditions

CPU implements the following functions:

- Continuous monitoring of inputs from RPS1 and SDS1
- Testing of RPS1 and SDS1 on request
- Drop time measurement, M\_ALPAS actuation, trip parameters monitoring on reactor trip.

### Main Processor Unit

The two MPUs are identical PC- based systems, with the following configuration:

Industrial PC (Pentium 4) with monitor, keyboard, mouse, one digital input card, one digital output card and three Ethernet ports.

Through the MPUs, the operator sends commands and gets responses. Only one MPU is enabled at a time. MPUs also log information and pass summary information to the plant wide Computerized Operator Information System (COIS).

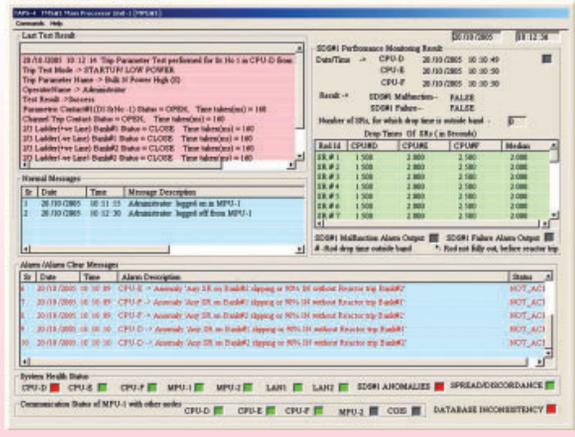


Fig. 3: TMS1 Main Screen



Fig. 4: TMS1 Operator Console

MPU provides a GUI for passing of commands and display of information. Windows 2000 has been used as a development and target platform, for MPU software.

Fig. 3 shows the default screen of TMS1. Fig. 4 shows

the TMS1 console in Main Control Room of TAPS 4.

**Access modes :** There are two types of commands in TMS1 – normal and privileged. Normal commands are accessible to all. These include display of signal values, trip test results log, clutch test results log, SDS1 performance monitoring results log, alarm messages, etc.

Privileged commands can be accessed only with mechanical pass-key and password. The pass-key also ensures that privileged commands can be passed only from one MPU at a time.

**Commands :** Various kinds of commands, that can be issued from MPU are as follows :

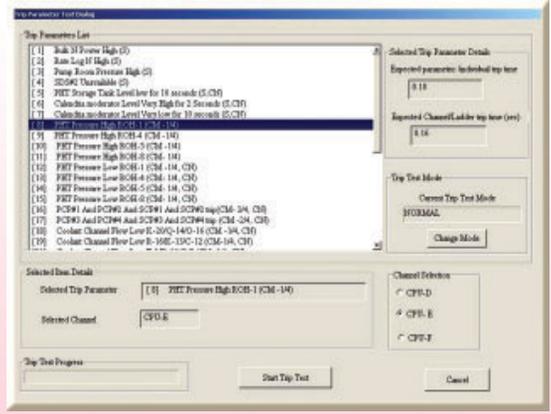


Fig. 5: Trip test screen

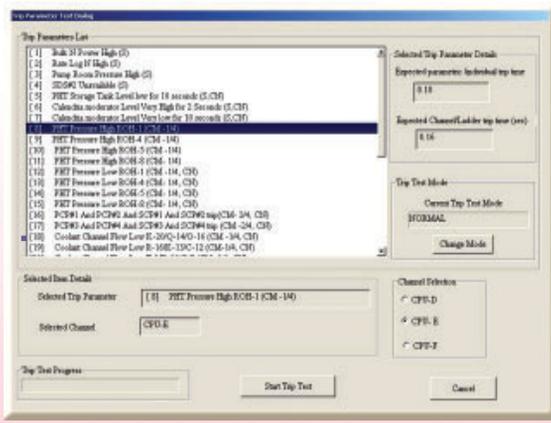


Fig. 6: Clutch test screen

### Normal Commands

- Display of TMS1 inputs/ outputs, channel health status, configurable parameters, test results, SDS1 performance evaluation results, alarm/normal messages etc.

### Privileged Commands

- Online Test initiation (Figs. 5 and 6)
- Configuration change

### Logging of information in MPU

MPU logs relevant data : alarms and anomalies detected during continuous monitoring, data collected at reactor trip events, results of operator initiated tests, etc.

### Information to COIS

MPU sends information such as test results, SR drop times on reactor trip, etc. to a centralized information logging system, COIS through an Ethernet link.



Fig. 7: Channel Processor Unit

### Channel Processor Unit

The CPUs are embedded systems: VME backplane, 733 MHz Pentium III processor, dual Ethernet controllers and necessary I/O boards. Each CPU has approximately 200 I/Os. Fig. 7 shows one channel of CPU.

TMS 1 performance requirements, specify a timing measurement accuracy of 20ms. Hence, CPU software is required in executing in a real-time, multitasking environment. QNX, a hard RTOS was chosen, based on earlier experience in CnID on QNX and the availability of board support packages, for the SBC hardware with the vendor.

Each channel of CPU receives inputs from respective channels of RPS1 and SDS1 and generates test outputs to them. All the logics for trip parameter test, clutch release test, drop time measurement, M\_LPAS alarm generation and trip parameter monitoring are performed in CPU.

- During *trip parameter test*, the CPU energises the corresponding test relay, for a predetermined time and monitors the expected trip inputs.
- During *clutch release test*, the CPU energises the test relay output to de-energise the clutch for a predetermined time, so that, the rod slips from it's fully OUT position.
- During a *reactor trip*, the CPU measures the time elapsed from sensing of reactor trip to each rod reaching the 90% in position. It records the parametric trip and channel trip events at reactor trip and monitors the same immediately after reactor trip, for a predetermined time and measures the time elapsed.

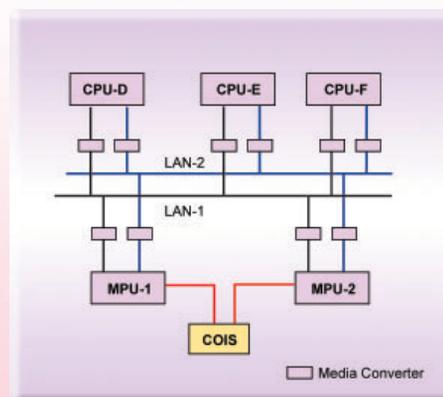


Fig. 8: Physical interface between TMS1 Nodes

The results of these operations are sent to both MPUs for display or for further analysis. The response time of CPU is 20 milliseconds and accuracy of measurements is 20 milliseconds.

### Dual Ethernet LAN

The physical interface between all sub-systems of TMS1 is via dual-redundant fiber optic Ethernet LAN. Ethernet implementation follows IEEE Std 802.3-1998 10BaseT specifications.

Fig. 8 shows the interface between all the TMS nodes on the network. TMS1 network is configured as Hirschmann's HIPER-Ring (Hirschmann Industrial Performance Redundant Ethernet Ring). This configuration increases the availability of the network.

Each TMS node has 2 Ethernet ports. Each Ethernet port is connected to an industrial hub, (RH1-TP/FL) via twisted pair cable. The hub converts the signal from electrical domain to optical domain and vice versa.

The specification is 10 BASE T and 10 BASE FL on the twisted pair side and on the fiber-optic side respectively.

### System Self Diagnostics

Self-Diagnostic is required, to bring in operator attention, in case of a fault or failure. Every node performs self-diagnostics on its hardware and software to the extent supported by hardware.

In TMS1, the following diagnostic checks are implemented:

- Check absence of any I/O board
- Finite impulse testing of Digital Inputs Board
- Read back check on Digital output board
- Analog Input Board test
- Irrationality check on analog inputs
- Configurable data integrity check
- Watchdog timer (WDT) test.
- Data inconsistency check
- Network Diagnostics

The MPU displays alarm message on every new fault detected in any MPU or any CPU. It also displays the health details of all nodes on operator request (Fig. 9).

### Operator Configurable parameters

TMS1 allows operator to configure timing parameters such as timeout period of trip parameter test, clutch de-energisation time, acceptable band limit for drop time, time out for generation of M\_LPAS alarm etc. and type of contact of digital input. There are possibilities of certain modifications in process system design, after some experience with the plant, which may lead to change in timings. TMS1 allows operator to change certain values on-line, under controlled access mode. This feature helped in avoiding the need, to re-build the software for TMS1, under such changed requirements.

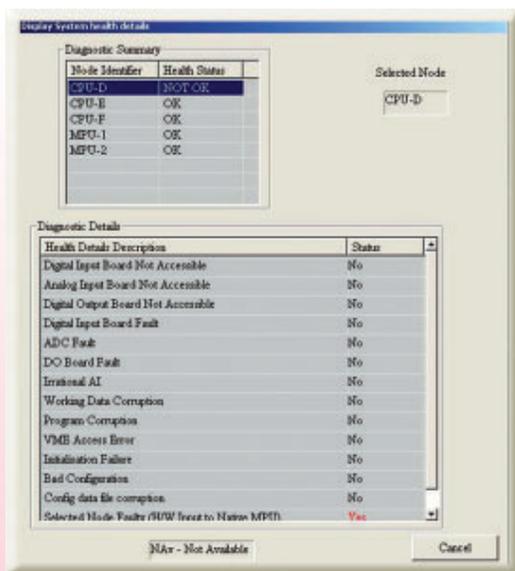


Fig. 9: System health details

## ANALYSIS OF GADOLINIUM BY UV-VISIBLE SPECTROPHOTOMETRIC METHOD AND ITS COMPARISON WITH ICP-AES METHOD

S.C. Dash

Tarapur Atomic Power Project 3 & 4, Tarapur

R.S. Sharma and S. Dutta

Research Reactor Services Division, BARC, Mumbai  
and

G. Venkateswaran

Water and Steam Chemistry Division, BARC, Mumbai

The 540 MWe Pressurized Heavy Water Reactor at TAPS 4, uses both Gadolinium (Gd) and Boron (B) in the moderator system, for reactivity control (specifically for use as a diverse shutdown system). About 30 ppb of Gd in the moderator, corresponds to a negative reactivity insertion of 1 mk, while for the same input of negative reactivity, about 120 ppb of B is required. Thus, use of Gadolinium helps in reducing the concentration of neutron poison in the moderator. However, based on the same concentration – reactivity relationship, the use of B helps in fine adjustment of reactivity (small reactivity changes for larger removal of Boron) towards approach to criticality. Thus, it was found advantageous, to use both B (anhydrous  $B_2O_3$ ) and Gd (as  $Gd(NO)_3 \cdot 6H_2O$ ) neutron poison in the moderator, before attaining the first criticality of TAPS 4.

According to reactor physics requirements, analyzing Gd reliably in the moderator, to a minimum detectable level of ~30 ppb, becomes desirable, to keep track of reactivity changes, when Gd removal is done, during approach to criticality. In this connection, two methods were looked at : namely 1) a UV-visible spectrophotometric method, which can be easily accomplished at site and 2) a more sophisticated Inductively Coupled Plasma Atomic Emission Spectroscopy (ICP-AES) method. Detailed calibration checks and independent analytical verification of Gd levels in synthetic samples was done, before choosing

the UV-visible spectrophotometric method. The effect of the presence of B along with Gd, on the chosen UV-visible analytical method, was evaluated.

### Analysis of Gd and B in the moderator system

Gd was estimated by UV-visible spectrophotometric method, based on the bathochromic shift, caused by the formation of a complex of Gd with Arsenazo(III) [(2,7-bis (2-arsenophenylazo)-1,8-dihydroxy-naphthalene-3,6-disulphonic acid) reagent and measuring the absorption at 652 nm. Though this colorimetric method is well known, there is a lack of published literature on its practical applications. Hence, a detailed investigation was undertaken to carry out elaborate calibration checks, using this method. The checks showed, that this method could easily fulfill the requirements of analyzing Gd, to a sensitivity of 30 ppb i.e., about 1 mk addition of reactivity, during the Gd removal operation. In fact, with a 1 cm path length cell, one can estimate to an accuracy of 25 ppb and with the use of 5 cm path length cell, estimation to a level of 5 ppb is also possible. A crosscheck of this method with the Inductively Coupled Plasma-Atomic Emission Spectroscopy (ICP-AES) method of Gd analysis was carried out and it was proved that, the UV-visible spectrophotometry method has a lower relative standard deviation, than the ICP-AES method.

Boron was estimated by the pH metric titration, after complexation with D-sorbitol.

In both the estimations of Gd and B, suitable blank corrections involving blank heavy water, were carried out.

In the spectrophotometric method, when analysis of heavy water sample is done, it is important to take the same volume of high purity heavy water, for blank solution preparation. Otherwise, if blank is prepared fully in light water, then, a reduced absorption results.

## Results and Discussion

### Minimum detectable levels of Gd by UV-Visible Spectrophotometry

Initially, experiments were done with 2 ml sample volume, in which 7 ml of 1:1 ethanol:1,4-dioxane mixture was added. This was followed by the addition of 0.4 ml of 0.1M HNO<sub>3</sub>, 0.4 ml of 0.025% arsenazo (III) and then making upto a full volume of 10 ml. With a 1 cm cell, we get a concentration (ppb Gd) / absorption slope of 1016 ± 46. (Fig.1). Here, the concentration of Gd refers to that in the full volume of 10 ml. At concentrations of 20-50 ppb Gd, a relative standard deviation (R.S.D.) of ~4.5 % was observed. However, at a concentration of 10 ppb Gd (in the full volume of 10 ml), an absorption of 0.01 with 5.6 % R.S.D. was observed. Since the volume taken is 2 ml in a full volume of 10 ml, the actual Gd concentration in the samples, amounted to 50-250 ppb. Since the minimum stable absorption readable is 0.005 with the UV-visible spectrophotometer available at site, Gd concentration at 25 ppb in the sample, could be analyzed.

Due to the carcinogenic nature of dioxane and the associated disposal problem of such solutions, the addition of this chemical was eliminated and hence, a sample additional volume of 8 ml became available. But elimination of dioxane, diminished the absorption obtained by a factor of 2, for the same concentration of Gd. Thus, by taking 8 ml, especially when the sample

Gd concentrations were less than 100 ppb, the absorption increased by a factor of 2. Therefore, at a minimum readable absorption of 0.005, estimation of Gd concentration at ~15 ppb became possible.

With the introduction of 5 cm cell for absorption measurements, even Gd at ~5 ppb level, could be

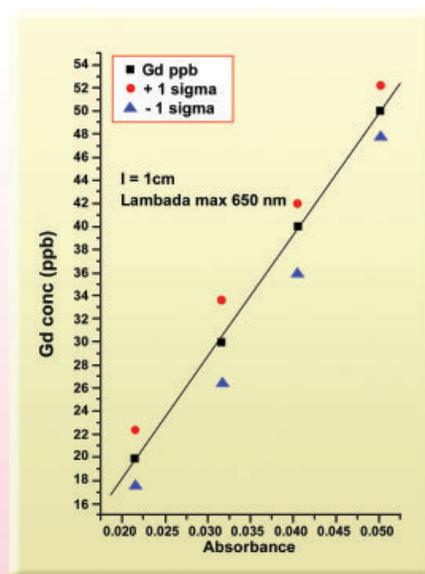


Fig. 1 : Absorption versus Gd concentration in 10 ml full volume, containing 2 ml sample volume

analyzed.

### Repetition of Gd analysis and influence of the presence of Boron

The repetition tests for Gd analysis for various standards, are shown in Table 1. At [B] / [Gd] concentration ratio of upto 10, it was observed, that Boron did not interfere in the Gd analysis and the practically encountered concentration ratios of B to Gd would only be, of this magnitude.

### Comparative account of Gd analysis by UV-Visible Spectrophotometric method and ICP-AES method

Working standard solutions of Gd were prepared, from ICP/DCP stock standard solutions of 10005 ppm Gd, in light water. Using this stock solution, a dilute standard

Table 1 : Gd-analysis repetition test in low range

S.No.	Gd. std. make	Gd concentration (ppb) in 10 ml full volume	Gd sample cone (ppb)	Absorption repeat values	Average absorption	Std. Deviation	% Relative Std. Deviation
1	ICP/DCP	5	25	0.0037	0.00395	0.0005	12.6
2		5	25	0.004			
3		5	25	0.0033			
4	BARC	5	25	0.0043			
5		5	25	0.0049			
6		5	25	0.0035			
7	Aldrich (99.9%)	5	25	0.0043			
8		5	25	0.0039			
9		5	25	0.0037			
10	ICP/DCP	10	50	0.0082	0.00832	0.0011	13.2
11		10	50	0.0086			
12		10	50	0.0095			
13	BARC	10	50	0.0106			
14		10	50	0.0069			
15		10	50	0.0077			
16	Aldrich (99.9%)	10	50	0.0072			
17		10	50	0.0077			
18		10	50	0.0085			

Table 2 : Comparison of Gd analysis by UV-Visible Spectrophotometry & ICP/AES methods

Serial No. / Sample No.	Gd concentration in the standards taken for analysis (ppb)	Gd concentration obtained by UV -Visible Spectrophotometric method (ppb)		Percentage deviation		Gd concentration obtained by ICP-AES method (ppb)
		By taking 4ml sample for analysis	By taking 8.5ml sample for analysis	By taking 4ml sample for analysis	By taking 8.5ml sample for analysis	
1/8	40.02	33.11	36.0	-17.3	-10	35.9±6
2/9	80.04	77.75	73.1	-2.9	-8.7	73.2±0.4
3/10	120.06	116.8	109	-2.7	-9.2	112±0.4
4/11	160.08	159.6	-	-0.3	-	-
5/12	200.10	203	-	+1.4	-	-

of 100.5 ppm Gd and from this solution, a more dilute standard of 4.002 ppm Gd was prepared. Light water was used for all dilutions. Working standards of 40.02, 80.04 and 120.06 ppb Gd were prepared and given for analysis. While the UV-visible spectrophotometric analysis was done by TAPS 3 & 4 chemical laboratory, the analysis by ICP-AES method was done, by the BARC Facility at Tarapur. Table 2 shows the results obtained.

It is seen from the results given in Table 2, that there is good agreement between the analysis results obtained by employing both the methods. At 40 ppb level, the analysis accuracy can be  $\pm 10\%$  in the UV-visible spectrophotometric method, while that obtained in the ICP-AES method, it is  $\pm 16\%$ . The UV-visible spectrophotometric method can be done in house in the chemical laboratory of TAPS 3 & 4. The concentration level of analysis and the accuracy obtained by this method, can meet the reactor physics requirements. While all the moderator samples can be analysed by this method, a few selected samples can be cross-checked, by the ICP-AES method.

### Conclusion

A simple UV-visible spectrophotometric method, for the determination of concentration levels of Gd, in 30 ppb range in the moderator water, was demonstrated. This was to meet reactor physics requirements, in terms of reactivity addition, when this neutron poison is removed from the moderator. That such a method is indeed more sensitive than these requirements, was also shown. In fact, the method is slightly better than the ICP-AES method, which involves much more costlier equipment, than a UV-visible spectrophotometer. The method so finalised, was successfully used in the estimation of Gd, during the first approach to criticality of TAPS 4 and the same method would be adopted, during the first criticality of TAPS 3 too.

### ANNOUNCEMENT

#### FORTHCOMING SEMINAR

#### NATIONAL SEMINAR ON PHYSICS AND TECHNOLOGY OF SENSORS (NSPTS-12)

The Technical Physics and Prototype Engineering Division, BARC, is organising a three-day seminar, during 7-9 March 2007 and it will be held at Anushaktinagar, Mumbai. The seminar is being sponsored by the University of Pune, BRNS, ISRO, Ministry of Information Technology, DST and CSIR. The seminar programme comprises invited talks (by scientists from India and abroad) and contributed papers (in the form of oral and poster presentations). Scientists are also welcome to exhibit sensors developed by them. Awards have been instituted for the best "continued sensor work". Seminar discussions would be on the following topics:

1. Sensors and actuators; fundamentals and physics
2. Magnetic, Optical, Thermal sensors.
3. Automotive sensors, Biosensors
4. Gas and Pollution related sensors
5. Electromechanic sensors
6. Nanomaterials for sensors
7. Sensor fabrication techniques
8. Intelligent/Smart sensor systems
9. Sensor materials and their processing
10. Sensor systems, actuators and applications
11. Micromachined and integrated sensors
12. Polymer and bio-sensors
13. Product and prototype presentation
14. Sensors for requirements of DAE and
15. Role of sensors for disaster management.

An abstract of less than 200 words and also subsequently the manuscript should be sent to the Secretary, NSPTS-12: [nspts\\_12@barc.gov.in](mailto:nspts_12@barc.gov.in)

For further details please contact :

**Dr S.C. Gadkari**

Local Convenor, NSPTS-12

TPPED, BARC,

Mumbai - 400 085, India

E-mail : [nspts\\_12@barc.gov.in](mailto:nspts_12@barc.gov.in)

Home page: <http://www.barc.gov.in>

## DEVELOPMENT OF BEARING AND SEALING MATERIAL FOR RAMs OF 540 MWe FUELLING MACHINE

Refuelling Technology Division

During the testing of Ram Assembly of 540 MWe PHWR at RTD, BARC, high torque was observed while moving the B-Ram. Front housing was opened and B-Ram labyrinth was removed. It was seen that there are lot of wear marks and wear debris on the body of the B-Ram labyrinth piston and ID of the Rear Housing [Figs. 1(a) & (b)]. Rear housing is made of SS 403 while the labyrinth is made out of Aluminum-Bronze material. Tribological tests were carried out and it was found out that material compatibility issue at higher contact stress is causing this severe abrasive wear.

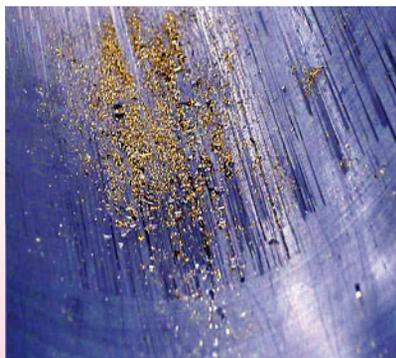


Fig. 1(a): Wear marks and wear debris at rear housing



Fig. 1(b): Wear marks on B-RAM labyrinth

Various materials were studied as a candidate for replacement of the Aluminum Bronze, which included Fibre-Reinforced-Polymer-Composite (FRPC). Selected FRPC material is made out of Polyester resin as matrix material and polyester fine-mesh as reinforcement. This material is extensively used in oil and water hydraulic cylinders and has excellent tribological properties. Literature survey shows that this material can withstand ionising radiation upto 100 MRad and it is readily available. 15 mm wide and 2.5 mm thick strips of this material were used for testing.

### Testing on RAM Assembly

- Initially one wear pad of was installed in the adapter, made to replace B-Ram labyrinth [Fig. 2]. The ram assembly was operated for 600 Cycles (equivalent to 50 Channel operations). Substantial wear (of the order of 0.16 to 0.2 mm) was observed due to scoring marks and rough surfaces of the rear housing. The inner surface of the rear housing was then polished and all the burrs were removed. After completing 1200 more cycles (total 1800 Cycles -150 Channel equivalent) wear pad was again removed and thickness was measured. Maximum total wear of the wear pad was noticed at, from 4 O'clock to 8 O'clock position and was of the order of 0.28 to 0.34 mm.
- No. of rings were increased to 2 and to study the effect of radiation, one of the rings was subjected to 10 MRad gamma radiation in ISOMED.
- Wear was measured after 1200 cycles (100 Channel equivalent) and the maximum wear was measured 0.08-0.10 mm around 6 O'clock position.

- d. Allowing wear of 0.5 to 0.6 mm of the projected pad height of 1.5mm, it can be concluded that, with two wear pads of 25 mm x 2.5 mm, 500 Channel equivalent operations can be carried out.

### Testing at Tribology Lab, RTD

- e. Reciprocating sliding wear test were carried out in the lab on un-irradiated and irradiated samples of (1 MR, 5 MR and 10 MR).
- f. There was no significant wear of the composites and counter body was noticed for a sliding distance of 635 m, equals to 50 Channel operations under simulated loading and underwater condition.
- g. There is no significant change in coefficient of friction due to various radiation doses.
- h. This material was further irradiated upto 100 MR and no significant change in tribological properties was recorded.

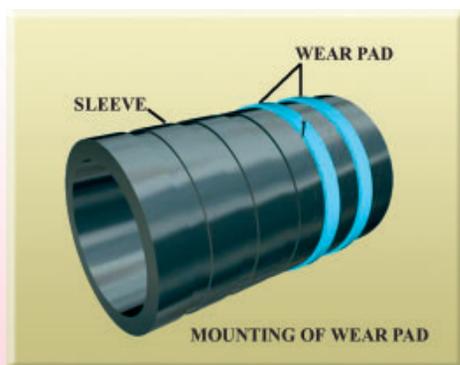


Fig. 2: Adapter sleeve and Mounting of Rings of Wear Pad

### Conclusion

Fibre-Reinforce-Polymer-Composite can be a very useful bearing and sealing material in the sliding contact, provided, the contact surface is smooth. Labyrinth piston was redesigned and two wear pads of 25 mm x 2.5 mm used in B-RAM, can be operated safely, for 500 Channel fuelling operation and can be replaced easily.

### ANNOUNCEMENT

#### FORTHCOMING SYMPOSIUM ON ADVANCES IN STAINLESS STEELS 2007 (ISAS - 2007)

The Indian Institute of Metals, Kalpakkam Chapter, the Metal Sciences & ferrous Divisions of IIM, the BRNS and IGCAR, Kalpakkam have jointly organised a three-day symposium from April 9-11, 2007, at Chennai, India.

The symposium schedule comprises both plenary and parallel sessions with key note addresses by invited guest speakers. Contributory papers are invited on the following topics. Abstracts of not more than 250 words are to be submitted. Exhibition on the theme of the symposium is also being organised.

Topics to be covered:

- Austenitic, ferritic, martensitic, precipitation hardening and duplex stainless steels
- New and special grades of stainless steels
- Production of stainless steels (Casting and Wrought Products)
- Powder metallurgy of stainless steels and products
- Production of stainless steel welding consumables
- Physical metallurgy of stainless steels and welds
- Mechanical properties of wrought products, casting and weldments of stainless steels
- Role of trace and tramp elements on the properties of stainless steels
- Design aspects, codes and standards for stainless steel components
- Modeling and Structural integrity assessment of stainless steel components
- Welding processes and weldability of stainless steels
- Corrosion aspects of stainless steels
- Non-Destructive Testing of castings, wrought products and welded components
- In-service inspection of stainless steel components
- Residual stresses and distortion in fabrication of stainless steel components
- Heat treatment of stainless steels
- Surface engineering of stainless steel components, tribology and wear
- Failure analysis of stainless steel components
- Global marketing

#### Important dates:

Abstract submission	: 15 <sup>th</sup> Dec., 2006
Intimation of acceptance	: 1 <sup>st</sup> Jan., 2006
Submission of paper	: 1 <sup>st</sup> Feb., 2007
Early bird Registration	: 15 <sup>th</sup> Feb., 2007
Deadline for registration	: 15 <sup>th</sup> Mar., 2007

For further details and correspondence one may contact:

**Dr K. Bhanu Sankara Rao**

Convenor, ISAS 2007  
Chairman, IIM Kalpakkam Chapter &  
Head, Mechanical Metallurgy Divn, IGCAR,  
Kalpakkam- 603 102, Tamil Nadu, India.  
Tel: +91 - 44 - 27480107 (0) / 27481044 (R)  
Fax: +91 - 44 - 27480060/27480075  
E-mail: bhanu@igcar.gov.in.isas@igcar.gov.in

## PRE-SERVICE INSPECTION OF COOLANT CHANNEL ROLLED JOINTS AND SURROUNDING AREAS IN 540 MWe TAPS 3 & 4 REACTORS

J. L. Singh and D.N. Sah  
Post Irradiation Examination Division, BARC  
and  
R. K. Mishra and N. P. Srivastava  
NPCIL, TAPS 3 & 4, Tarapur

Structural integrity of coolant channels of Pressurized Heavy Water Reactors, is vital for the safe operation of these reactors. The pressure tubes carrying fuel bundles and hot coolant is supported at the rolled joints. The leak-tight rolled joints are made, between stainless steel end fitting with three circumferential grooves, to which Zr-2.5%Nb coolant tube is roll expanded and the two different metals are mechanically bonded. During the process of rolling, the tube expander exerts a very high force on the tube, so that, the tube metal is plastically deformed and flows into the grooves. The early generation rolled joints with higher clearance, were characterized by high tensile residual stress in the adjacent region of pressure tube, where Delayed Hydride Cracking (DHC) occurred in Canadian reactors. DHC problem was solved by reducing residual stresses incorporating zero clearance rolled joints in subsequent reactors. To prevent formation of incipient hydride cracks, it is essential that rolled joint area should be free of flaws.

### Probe Holder

Two inspection heads housing two axial and two circumferential line focused ultrasonic probes were fabricated. Holes at 27° were

drilled in axial and circumferential orientations in an annular Perspex cylindrical rod of 4" diameter. Immersion line focused probes of 10 MHz frequency, were fitted in the drilled holes to generate 45° mode converted shear waves, in the wall of pressure tube.

### Calibration Standard

Sample rolled joint was cut open and freed from end fitting for making 2%(0.086mm) depth artificial notches. All the notches were of 6 mm length and 60° included angle. The calibration notches were made by EDM both

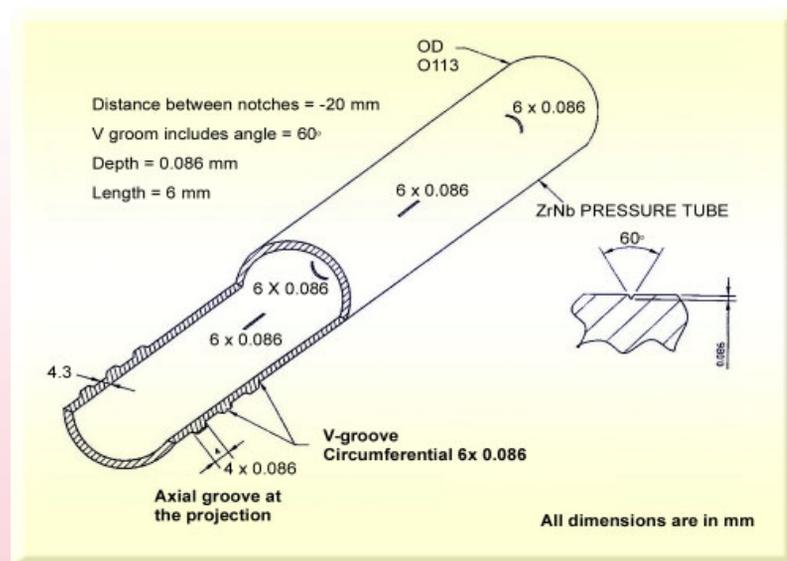


Fig.1: Section of rolled joint and surrounding area with calibration notches

in axial and circumferential directions on OD and ID as well as in adjacent and upset region. Immersion ultrasonic procedure was developed in our PIED Laboratory. In Figs.2a and b, small signals appearing at the left, are due to ID surface roughness. In Fig. 2a, the tallest signal at half V path is from OD calibration defect. In Fig.2b, the tallest signal at one V path is from ID calibration defect.

### Mock-Up Trials

At reactor site, one of the mock up coolant channel assembly was filled with water. End fitting face was sealed using a special gadget developed by the site engineers. A constant water feed was maintained during scanning to make up for water loss. The 12 'O' clock region could not be scanned due to water bubbles.

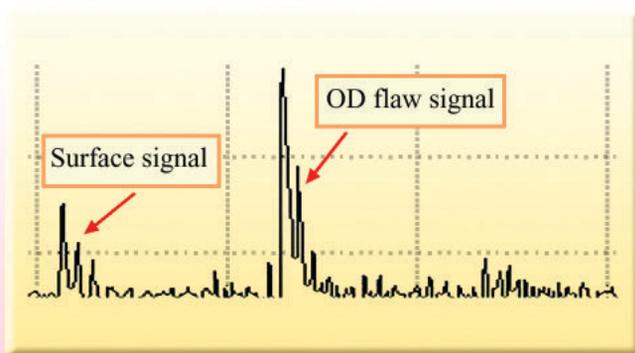


Fig. 2a: Signal from OD cal. std. defect

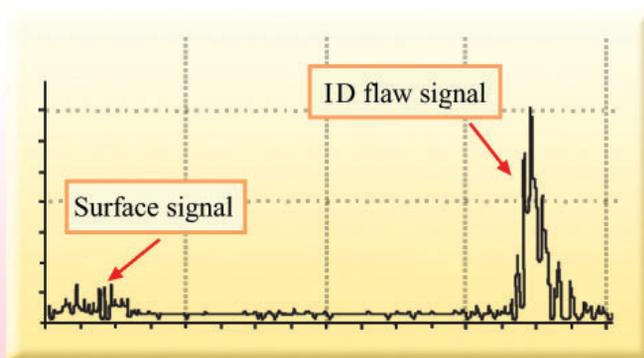


Fig.2b: Signal from ID cal. std. defect

### In-Core Scanning

The channels were first cleaned using water and nylon brush thoroughly. Two EEC-make ultrasonic flaw detectors and scanner were stationed on the hanging platform. Calibration was rechecked before inserting into the channel. Immersion ultrasonic testing was carried out in-situ, to ascertain the condition of rolled joint and entire pressure tube length from 11 to 1 'O' clock. Axial and circumferential scanning was performed in 20 selected coolant channels in each reactor. To hold water inside the pressure tube, end fitting face seal assembly was developed by site engineers. Two axial and two circumferential ultrasonic probes were used for scanning. Scanning was carried out on twenty coolant channels in each of TAPS 3&4 reactors, to collect pre-service base line data.

### Conclusion

No reportable flaw in the region adjacent to the rolled joint was observed in any channel of any of the two reactors. The coolant tube 11 to 1 'O' clock and between the two rolled joint was also free from any discernible defect. The top 12 'O' clock axial portion could not be scanned due to air bubbles. The groove area posed problem for axial probe, but circumferential probe did not find any axial flaw.

### Scanner

Ten meter long condenser tube was used to support the Perspex probe head and cables coming out through it. To avoid spillage and loss of water during in and out movement, double O-ring sealing was provided. The ends and rolled joint region, were identified by signals, picked up by axial probe.

## DESIGN OF FLUID POWER CIRCUIT FOR FUEL HANDLING SYSTEM OF INDIAN PHWRs FOR BETTER REDUNDANCY AND REDUCTION OF CATANERY HOSES

Refuelling Technology Division

A new flow power circuit was proposed for the fuel handling system in 1986. The new fluid power control circuit consists of two electro hydraulic P-Q proportional valves connected in parallel. One valve shall be in operation at a time, the other being the stand by. Anyone of the P-Q valves can be selected with the help of directional valve MVD-1. The P-Q valve shall control both flow rate and pressure of all the actuators.

As per Fuelling Machine operation programme, only one actuator shall be in action at a time, out of several actuators. At each step of the programme, one pressure setting and one flow set ting is required for a particular actuator. These set points are fed to the P-Q valve. The selection of a particular actuator and its direction control, is achieved, with the help of individual directional valves, mounted on or nearby the respective actuator. The particular actuator will actuate in required direction with controlled speed, as per the flow set point "UQs". If the load increases during the movement either by increase in the resistance in motion or the mechanism stalled against some fixed structure, the actual supply pressure to the actuator, the pressure limiting controller will limit the pressure up to the set point "UPs" and hence the force or torque developed by the actuator, will be proportional to the "UPs".

The actuators and P-Q valve are interconnected with the help of two long rubber hose catenaries. All the actuators shall be actuated with the help of these two hose catenaries. One drain line hose is also provided to facilitate the passage of oil leakage, from the different actuators to the oil tank, If pressure line hose fails, the tank line hose can be used as pressure line hose. This switching

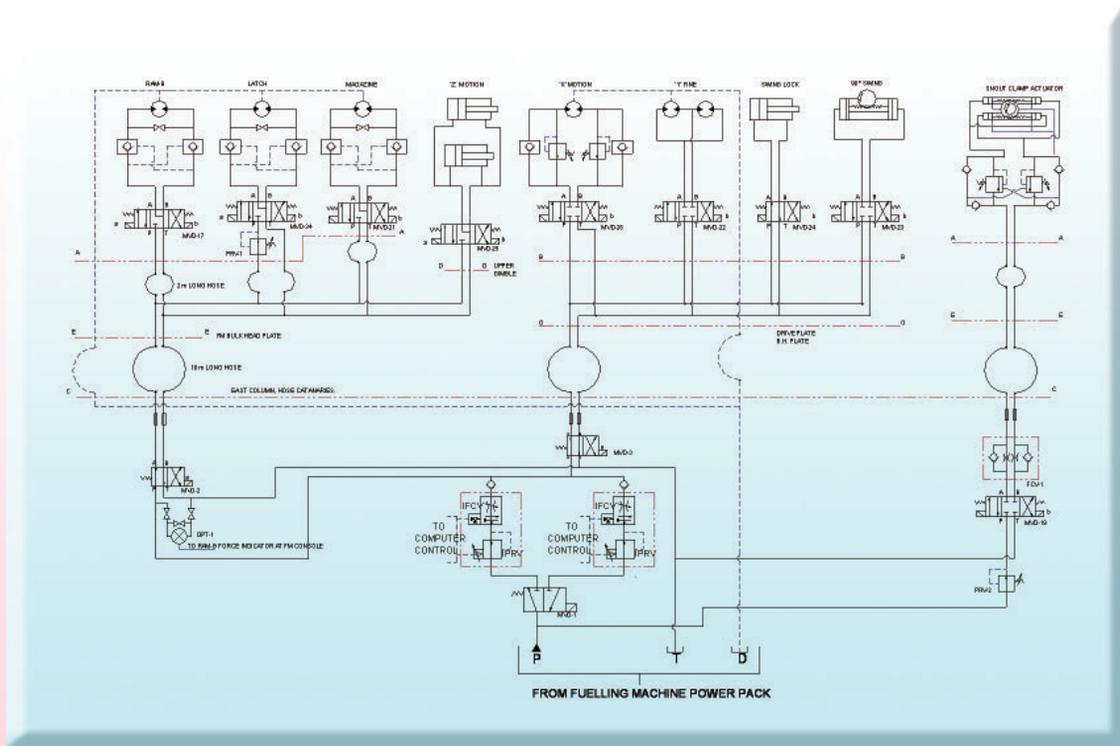
over can be achieved with the help of emergency operation of directional valve MVD-2. All the emergency operations can be carried out with this change and fuelling machine can be safely taken outside the radioactive area.

In case any directional valve of any individual actuator fails, the operation of that actuator can be carried out with help of emergency operation. It will be as a stand by directional valve, common to all individual actuators.

### Advantage of Proposed Fluid Power Circuit

Following are the main advantages of using this new proposed circuit.

- Improve in redundancy: The complete system redundancy has improved at all levels. The pressure control circuit and flow control circuit have redundancy because of two P-Q valves connected in parallel. Hose redundancy is achieved, by switching pressure line hose. MVD-2 will act as redundant for all individual actuator circuits.
- Reduction in number of component.
- Most of the components are located in accessible area.
- Pressure and flow parameters of any actuator can be set or reset remotely from control room.
- Any number of pressure and flow set points can be provided for any actuator.
- Acceleration and deceleration of any actuator can be controlled with the help of electronic ramp generation circuit.



Oil Hydraulic Circuit using P-Q valve for FM and Carriage

- The feedback of actual pressure acting on the actuator can be brought to control room and it can be used for display or for further control purpose.
- The 'X' drive servo-hydraulic system is eliminated.
- Considerable reduction is achieved in tubing work at fuelling machine head and at valve panel. The number of fittings and joints is also reduced. Number of hose catenaries reduces drastically.
- The space required for complete system, inside the reactor building, would be very small, compared to conventional system.
- If power fails, P-Q valve will bring the actuator pressure to zero, which will be a failsafe operation.

### Conclusion

The proposed circuit improves the redundancy of the complete system and its controllability. It eliminates large number of valves, fittings, tubing, hoses and joints. It also eliminates a large number of long rubber hose catenaries. A similar circuit can also be adopted in various other industrial applications, where large number of actuators are used and each actuator requires various pressure and flow set points. This type of circuit will be more economical for such applications. This circuit along with its controls was installed at FMTF and successfully demonstrated, by operating 220 MWe FM.

## RADIATION STREAMING ANALYSIS OF HORIZONTAL PENETRATIONS IN CALANDRIA VAULT OF TAPS 3 & 4

M. V. Gathibandhe, A. K. Kamath and R. A. Agrawal  
Reactor Projects Division, BARC  
and  
Manish Kaushal and A.S. Pradhan  
RSAD, NPCIL

In TAPS 3 and 4 (540 MWe PHWR) there are a total of 19 penetrations, penetrating the east and west walls of calandria vault. Space adjacent to these 16 penetrations is accessible during reactor operation. Compared to this, there are only six IC (Ion Chamber) penetrations of different shielding arrangements, in 220 MWe PHWR. These penetrations pass right through the core, vault water and calandria vault and open outside in the habitable area.

Estimate of radiation streaming through these long penetrations is complicated. Methods based on semi-empirical formula are applicable for limited types of geometry and involve many limiting assumptions. Codes based on Monte Carlo methods are not suitable, due to excessive computer time and statistical error. Transport theory codes based on discrete ordinate methods, are also not suitable for these types of calculations.

For radiation streaming estimate through such long penetrations, a methodology needs to be developed and validated. Analysis is carried out using discrete ordinate transport theory code DOT-III, for larger diameter penetration, with asymmetric 210-angle quadrature set, with more angles in the direction of penetration. Results were corrected for actual diameter, using semi-empirical formula. The method is validated with computer code MCNP (for smaller length) and also with the measured values for 220MWe IC penetrations. Using the above method, streaming dose rate calculation through all the horizontal penetrations in 540 MWe was carried out and

shield arrangement was modified or augmented, wherever required.

### Description

*Ion Chamber (IC) Penetrations - 220 MWe:* There are three penetrations each in the west and east wall of the calandria vault. 52.6 cm long lead shield plug, at one end of the penetration, is provided.

*Ion Chamber (IC) Penetrations - 540 MWe:* There are three penetrations each in the west and east wall of the calandria vault. These penetrations start from 2cm distance from the calandria shell and are 82 cm away from each other (center to center). Detailed geometry/location of the penetrations is given in Figs. 1 to 3. 180cm long shield plug made up of carbon steel (105cm) and borated wood (75cm) (Jabroc) is provided for shielding.

*Horizontal Flux Unit (HFU) Penetrations:* All the HFUs are located on the west side of the reactor. HFUs penetrate the full core from east to west, come out of the calandria vault, crossing vault water and concrete wall of the calandria vault. The locations of all the seven HFUs are shown in Fig. 1. These are 2 cm radius standpipes, running through the core. Each standpipes carries an assembly of 8 carrier tubes. Generally, these tubes carry MI cable connecting Self Powered Neutron Detectors (SPNDs). However, some of the HFUs are empty, without MI cable and SPND. Details are shown in Fig. 4. There is a 30 cm long steel plug in the standpipe (Fig. 4). This plug has 8

bores for passing carrier tubes.

**Shut Down System 2 (SDS 2):** All SDS 2 penetrations are located on the west side of the reactor. There are six SDS 2 penetrations in the west wall of the calandria vault (Fig. 5). A standpipe of radius 2.25 cm, runs through each penetration in the west wall. Subsequently, the standpipe runs through the vault water, enters in reflector region and then crosses the entire core. The standpipe is always filled with D<sub>2</sub>O.

**Computer Code**

Analysis was done using computer codes DOT-III and MCNP.

**Method of Analysis**

Neutron and gamma dose rate calculations through the penetrations for IC, HFU and SDS 2 have been carried out using code DOT-III with larger diameter of penetration. Correction for actual diameter was done using semi-empirical formula. Method was validated for 1m length of penetration using computer code MCNP.

Core is homogenized and partial core, adjacent to the penetrations is considered to be effective for streaming calculations. 210 angle of scattering (quadrature set) has been used for better accuracy. Neutron dose rate calculations are done using 100-group neutron cross-section library DLC-2. For core and capture gamma calculations, 40 group (22 neutron and 18 gamma) coupled library CASK has been used. P<sub>3</sub>-S<sub>8</sub> calculations, in fixed source mode and point convergence has been carried out in R-Z geometry. Reflective boundary condition on left and vacuum boundary

conditions on other sides is used.

**Ion Chamber (IC) Penetrations:** Neutron and gamma streaming analysis is carried out for IC penetrations of 220 MWe and 540 MWe PHWRs. The models used in DOT-III for 540 MWe are given in Fig. 6. Calculations were done for 220 MWe and the results were compared with the measurements at KGS-1&2.

**Horizontal Flux Unit (HFU) Penetrations:** HFU-3/5 is selected for analysis, as these penetrations have minimum length among all HFUs. Fig. 7 gives the geometrical model used in DOT-III. HFU penetrations

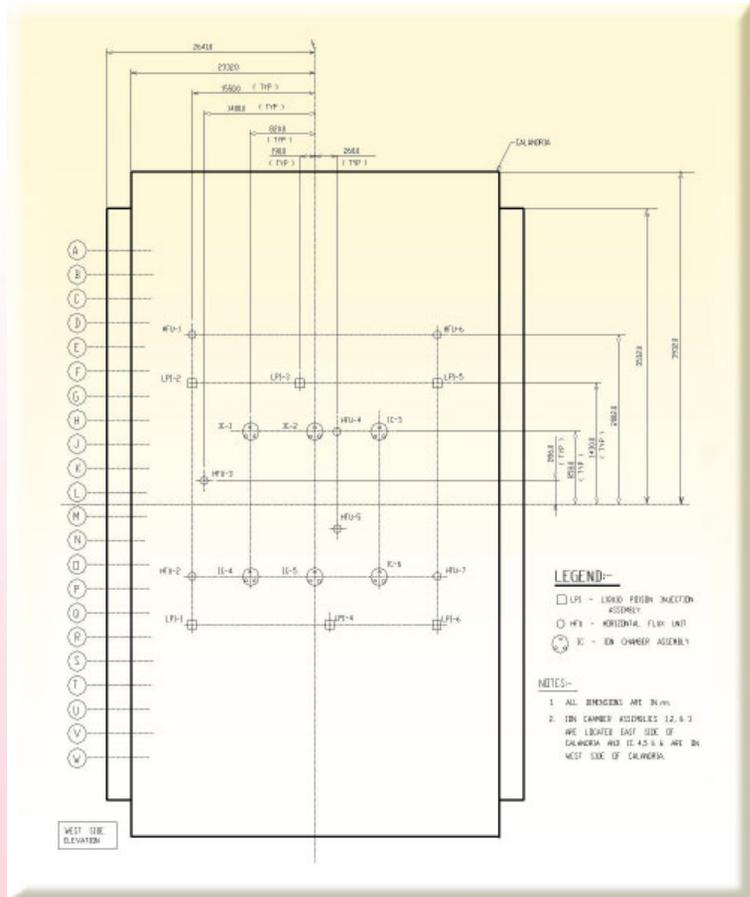


Fig.1: Location of horizontal reactivity devices horizontal flux units and ion chambers

were analyzed, assuming a penetration diameter of 8 cm, in place of 4 cm. This was done to get better convergence and to avoid negative flux problem. Adjustment for actual 4 cm size was done using semi-empirical correlation. Streaming analysis for the following cases was done.

- (1) An empty penetration of diameter 8 cm and length 476 cm from core edge.
- (2) Penetration of diameter 8 cm with carrier tubes filled with Mineral Insulated (MI) cable (equivalent to 10% volume) up to full length.
- (3) Penetration of diameter 8 cm with carrier tubes filled with Aluminum cable (equivalent to 10% volume) up to full length.

*Shut Down System 2 (SDS 2):* Full length of the penetration (f 2.25 cm) is filled with D<sub>2</sub>O. Neutron and gamma streaming calculations are done for 8 cm diameter penetration as in the case of HFU.

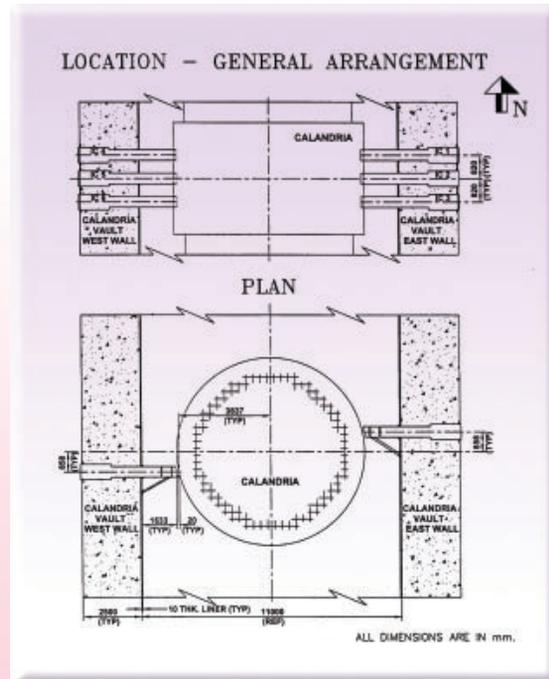


Fig. 2: Ion chamber housing assembly

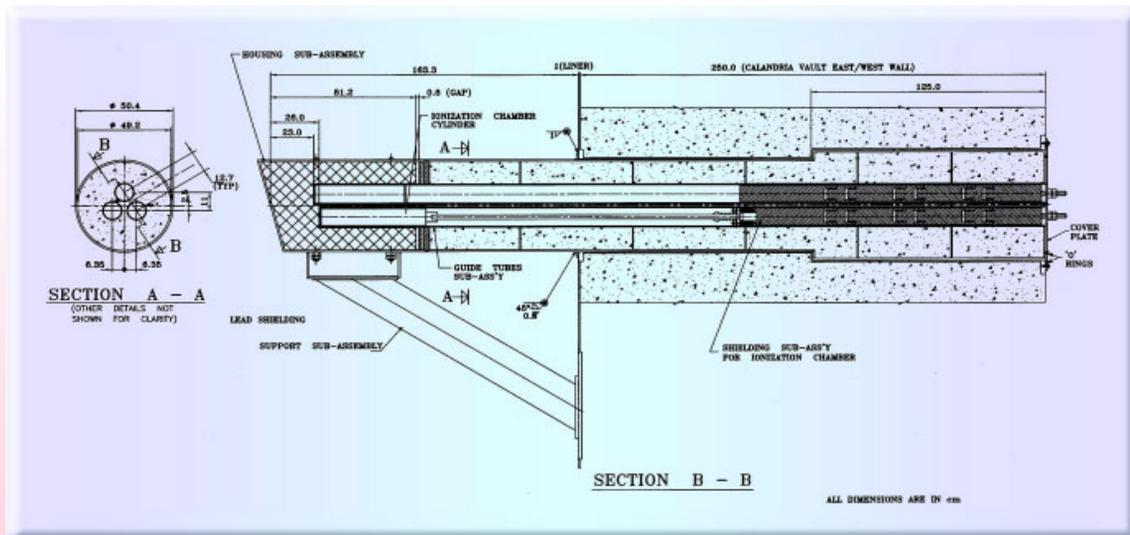


Fig. 3: Shielding arrangement of ion chamber housing



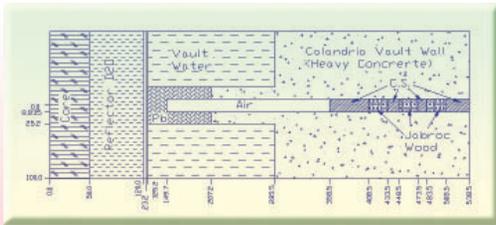


Fig. 6 : Model of ion chamber 540 MWe

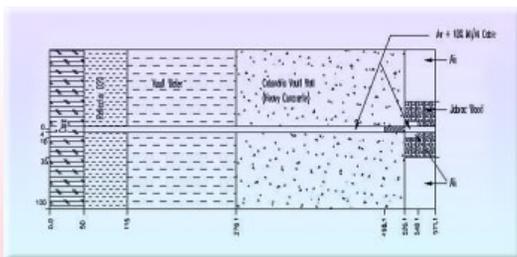


Fig. 7: Model of HFU – additional shield of jabroc wood

### Validation

*Ion-Chamber – 220MWe:* Neutron Dose rate of 35.3 mR/hr through ion-chamber penetration has been calculated using DOT-III. Absence of neutron absorbing material inside the penetration is the cause of higher neutron dose rate. The measurements in KGS 1&2 indicate neutron dose rate of 40 mR/hr at full power on contact. Good matching between calculations and measurements validates the method of analysis. Method of analysis was also validated against the results, from code MCNP, for 1m length of penetration.

### Results and Discussion

*Ion-Chamber – 540 MWe:* Neutron and gamma dose rates through ion-chamber penetration is plotted in Fig. 8. Dose rate are given in Table 1.0.

As 180 cm long shield plug has 75 cm long Jabroc wood, the neutron dose rate is low as compared to 220 MWe. Gamma dose rate of 0.78 mR/hr is calculated on the

surface of the vault wall. Shielding offered by shielding plug in the penetration is better in 540 MWe as compared to that in 220 MWe.

*Horizontal Flux Unit (HFU):* Variation of neutron and gamma dose rates through HFU penetration of diameter 4.0 cm has been plotted in Fig. 9. Similar figures have been plotted for HFU penetrations with 10% volume in stand pipe filled with MI cable (Fig.10) and 10% volume in stand pipe filled with Aluminum cable (Fig. 11). The dose rate values on vault wall surface around HFU penetrations are given in Table 1.0. Total dose rate of 190R/hr, through empty HFU standpipe is dominated by neutrons.

Dose rate reduces to ~500 mR/hr, when effect of Al

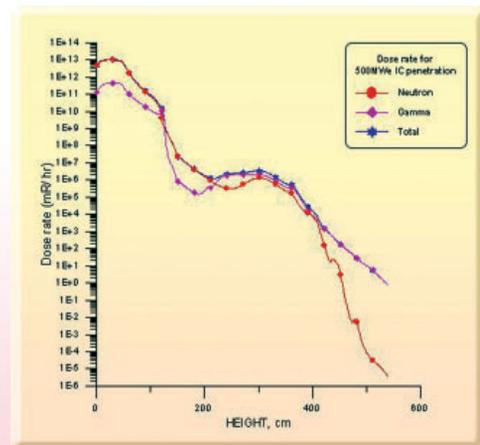


Fig. 8 : Variation of dose rates along I/C penetrations - 540 MWe

Penetrations	Dose rates, mR/hr		
	Neutron	Gamma	Total
I/C 540 MWe; on axis	negligible	0.78	0.78
HFU empty	1.45E+0	4.54E+0	1.9E+05
HFU carrier tube (a) filled	65.11	36.59	101.7
HFU Q filled with Al Cable	291.6	153	444.5
SDS#2; on axis	-	-	-

cable along with shield plug is taken into account. Use of MI cables in place of Al cables, further reduces the dose rate to ~ 110 mR/hr.

*HFU with Jabroc shielding around Carrier Tubes (CT):* Additional shielding is required to be put on the outer surface of concrete, to reduce neutron dose rate to permissible values. 25cm thick Jabroc wood with MI cable in CT (as shown in Fig. 7) is recommended. With Al cable in CT, 35cm thick Jabroc will be required.

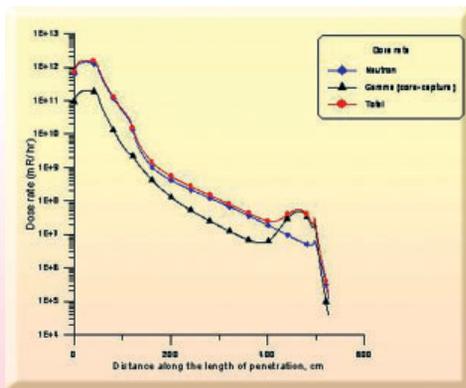


Fig. 9 : Variation of dose rates along the penetration of HFU

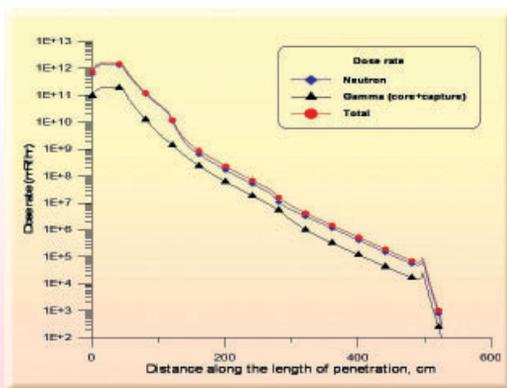


Fig. 10 : Variation of dose rates along the penetration of HFU - 10% volume of stand pipe filled with aluminum cable

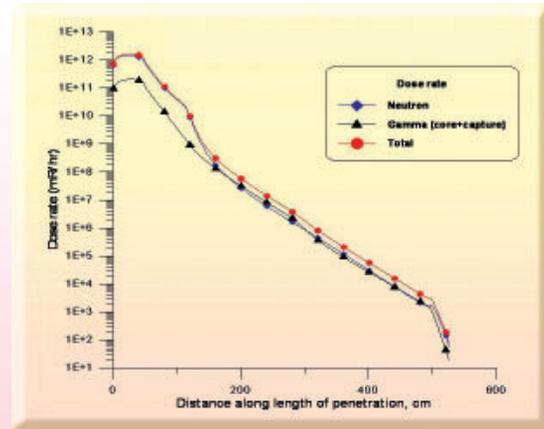


Fig.11: Variation of dose rates along the penetration of HFU - 10% volume of stand pipe filled with MI cable

*Shut Down System 2:* Dose rates through SDS 2 penetration are given in Table 1.0. As the penetration is always filled with  $D_2O$ , neutron and gamma dose rates are low.

### Recommendations

*Ion Chamber:* 180 cm shield plug of steel and Jabroc wood, placed inside Ion Chamber penetration looks to be adequate.

*Horizontal Flux Unit:* MI cable should be routed through the carrier tubes starting from core reflector interface and an additional neutron shielding of Jabroc wood of 25-35 cm thickness all around.

*Shut Down System 2:* Neutron and gamma dose rates on contact of SDS 2 standpipe, are within acceptable dose rate limit.

## DESIGN AND DEVELOPMENT OF WATER HYDRAULIC AUTO-DIFFERENTIAL PRESSURE CONTROL VALVE IN FUELLING MACHINE RAM B AND RAM C CIRCUITS

Refuelling Technology Division

On-power refuelling is carried out, with the help of two robotized fuelling machines FM, located at each side of one of the horizontal reactor coolant channels, which is to be refueled. The operation of pushing the fuel bundle in the reactor, is carried out with the help of rams. The RAM is a heavy water operated piston cylinder. These RAMS perform various functions at different force levels and directions. The force setting of the RAMS, depends upon the differential pressure between FM, Housing pressure and ram supply pressure, since the RAMS are handling spent nuclear fuel bundles, which are fragile and therefore, the force control becomes a critical parameter, as far as fuel safety is concerned. Hence, the RAM pressure control should be precise and it should remain within prescribed limits, even after fluctuation in pump supply pressure, FM pressure and flow requirement for RAM supply.

The force, direction and velocity control of RAM C was carried out with conventional control valves. This system requires large piping, lot of transducers, control valves, accumulators and imported components.

RTD developed an Auto Differential Pressure Control Valve (ADPCV), that can control force, direction and velocity of RAM B and RAM C of fuelling machine. Use of ADPCV eliminates lots of imported components, transducers and piping work.

### Working Principle of ADPCV

Fig. 1 shows the cross section of ADPCV. The ADPCV consists of a moving double piston and a poppet

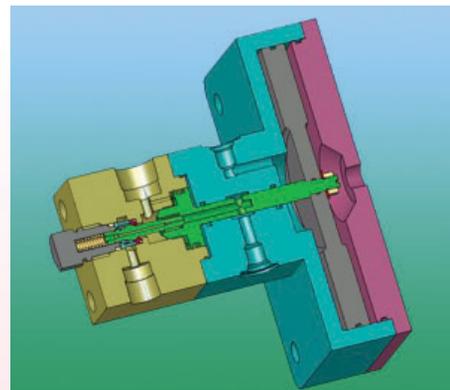


Fig. 1: Auto-Differential Pressure Control Valve

arrangement. The smaller piston, called balance piston, is subjected to the valve outlet pressure ( $P_o$ ) at one end and at the other end, it is subjected to FM Housing pressure ( $P_{ref}$  called reference pressure).

The bigger piston is piston of air motor, that applies additional force on the balance piston. Thus, for the balance piston to be in equilibrium: Force generated by  $P_o$  on Balance Piston - Force generated by  $P_{ref}$  on

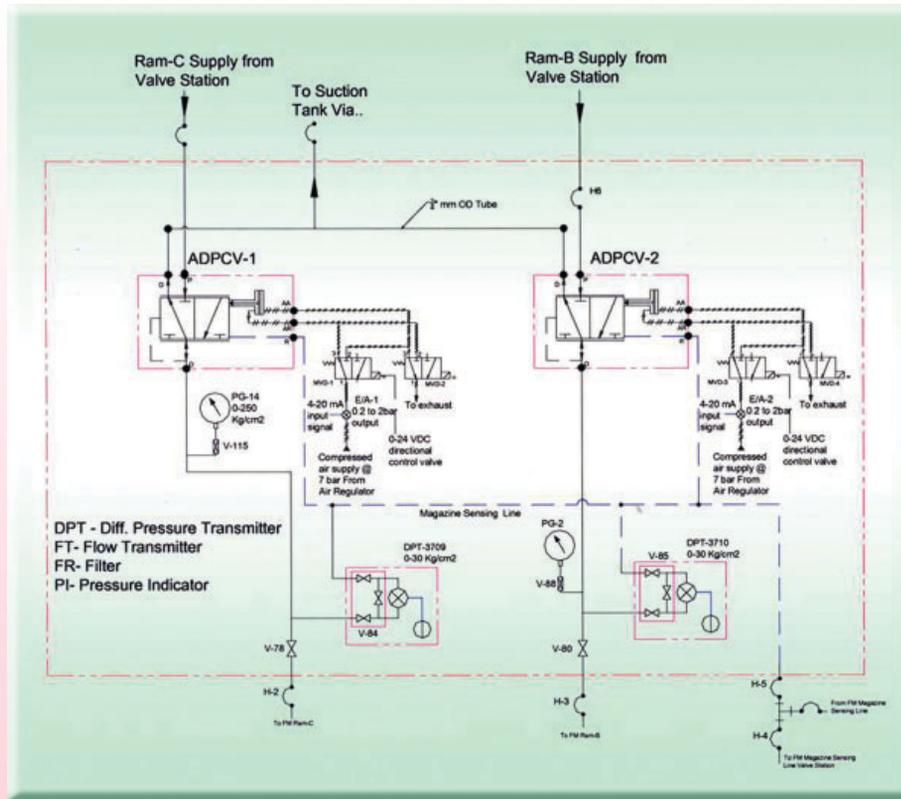


Fig. 2 : Circuit diagram of FMTF where ADPCV is used for force, direction and velocity control of RAM C

Balance Piston = Force generated by Air signal on Air Piston.

Therefore, a constant differential pressure equivalent to air pressure signal applied to the air motor piston, is always maintained between the outlet and reference ports of the valve. Air pressure signal is generated using I/P converter. The valve has provision to apply the air pressure on head as well as rod side of the piston. If the air pressure signal is applied to the head side of the air motor, it will maintain a positive differential between outlet and reference port, making the RAM to advance. If the air pressure signal is applied to the rod side of the air motor, it will maintain a negative differential between outlet and reference port, making the RAM to retract. While retracting, the poppet of the valve will fully close the

main orifice and the small piston will still move back, allowing orifice 2 to open. This will make the water from the actuator to flow to drain connection, thus retracting the RAM C.

### Specifications

Supply Pressure : 147 bar  
 Reference Pressure : 94 – 100 bar  
 Max. operating Pressure : 150 bar  
 Max. operating temperature : 600° C  
 Max. Operating differential pressure to be maintained : 35 bar and 10 bar  
 Min. operating difference Pressure : 0 bar.  
 Max. Fluctuation in supply Pressure : 5 bar  
 Max. Fluctuation in Retract Pressure : 2 bar

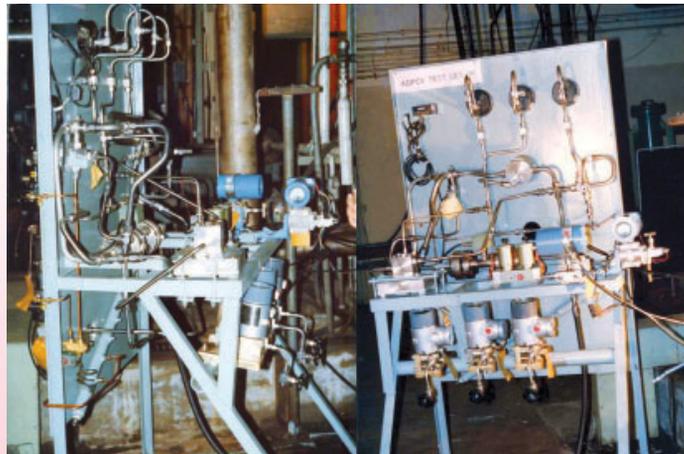


Fig. 3: ADPCV test set-up

Max. Fluctuation in flow : 30 lpm range 0.30 Lpm  
Max fluctuation of controlled force allowed  
in RAM C : 10% of set point.

- Accumulators.
- Controllers.
- Large piping works.

### Advantages

- Less components/joints and occupy less space and smaller hold up capacity
- Simple in design
- Response to any change in operating condition will be very fast and no accumulator shall be required for increasing the system time period
- The electrical signal can easily be given from the microcomputer, which controls the overall FM operating process
- System occupy less space in reactor building
- Less numbers of components, less no. of joints, hence less D2O leakage and less radioactive problem
- Cost wise cheaper and fully indigenous.

Following components can be replaced by a single ADPCV in FM circuit

- Two air actuated control valves
- Different pressure transmitters
- W. H. directional control valves

### Testing of ADPCV

The performance of ADPCV developed at RTD has been evaluated in ADPCV test facility as shown in Fig. 3

Effects of following parameters have been studied on ADPCV.

- Reference Pressure
- Supply Pressure
- Flow through ADPCV
- Flow air pressure in advance air motor chamber.
- Air pressure of retract chamber of air.

To study dynamic characteristics of the valve, following tests have been carried out

- Step input of pump supply pressure
- Step input of air pressure signal
- Step of output flow.

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





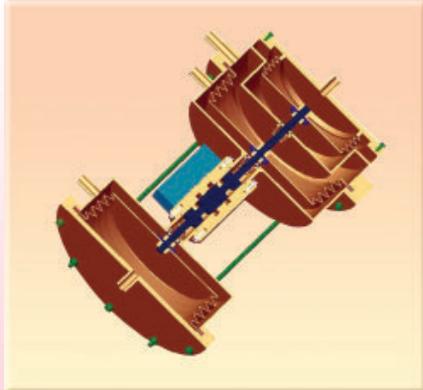


Fig. 3: Schematic cross sectional view of Level Control Valve

from Water Inlet Header (WIH) to inlet port of ZCC. In central position or in equilibrium position, a minimum flow is circulating from WIH to ZCC and ZCC to delay tank, for maintaining the temperature and to carry away the tritium activity. If the water level of the ZCC is lower than the signal available at the gas motor, gas motor will push the spool in such way that it will increase the level of ZCC. If the level of ZCC is more than the signal at gas motor, the spool will shift in such a way, that the level in the ZCC will decrease.

The corrective forces for handling the disturbance, friction and inertial force on the spool can be increased by taking sufficient size of balance bellow. The differential pressure across the balance bellow, is about 0.02 to 0.5 bar which is equal to the head due to water level in the compartment plus pressure drop in lines.

The valve uses no dynamic elastomeric seal. Labyrinth are used to reduce the inter port leakage. The spool is supported by the hydrostatic bearing, which keeps the friction in the system to minimum. This helps for fast response and accurate positioning of the valve. The prototype LCV has been designed for 57 lpm flow and a test facility has been designed to test this valve for long term endurance test for generating reliability data of this newly developed valve. This single valve will replace control valves, differential pressure

transmitter and level controller from the existing system and it will respond very fast and corrective action will be taken promptly.

### Constant Level Valve

The cross section of the constant level valve is shown in Fig. 4. In the constant level valve, the drain from the tank is connected as inlet to the valve. This inlet pressure acts on one side of a diaphragm in the valve. The gas pressure is taken as a reference pressure to this valve, which acts on the other side of the diaphragm. An additional spring force which is equivalent to the level to be maintained, also acts on this side of the diaphragm. Thus, for the diaphragm in equilibrium:



Fig. 4: Schematic cross sectional view of Constant Level Valve

Force due to inlet pressure = Force due to reference pressure + Spring force.

Whenever the level in the tank increases above the set point, a net force moving the poppet upward is developed, which opens the valve and allows the fluid to flow to the tank, until the set level is achieved. As soon as the set level is achieved, the valve will be closed and the level will be maintained. The use of this valve, will eliminate the use of components like level transmitter, controller and costly flow control valve from the system.

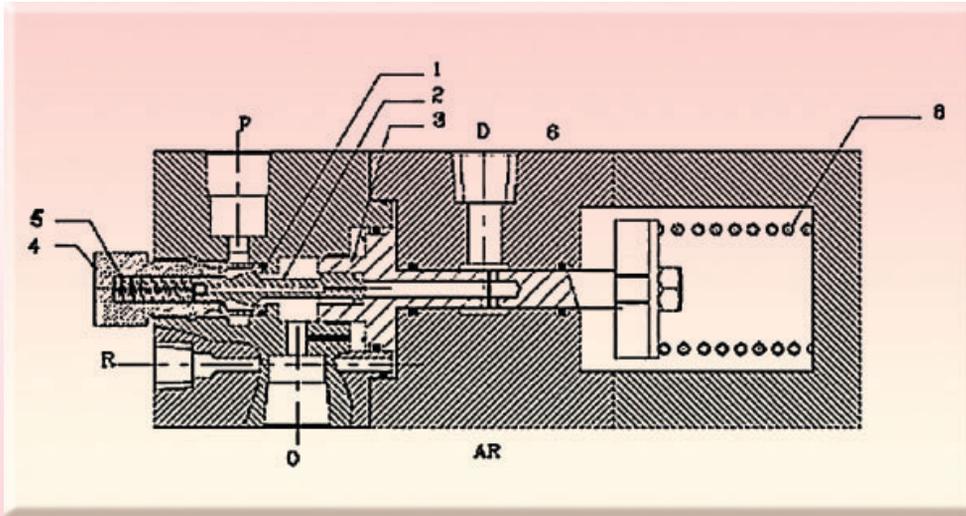


Fig. 5: Schematic cross sectional view of Passive Feed and Bleed Valve

### Passive Feed and Bleed valve (FBV)

Fig. 5 shows schematic of FBV to be used in LZCS. The valve works as both feed valve as well as bleed valve. It can maintain constant differential pressure between reference and outlet pressure of the valve. The valve has two orifices, feed orifice and a bleed orifice. The system in which pressure/ differential pressure is to be controlled, is connected to outlet port (OUT). Tank is connected to drain port (D) of the valve.

The outlet pressure in the valve is acts on one side of a piston/diaphragm. On the other side, the reference pressure is applied, if differential pressure is to be controlled. For pressure control reference pressure is taken as atmospheric pressure. In addition to this reference pressure, an additional force is also applied to this diaphragm, using spring or air motor. This force is equivalent to the required pressure/differential pressure to be maintained. Thus, for the piston/ diaphragm to be in equilibrium:

Force due to outlet pressure = Force due to reference pressure + Force applied by air motor/spring.

The piston/diaphragm is connected to the poppet of

the valve. The change in position of poppet, changes the outlet pressure. The magnitude of differential pressure can be set, by changing the spring force. This valve will eliminate the use of feed and bleed control valves, feed and bleed controller and differential pressure transmitter from the existing system.

### Advantages

The LCV not only controls inlet flow, but also the outlet flow simultaneously. This bi-directional control will make the system more prompt and capable to handle the severe inter-compartment water leakage disturbance problem, due to higher clearance between the bulk head and passing tubes. By adopting the LCV, it is possible to give more manufacturing tolerance to the components. This will reduce the manufacturing cost and rejection. Since, the level control system using these valves will make an open loop control system, it will eliminate the problem of feedback control system like: oscillation, improper tuning slow response and high stability time. The use of these passive valves, will eliminate the use of large number of components like level transmitter, level controller, control valves, differential pressure transmitters etc. from the system.

## DEVELOPMENT OF MAN-RAM SAVING TOOLS FOR 540 MWe PHWRs

Refuelling Technology Division

### Design and Development of End fitting blanking assembly

540 MWe Pressurised Heavy Water Reactors (PHWRs), consist of 392 coolant channels. Heavy water(D<sub>2</sub>O) flows inside the channels, at about 100 bar pressure and around 300°C temperature. 784 sealing plugs are used, to close the coolant channel to maintain pressure boundary. Each sealing plug has metallic seal, which butts against a machined seal face, provided in the end fitting, at each end of the coolant channel. When the reactor is on power, it is necessary to do refuelling operation of the coolant channels. During refuelling, coolant channel sealing plugs are removed remotely, by the fuelling machines and restored after the operation. It is observed that, because of repeated installation / removal operations of sealing plug and due to any foreign materials, the seal face of the end fitting, is likely to get damaged. This damage causes unacceptable leakage from the sealing plugs.

In such case boxing-up of the channel become difficult and the situation may warrant the long shut down of the reactor for attending to the problem immediately.

This imposes economic penalty and affects the capacity factor of the reactor.

It is desired to arrest such seal plug leakage temporarily by using a special device, which can blank the end fitting. Use of such devices can avoid the long emergency reactor shut down and facilitates attending to the problem during the next opportunity of planned reactor shut down. End fitting blanking assembly is designed to blank the defective channel temporarily. It is easy to install and remove from the channel. This assembly can be manually installed in about one minute time, on the problematic end fittings. Use of this device does not prevent the fuelling machine from refuelling the neighboring channels. Normal fuelling and de-fuelling operations can be performed in the adjacent channels. Pressure releasing system is also available with the device, which ensures safety during operation.

A prototype end fitting blanking assembly for a 540 MWe (Fig. 1) was manufactured. It was subjected to hydro test pressure of 125 Kg per sq. cm. Leak performance of the device

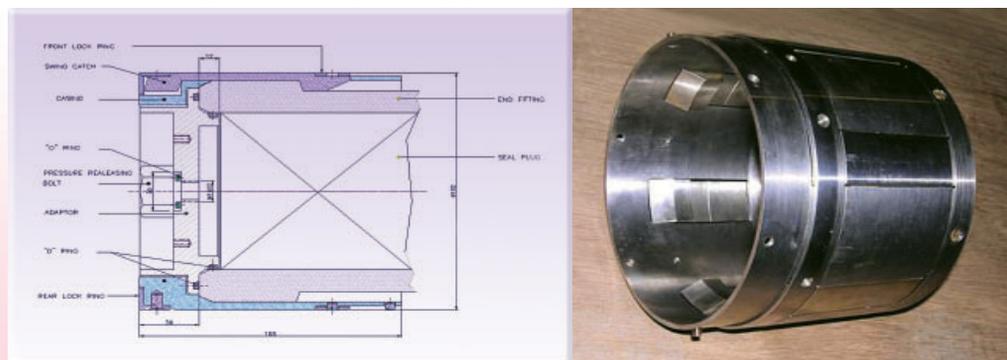


Fig. 1: 540 MW(e) PHWRs End Fitting Blanking Assembly

was satisfactory. The device is ready for use in TAPS 3 & 4.

**Channel flow blocking plug AND flow blocking extension**

540 MWe Pressurised Heavy Water Reactors (PHWRs) consists of 392 coolant channels. Channels are directly connected to the feeder pipes and heavy water flows inside the coolant channels. During emergency maintenance work, it is necessary to isolate the channel from the feeder line, to obtain a dry channel. Ice

plugging is a common method used for these purposes. It is necessary to block the flow inside a channel, prior to carrying out ice plugging. In 540 MWe coolant channels, liner tube water inlet / outlet passages are about 2 meters inside from 'E' face. It becomes difficult to block the flow inside the channel with the help of a single plug. A combination of channel flow blocking plug (Figs. 2 and 3) and channel flow blocking extension (Figs. 4 and 5) has been designed, developed for this purpose. The device has been tested satisfactorily. Sketches and photographs of the devices are given below.

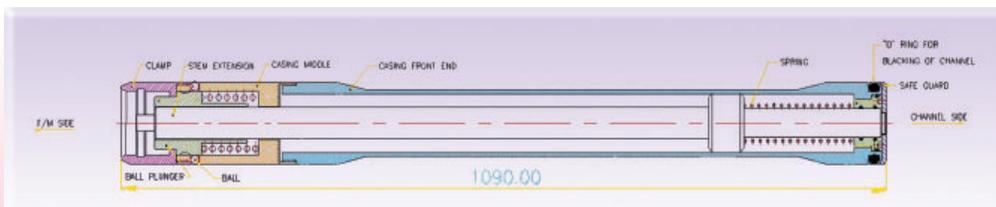


Fig. 2: Channel Flow Blocking Plug (sectional view)



Fig. 3: Channel Flow Blocking Plug

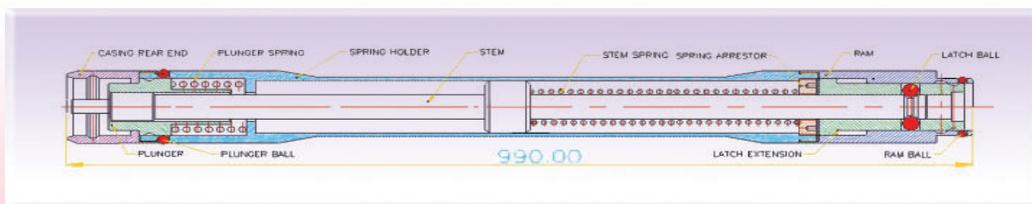


Fig. 4: Flow Blocking Extension (sectional view)

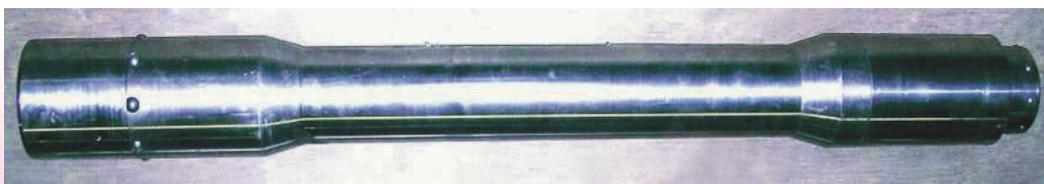


Fig. 5: Flow Blocking Extension

These tools are ready for deployment in the reactor.

### Development of Special Seal Plug for BARCIS

A special seal plug (Fig. 6) has been designed for providing a passage of 50 mm diameter. through its center. It allows the entry of BARCIS drive tube for

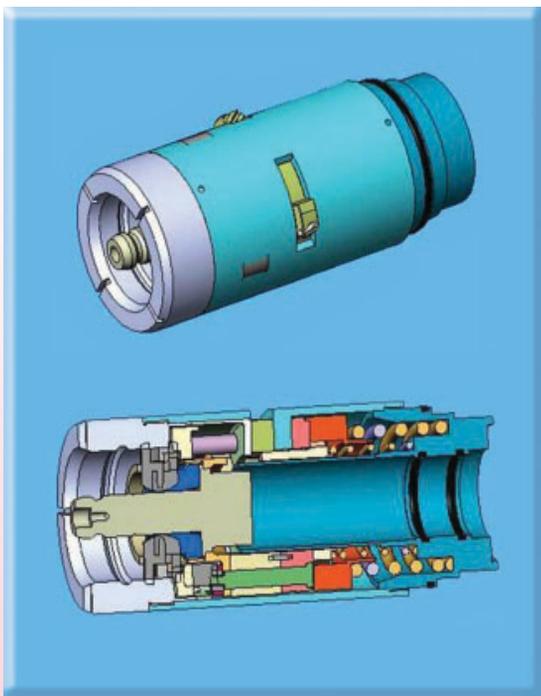


Fig. 6: Special Seal Plug for BARCIS of 540 MWe PHWRs

540 MWe PHWR. Detailing of a new concept was carried out incorporating advanced features like use of jaws for better strength, safety latch operation by Latch ram for reliable operation. Detail design has been worked out. Design has been verified by 3-D models. A prototype is under manufacture at CDM.

## ANNOUNCEMENT

Forthcoming Symposium

### TROMBAY SYMPOSIUM ON DESALINATION AND WATER REUSE (TSDWR 07)

The DAE-BRNS has organised a three-day symposium from Feb. 7-9, 2007, at the Multipurpose Hall, Training School Hostel and Guest House, Anushaktinagar, Mumbai. It is being co-sponsored by the Indian Desalination Association and The International Journal of Nuclear Desalination. The technical programme would consist of invited talks, oral and poster presentations. Extended abstracts of not more than 500 words with complete details about the authors (including e-mail IDs) are to be sent to the Secretary, Organising Committee, through e-mail at [tsdwr07@gmail.com](mailto:tsdwr07@gmail.com). The final paper in the prescribed format is to be submitted both as hard and soft copies. The topics to be covered in the symposium would be :

- Water Scenario in India
- Sea Water Desalination Technologies
- Brackish Water Desalination Technologies
- Waste Water Treatment, Water Recovery and Water Reuse
- Nuclear Desalination
- Reject Brine Management including Recovery of valuables
- Financial & Infrastructural Aspects
- Innovative Desalination Technologies
- Water Purification Technologies
- Integrated Water Resource Management
- Rain Water Harvesting
- Drinking Water supply for Rural and Remote Areas
- Research and Development Scenario
- Water Quality Assurance and Monitoring

For further details please contact :

**Dr P.K. Tiwari**

Chairman, Organising Committee &  
Head, Desalination Division,  
BARC, Trombay,  
Mumbai - 400 085, India  
Tel.: 0091-22-25594619/25594625  
Fax: 0091-22-25505151/25519613  
E-mail : [pktiwari@barc.gov.in](mailto:pktiwari@barc.gov.in)

**Dr S. Prabhakar**

Secretary, Organising Committee &  
Head, Separating Technology Section,  
Desalination Division,  
BARC, Trombay, Mumbai - 400 085, India  
Tel.: 0091-22-25594647  
Fax: 0091-22-25505151/25519613  
E-mail : [sprabha@barc.gov.in](mailto:sprabha@barc.gov.in)

## DEVELOPMENT AND TESTING OF MODIFIED LEVELING MECHANISM FOR 540 MWe F/M HEAD

Refuelling Technology Division

Y-tilt and X-tilt correction is required, during fuelling machine alignment to channel. Problem is faced regarding Y-tilt correction in 540 MWe fuelling machine. Due to heavy load of B-RAM, an imbalance is created in the machine, which needs to be compensated. A system was developed, to achieve automatic control of oil pressure in leveling assembly, which creates a force, to compensate the imbalance generated by B-RAM movement. The machine was balanced with B-RAM at home position. For implementing the scheme in 340 MWe PHWR FM, the existing leveling mechanism was reworked with minimum modification and one O-ring was used to generate a hydraulic piston, for generating the required compensated force.

This has been successfully tested in two different Pre-load of leveling mechanisms i.e. 426 Kgf, 200 Kgf. The oil pressure that compensates the imbalance created by B-RAM movement was automatically getting controlled to the intended values with different position of B-RAM during advancing and also while retracting. The readings of Y-tilt at different B-RAM positions in steps of 100 mm up to 1200 mm and corresponding oil pressure were experimentally measured and tabulated and used in the control scheme. The change in Y-tilt reading up to 1200 mm of B-RAM position was within 0.3/0.4 mm as compared to the initial reading of 9.5 mm.

### Conceptual Scheme

A leveling mechanism using IRV (Instrumented Relief Valve) was developed at RTD as shown in Fig. 1. The IRV controls the pressure to leveling mechanism piston, which generates a force to compensate the imbalance created by B-RAM. A micro-controller based control circuit provides required signal to IRV.

### Controller

A micro controller based control circuit was designed and developed for generating optimum oil pressure in leveling assembly through IRV. The main component of the system is an ADuC831 data acquisition system incorporating 12-bit ADC, dual 12-bit DACs and programmable 8-bit MCU on a single chip. The control

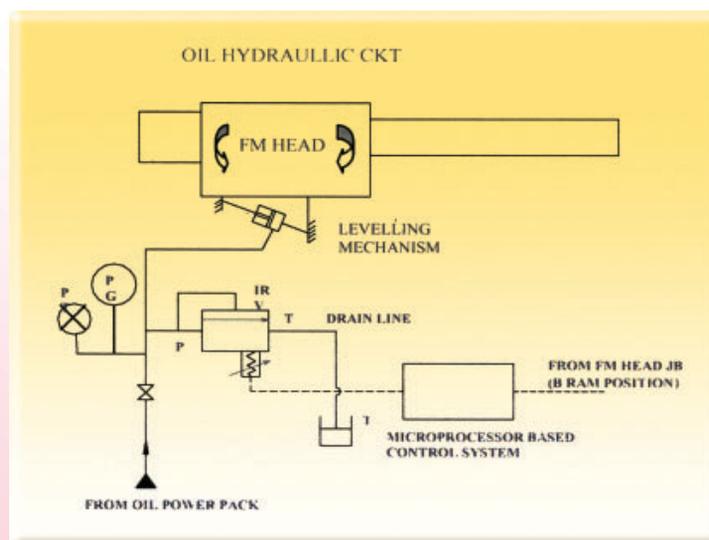


Fig. 1 : Modified leveling mechanism for 540 MWe fuelling machine head

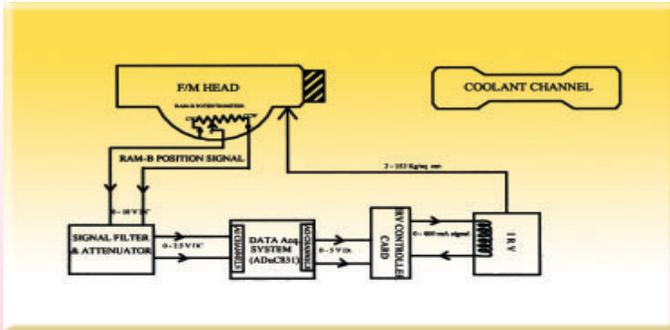


Fig. 2: Control system block diagram for modified leveling mechanism

system is an open loop control system; hence it requires calibration of IRV. The system acquires RAM B position signal as set point tapped from fuelling machine. Range of the RAM B position signal is 0 – 10 V dc whereas, analog input range of the data acquisition system is 0 – 2.5 V dc. Hence, a signal attenuator designed and developed in house, is used for interfacing the signal with data acquisition system. Output of the Aduc831 through its DAC channel (0 – 5 V) is given to the controller card of IRV. The IRV generates pressure proportional to the control signal given to the controller card.

### Experiment

The system was implemented on one fuelling machine. The control circuit and hydraulic circuit were commissioned as per Figs. 1 and 2 given above. Following steps are carried out for using the system:

- Input signal to the data acquisition system ( $V_i$ ) for different positions of RAM B was noted.
- Flow through valve is adjusted initially for the required pressure at RAM B home position. The valve is calibrated by providing signal ( $V_o$ ) from external source to the controller card. The linear relationship between  $V_i$  and  $V_o$  is worked out and implemented in the algorithm, developed for MCU

of ADuC831 data acquisition system.

- The B-RAM was moved to different positions and tilt was observed.
- The experiment was conducted successfully in two different Pre-load of leveling mechanisms i.e. 426 Kgf, 200 Kgf. The oil pressure

that compensates the imbalance created by B-RAM movement was automatically getting controlled to the intended values, with

different positions of B-RAM during advancing and also while retracting. The change in Y-tilt reading up to 1200 mm of B-RAM position, was with in 0.3 /0.4 mm.

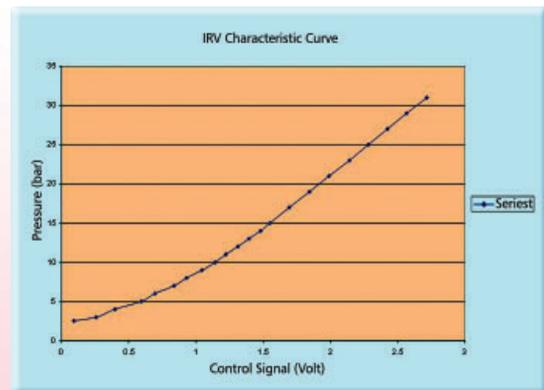


Fig. 3: I/O characteristic curve of IRV

### Conclusion

With this system, the F/M head can be balanced at B-RAM home position and at 200 kg. Preload of leveling mechanism, Y-tilt reading can be achieved within acceptable limits and imbalance created by B-RAM movement can be taken care of by the above-mentioned system. This can be implemented in TAPS 3&4 F/M heads.

## DEVELOPMENT OF RUBBER FACE SEAL FOR 540 MWe PHWR FUELING MACHINE TO END FITTING JOINT AT HIGH PRESSURE AND HIGH TEMPERATURE

Refuelling Technology Division

A rubber face seal and its seal holder was designed, for a 220MWe PHWR and was successfully implemented, for refueling of some of the coolant channels, where normal metallic seals were leaky in shutdown load condition. This seal is being extensively used in various 220 MWe power plants. The seal is made of ethylene propylene rubber (EPR). The conventional EPR material can withstand a temperature of 160°C, where as the end fitting face temperature is of the order of 260°C. The EPR can be used for this high temperature for short duration (order of

1 hour). During this refueling time, cold water is flowing from fueling machine to channel. Due to this, the end fitting face temperature reduces from 260°C to less than 100°C within 5 minutes and for the rest of the refueling time, the end fitting temperature remains low. Therefore, the EPR seal is in high temperature for not more than 5 minutes and the same seal can be reused for refueling many more channels, before it eventually fails. This was verified experimentally. For the testing, a small test setup was made and the seal was subjected to high pressure and high temperature conditions.



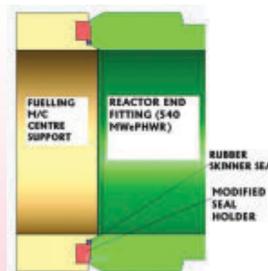
Rubber face seal holder for 500 MWe PHWR



New rubber face seal for 500 MWe PHWR



Rubber face seal after 120 cycles



Fuelling M/C to end fitting rubber skinner seal (540 MWe PHWR)

It gave a good performance even up to 16 hours of high temperature. The 500 MWe fueling machine test facility (FMTF) was used, for testing the rubber seal. It was subjected to 120 cycles of refueling channel operation under full high-temperature high-pressure conditions. The seal has successfully passed the test. When it was removed from the FM, minor extrusion marks were observed (refer figure). The Face seal was again installed with some cleaning. The seal was working perfectly. Later on, the seal successfully passed 225 cycles of channel operation, without failing. The seal is undergoing endurance test at FMTF, BARC.

### Conclusions

The rubber seal and seal holder replaces the existing metallic seal holder of the fueling machine. The normal procedure of checking fueling machine alignment remains same. The seal has successfully passed 225 channel operations and was healthy for reuse.

The rubber Face seal made of indigenous EPDM has given good performance. But in international market high temperature resistant EPDM has arrived which can withstand up to 270°C for continuous operating conditions. These special materials will further strengthen the confidence in the rubber as a seal material for Face seal application. Compare to the metallic seal, rubber Face seal is more capable to handle defective / scratched end fittings and at the same time they are easy to fabricate, easy to install, easy to remove, easy to stored for long time and above all cost wise very cheap.

## ANNOUNCEMENT

Discussion Meet on

### CURRENT TRENDS AND FUTURE PERSPECTIVES OF NEUTRON ACTIVATION ANALYSIS (CFNAA - 2006)

A two-day (Nov. 16-17, 2006) DAE- BRNS Symposium has been organised by the Radiochemistry Division, BARC on the above topics at the Multipurpose Hall, BARC Training School Hostel & Guest House, Anushaktinagar, Mumbai. The scientific programme of the discussion meet include invited talks by speakers as well as review paper presentations.

The scope of the meet:

- Development and applications of NAA and PGNAA
- Chemical and Speciation NAA methods
- Fast neutron NAA
- Gamma ray spectrometry and software for spectral analysis in NAA
- Method validation and quality assurance
- Uncertainty measurement
- Position of NAA in comparison to other competitive techniques.

For further information please contact:

#### Dr R. Acharya

Secretary, CFNAA 2006  
Radiochemistry Division,  
Bhabha Atomic Research Centre,  
Trombay, Mumbai - 400085, India  
Fax: 91-22- 25505151 / 2551 9613  
Phone: 91-22-2559 4089 / 25593862  
E-mail: racharya@barc.gov.in  
Home page: www.barc.gov.in

## SAFETY AND RELIABILITY ASSESSMENT OF 540 MWe PHWR

Reactor Safety Division

### Thermal Hydraulics

Computer code DYNA540 was developed, validated and applied for TAPS 3 and 4 reactors to simulate various systems e.g. Primary Heat Transport and Steam Generators, and components like pumps, different types of valves, and controllers. (Fig.1). Validation of code was done using plant data. It is applied for optimizing performance of systems, evaluation of system capabilities and response, sizing of valves, selection of controller logics. It was also used for independent verification of transient studies carried out by NPCIL.

Fig. 2 illustrates validation of code with plant data for load rejection test at 90 % power.

Performance of three different Steam generator Pressure Controller programs was evaluated by simulating Reactor Trip, Turbine Trip and the IRV sizing design basis transient. (Turbine trip with non availability of atmospheric and steam discharge valves).

With the help of these analyses, NPCIL adopted the VBPP similar to 220MWe PHWR. This pressure program adopted for TAPS 3 and 4 is working smoothly since the commissioning of both the units.

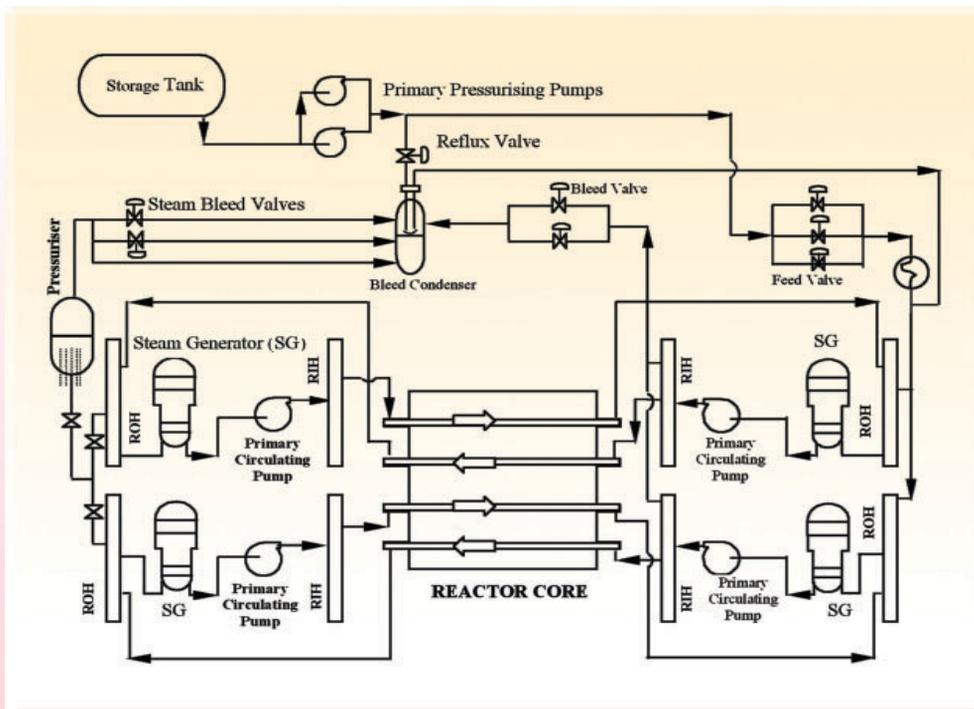


Fig. 1: Some of the major components simulated by DYNA540 for 540 MWe PHWR

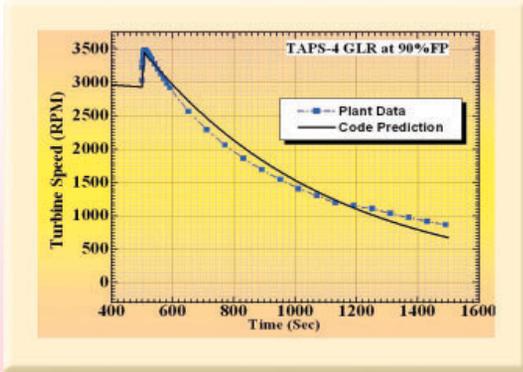


Fig. 2: Validation of code, load rejection test

structures was carried out, for Boeing 707-320 and Airbus A300, B4-200 impact with simulation of concrete cracking, crushing and reinforcement bar yielding.

It was demonstrated that OC would suffer local perforation. However, integrity of OC structure would be maintained, with negligibly small displacement at other locations. There will be local cracking and rebar yielding in IC, but there would be no perforation in IC. The total thickness of 1.36 m of OCW and ICW of TAPS-3/4 Double Wall Containment Structure would be capable of sustaining the full impulsive load of Boeing 707-320 and Air Bus A300, B4-200.

Transient studies were carried out to evaluate the capabilities of the Feed/Bleed system and the pressuriser. Optimisation studies were carried out for different surge line sizes and pressuriser liquid inventory. All the code predictions were used by the NPCIL PHT group for design verification.

### Containment Structure

Analysis and design were performed at various stages for dead loads, accident pressure and temperature loads and operating basis earthquake and safe shutdown earthquake loads: from the conceptual design stage to the final stage design of the containment structure.

#### Response to Aircraft Impact Load

As per the current international safety standard accidental impact of commercial aircraft was considered for the safety evaluation of the containment structure of 540 MWe PHWR TAPS 3 & 4 plants. Nonlinear transient dynamic analysis of the Outer Containment (OC) and Inner Containment (IC)



Fig. 3: Scenario during the Postulated Accidental Aircraft impact on containment structure of Tarapur 3 & 4 Plant Containment Structures

#### Integrated Proof and Leakage Rate Testing of Containment Structures

Surface Mounted Electrical Resistance (SMER) strain gauges were installed on the concrete surface, embedded parts and Steam Generator opening metallic dish end, in addition to the large number of embedded type Vibratory Wire Strain Gauges (VWSG) and surface mounted dial gauges. With detailed quality assurance

programme, the performance of the various sensors were satisfactory and the subsequent data analysis showed that the containment structure response was elastic during the loading and unloading stages and the test results were found to be in reasonable agreement with the in-house finite element analysis.

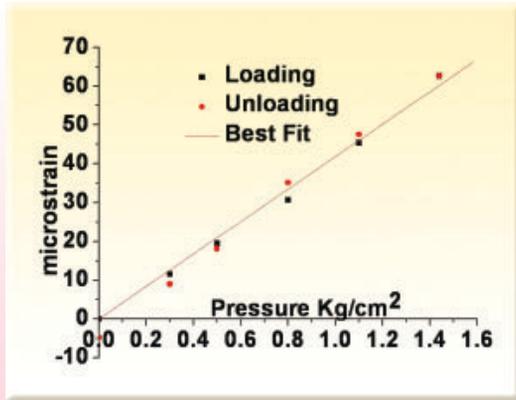


Fig. 4a: Strain vs Pressure for SMER installed on concrete surface

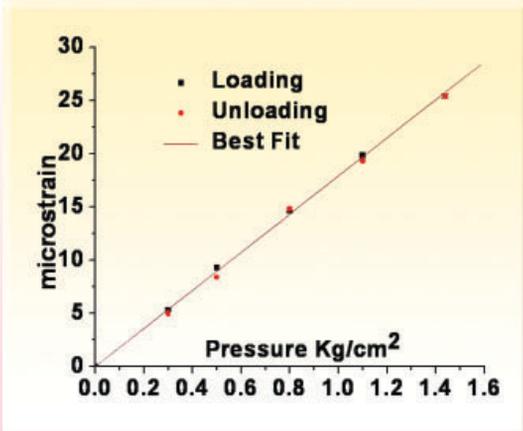


Fig. 4b: Strain vs Pressure for SMER installed on metallic surface

Ultimate Load Capacity Evaluation of 540 MWe PHWR Tarapur Containment Structure for Beyond Design Over-Pressurization.

A detailed finite element analysis was carried out by an in-house code. The conclusions made from the analysis are:

Location of the Gauges along with Elevation	Predicted Displacement in mm (FE Analysis)	Experimental displacement in mm
DT - 21 91 (E)	1.60	1.90
DT - 12 97(S)	2.33	2.93
DT - 03 104(N)	2.98	3.45
DT - 25 119 (E)	2.23	2.45
DT - 26 127 (E)	2.2	3.49
DT - 17 137(S)	-0.33	-0.37
DT - 28 139(E),VERT	2.23	2.48

Comparison of Analytical and Experimental Results upto Proof Test

- The initiation of cracking is observed at the pressure of 2.016 kg/cm<sup>2</sup>.
- The through thickness cracking of the inner containment is observed at the pressure of 2.30 kg/cm<sup>2</sup>.
- Reinforcement yielding is observed at the pressure of 3.70 kg/cm<sup>2</sup>.
- The ultimate pressure based on the limiting plastic strain criteria in reinforcement is 3.75 kg/cm<sup>2</sup> with a safety factor of 2.6 over design pressure of 1.44 Kg/cm<sup>2</sup>.

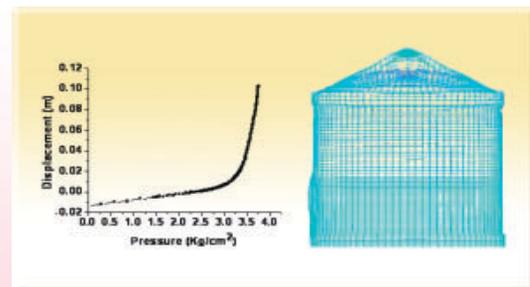


Fig. 5: Failure mode and deformed shape of Inner Containment Structure at the Ultimate Load of 3.75 Kg/cm<sup>2</sup> (2.6 Times Design Pressure)

### Seismic Analysis

During the above mentioned rigorous design exercise, a methodology to convert a complex internal structure of reactor building to axi-symmetric structure, was

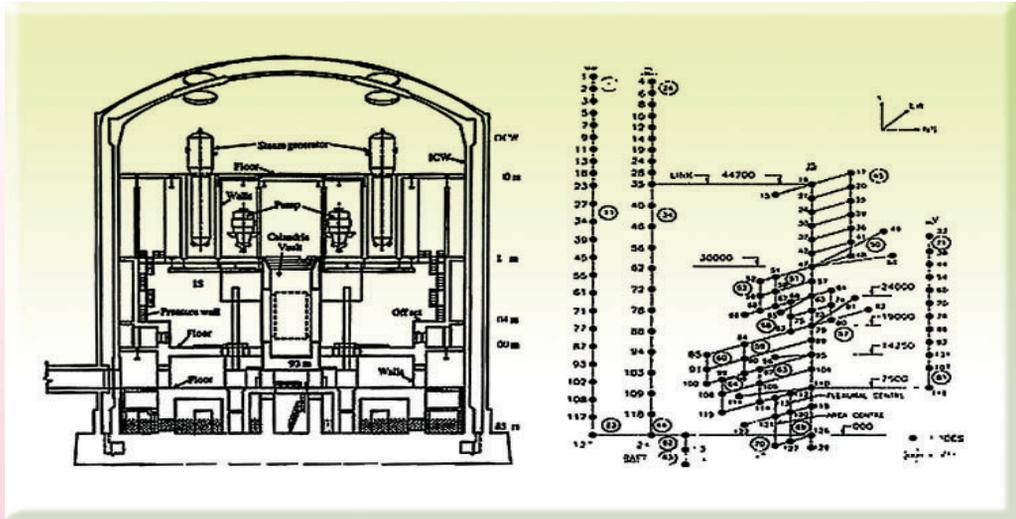


Fig. 6: Reactor building and 3D Beam Model

developed. Also, an efficient methodology was developed, based on the energy equivalence approach of making 3D beam model as shown in Fig. 6, which accounts for stiffness variation along the plan and elevation and mass distribution. In this method, the structure is modeled using 3D beam properties, based on strain energy equivalence between 3D finite element model and 3D beam model. Lateral torsional coupling and the effect of flexibility of floors, offset, partial support of walls is accounted for. In this method, the beams are located at shear center. Also developed are decoupling criteria for many structures, supported on common foundation and one of the structures also supports large size equipment.

These techniques were used in the design of containment structures and to develop floor response spectrum, that was used for the various systems and components.

### Equipment and Piping Systems

Calandria end-shield assembly and

other safety related systems, are qualified for dead loads, hydrostatic loads and earthquake loads. It was difficult to qualify with the conventional procedure of qualifying the equipment using floor response spectrum. To overcome this problem and to use the high energy absorbing capability (i.e higher damping) of vault, on which the calandria end-shield is supported, a coupled analysis was performed, using Finite Element Model as shown in Fig. 7 and thus quality the system, for earthquake loading.

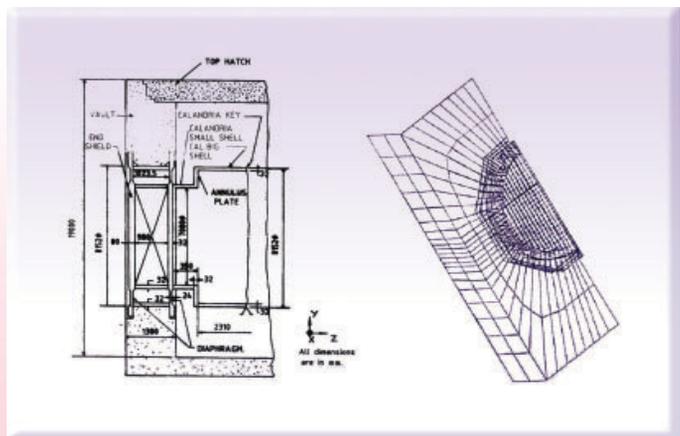


Fig. 7: Calandria End-shield and 3D Finite Element Model

The Primary Heat Transport system is the most important safety system and is qualified for dead load, pressure, temperature and earthquake loading. The key objective in this analysis, was to avoid Snubbers as far as possible, to eliminate the problems associated with the functioning of Snubbers. To improve the fatigue life of the system, the operating stresses were limited to 33% more than the yield value.

Since the Fuelling machine becomes part of the pressure boundary during the refueling process, analysis was carried out for both clamped and unclamped conditions. It was found that, with the clamped condition, the loads and stresses on the tie rod and coolant channel, exceeded the design limits. To make the system safe, lead extrusion dampers were designed and suggested to support the F/M head on LED, to reduce the loads on the coolant channel and tie rod.

Stress analysis of spent fuel transfer system (SFTS) was required to be performed, in order to qualify the design as per ASME Section III, Division 1. The transfer magazine and SFTS work together, to receive spent fuel from fuelling machine and to transfer the fuel

bundle to storage bay. The SFTS consists of top and bottom housing. Based on the stress/seismic analysis of SFTS, some design modifications were suggested. The stresses evaluated using finite element method satisfied the limits laid down by the ASME code.

### Reliability Analysis : Off-Site Electrical Power Supply

Class IV power supply system is a support system. This is required for the normal operation of the power plant. Any failure/interruption in class IV power, results in a reactor trip and activates demands on the safety systems, which include emergency power supply system as well. Two alternate class IV power supply schemes, for the 540 MW(e) Indian Pressurised Heavy Water Reactor (PHWR) were examined, to suggest the one having a low failure frequency and unavailability.

Class IV power is supplied through two sources (i) the grid through start up transformer (Fig. 8) and (ii) the station generator through unit transformer (Fig. 9). The start-up transformers are envisaged to receive the supply from 400 KV grid, through

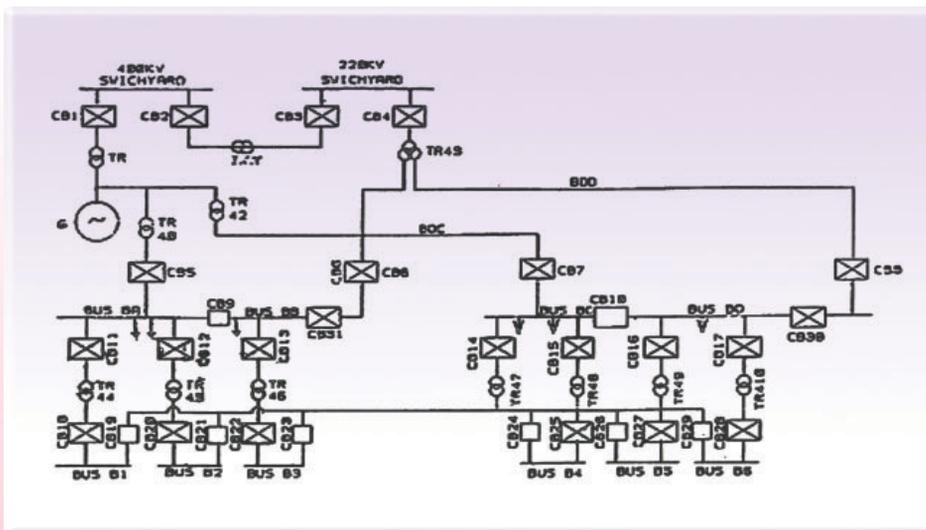


Fig. 8: Line Diagram for 540 MWe PHWR Class IV Power Supply System – Scheme I

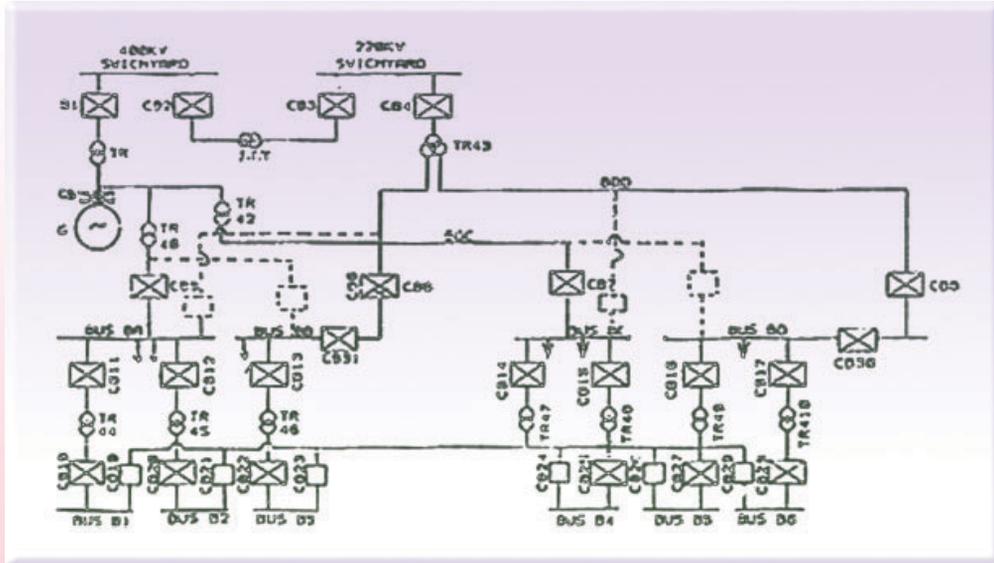


Fig. 9: Line Diagram for 540 MWe PHWR Class IV Power Supply System – Scheme II

interconnecting transformer, whenever there is 220 KV supply failure in scheme I for the 540 MWe unit. In scheme II, unit transformers receive the supply from 400 KV grid through generator transformers, whenever station generation is affected.

For quantifying the unavailability of the fault trees developed, PSAPACK, an IAEA software package was employed. For the system failure rate evaluations, a computer program MARKOV was developed. Since there is one more source of power (400 KV supply) available to the station and the station can operate on the house load for sufficiently long time, the grid station interaction factor (fraction of the occasions one failure leads to the other) is likely to be low. However, a sensitivity analysis of the Class IV supply failure frequency ( $\lambda$ ) to the grid station interaction (GSI) factor, was also carried out.

The unavailability of total Class IV are evaluated to be  $8.69 \times 10^{-3}$  and  $5.08 \times 10^{-5}$  for schemes I and II respectively. Between the two

schemes, scheme II is found to be better than scheme I. The total class IV failure frequencies, also indicate the same trend. In both the cases, the system failure frequency has a value less than 0.1 per year.

System Failure Frequencies			
Sl.No.	GSI Factor	$\lambda$ (per year)	
		Scheme I	Scheme II
1.	0.01	0.086	0.023
2.	0.1	0.085	0.023
3.	0.2	0.084	0.023
4.	0.3	0.0831	0.0235
5.	0.4	0.0822	0.0236
6.	0.5	0.0814	0.0237

### Leak Before Break (LBB)

LBB analyses involving three levels, was conducted for TAPS 3 & 4, to eliminate pipe whip restraints and jet impingement shields etc., which can impede accessibility to pipes and increase radiation exposure during maintenance operations and in-service inspections. The complete assessment, calls for generation of complete material database, fatigue analysis (level-2), leak rate and fracture analysis (level-3) and corresponding component tests for experimental verification.

#### *Material Database/Property Generation: Part of Level 1 LBB*

In level-1, it is ensured that materials are ductile, components have adequate design margins and there are no objectionable flaws. The Primary Heat Transport (PHT) system of TAPS 3 & 4 is primarily made of C-Mn steel, one of the important exceptions being, pressure tubes, which are made of zirconium alloy (Zr-2 1/2 Nb). To carry out rigorous fatigue/fracture assessments, exhaustive material testing on specimens was conducted, to generate the complete material property database such as tensile properties/stress strain curves, low cycle fatigue curves, cyclic stress strain curves, fatigue crack growth rate constants, monotonic and cyclic fracture properties (i.e., J-resistance curves), impact toughness, damage model constants etc. In all the cases, specimen tests were conducted for different product forms such as pipes, elbows and reactor inlet/outlet header forgings. Further specimens were machined from base material of respective products, weld material, composite zones and Heat Affected Zone (HAZ). The tests were conducted in different orientations, that is, longitudinal and circumferential directions and different heats (or mill lots) of material. Over and above this, a wide range of temperature was covered, viz., room temperature ( $\gg 30\text{ }^{\circ}\text{C}$ ) to maximum operating temperature ( $= 288\text{ }^{\circ}\text{C}$ ) for all the types of tests. It was also observed, that like any typical C-Mn steel, the TAPS 3 & 4 grade steel,

also undergoes Dynamic Strain Ageing (DSA). The DSA effects are most pronounced at  $250^{\circ}\text{C}$ , however, in case of TAPS 3 & 4 the degree of loss of toughness is not very significant, to warrant any undue concern. These properties were used for analytical assessment of LBB applicability.

#### *Fatigue Analysis: Level 2 LBB*

In level 2, fatigue analyses are done to ensure that postulated cracks would not grow through the thickness during the life time of the component. In addition to this, the tendency of growth should be more towards thickness. In the TAPS 3 & 4 Primary Heat Transport (PHT) piping system, all the potential locations were identified, for rigorous fatigue/fracture assessment. These locations are pipe welds and elbow/pipe bend flanks of Steam Generator Inlet (SGI) pipe, Steam Generator Outlet (SGO) pipe and Pump Discharge Line (PDL) pipe. These piping sub-systems are shown schematically in Fig. 8. At all these locations, part through thickness crack was postulated and fatigue crack growth calculations were performed, for all the significant transients. It was concluded, that total crack growth, under all the transients and transient combinations, is very small. There is significant margin, both in terms of crack length and number of cycles, against through thickness cracking.



Fig. 8: Schematic of PHT System: TAPS 3 & 4

### Fracture Analysis: Level 3 LBB

In level-3, as a worst-case assumption, it is postulated that a through-wall crack exists with maximum credible size such that, flow-through can be detected, using leakage sensors (3 different) under normal operating loads (leakage size crack). In case of TAPS 3 & 4 the leakage sensors identified are tritium activity monitor, beetle alarms and sump level monitors. It is then desired, that this through-wall crack will remain stable at normal operating plus safe shutdown earthquake (N+SSE) loads. Throughwall crack was postulated, at all the potential locations. At each of these potential locations, Leakage Size Crack (which is treated as reference crack size) was determined, based on series of thermal hydraulic analyses. The Leakage Size Crack is a through wall crack, of a size such that, its opening under normal operating loads, would result in a leakage, which is at least 10 times higher than the detection threshold of leakage sensors. The factor of 10 ensures that confidence level in detection is high.

The components made with material of high toughness with a flaw, can fail either by plastic collapse or by ductile fracture and call for detailed assessments of these failure modes. The ductile fracture assessment and collapse load assessment was done, using detailed finite element method as well as using various popular models, which are either based on J-Tearing or Net Section Collapse or their combinations. The J-Tearing (J-T) based methods used were: GE-EPRI, LBB-NRC, LBB.BCL1 and LBB.BCL2. The Net Section Collapse (NSC) based methods used were: MPA-Limit Moment, Modified Limit Load etc. The hybrid method which combines the fracture assessment and collapse load methods are R-6 and ASME Section XI Z-Factor method. The most popular method is the R6 method. It analyses a cracked component on a "Failure assessment diagram" (FAD). The method is highly graphical. A computer program called "BARC-R6" was developed in RSD, BARC. The summary of results, in terms of unstable fracture load factor, is shown in Fig.

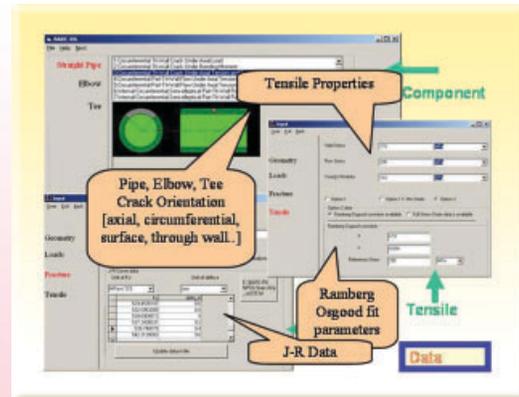


Fig. 9a: Typical input screens of BARC-R6 Software

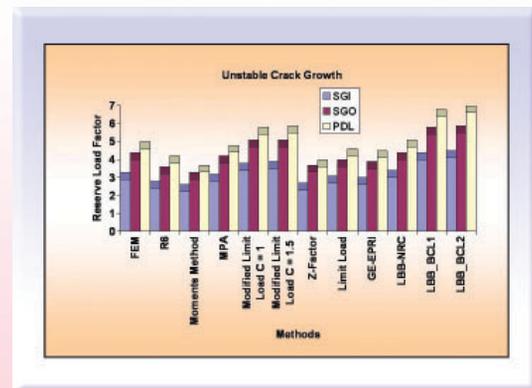


Fig. 9b: Comparing margin on load using different method

9b. It was concluded that, at all the locations, the requisite margins are available.

Certain developments were done, to enhance the applicability of these methods. One of them is the development of Z-factors for elbows. The Z-factors as presented in ASME Boiler and Pressure Vessel Code, are applicable to straight pipes only. With this development, it is also possible to qualify the elbows too. In addition to this system level analyses were performed, taking into account, the piping compliance. It was concluded, that the actual margins are about 20 to 30% more than those reported earlier.

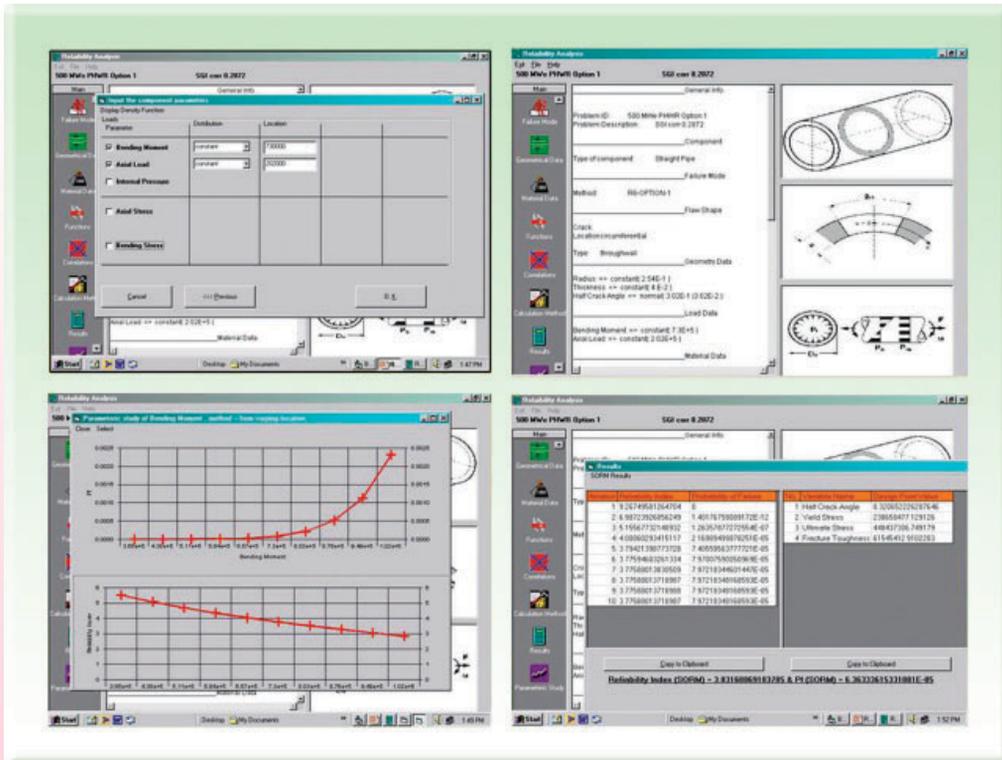


Fig. 9c: Typical screens of a BARC-RAS software

**Probabilistic LBB Analysis**

The scatter in input parameters of LBB, can be in material properties used for analysis, the leakage size crack calculated from thermal hydraulic studies and the accident loads during earthquake. Analysis was done to estimate the probability of not meeting the level 2 and 3 LBB. The probability of failure of structure components, designed and fabricated as per high standards is usually very low. A typical value of failure rate is less than  $10^{-7}$  per year. These kinds of numbers cannot be established using repetitive testing of components. The estimation through the route of structural reliability methods is possible. In this route, the failure probability is evaluated from mechanistic models of fatigue and fracture. There are various methods of assessing the probability of failure. The popular ones are Monte Carlo Method, Monte Carlo

Method with sampling and First and Second Order Reliability Method (FORM, SORM). Computer programs were developed in RSD, BARC to perform such reliability analysis. Program BARC-PRAISE (Piping Reliability Analysis Including Seismic Events) analyses the probability of failure of piping with surface cracks, using Monte Carlo Method. This program can be used for assessing the effect of alternate inspection schedules on the reliability of piping. This study was taken up for TAPS 3 & 4 piping also. It can model the effect of hydro-test, inspections and uncertain arrival rates of transients. BARC-RAS (Reliability Assessment Software) can analyze the cracked piping components, using a variety of probability assessment methods and for different modes of failures. For TAPS 3 & 4 primary piping, the probabilities of not meeting the LBB numbers are small. Typical values of the equivalent safety factor, based on probabilistic analysis are listed below:

	NSC		R6 rupture		J-T	
	D	P	D	P	D	P
SGI	2.65	3.01	2.40	2.53	2.60	2.53
SGO	3.63	4.20	3.20	3.41	3.47	3.93
PDL	4.15	4.91	3.82	4.12	4.13	4.89

Comparing the margins on Level 3, LBB  
(D = deterministic margins, P = probabilistic margins)

### Experimental Verification of Leak Rates

A facility (Fig.10) was created, through a BRNS project, for measurement of leakage through crack pipes. Its maximum operating pressure is 100 bars and temperature is 250 °C. Several experiments were conducted to verify methods of leak rate calculations.

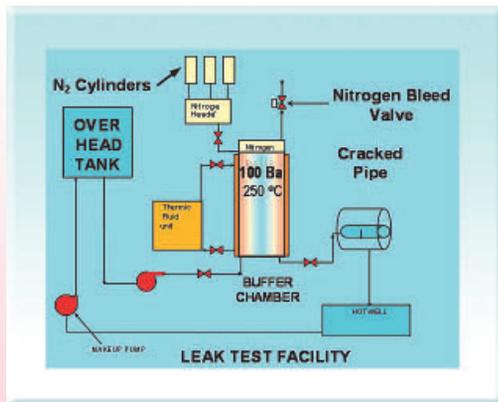


Fig. 10: Pipe Leak Test Facility

### Experimental Verification: Fatigue (Level 2 LBB)

Fatigue crack initiation and crack growth tests were conducted on 45 C-Mn pipes and elbows. In these tests, part-through thickness notches were machined in circumferential direction (for pipes) and in circumferential or axial direction for elbows. A typical test setup is shown in Fig. 11a.

The salient observations are:

- The crack growth in depth direction is much faster than in surface direction. This is depicted in



Fig. 11a: Elbow Fatigue Test Set-up

Fig. 11b, which shows that for large variation in initial flaw aspect ratio (crack length/crack depth,  $2C/a$ ) the final aspect ratio is around 4. This is one of the important requirements of level 2 LBB qualification.

- The number of cycles leading to through wall crack penetration, are significantly higher than the number of anticipated transients of significant stress range.
- The transferability of fatigue crack growth rate data, from specimen to component, is not a major issue, (Fig.11c).

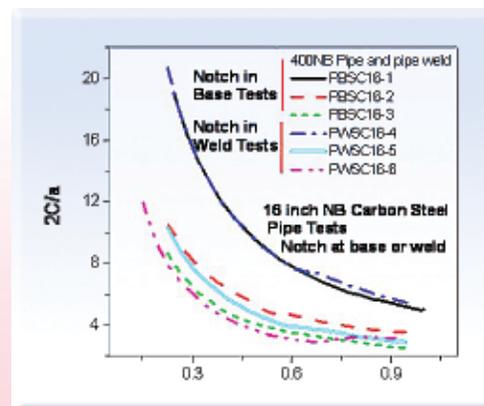


Fig. 11b: Crack aspect ratio variation

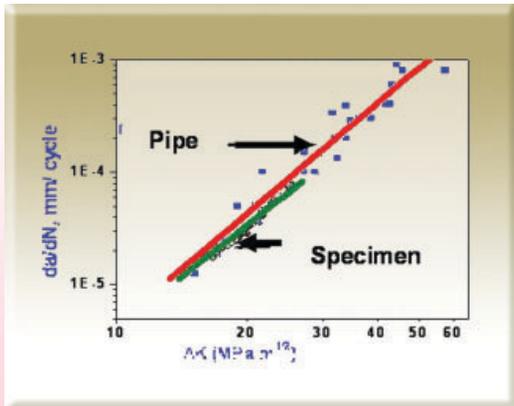


Fig. 11c: Pipe v/s specimen fatigue crack growth rate

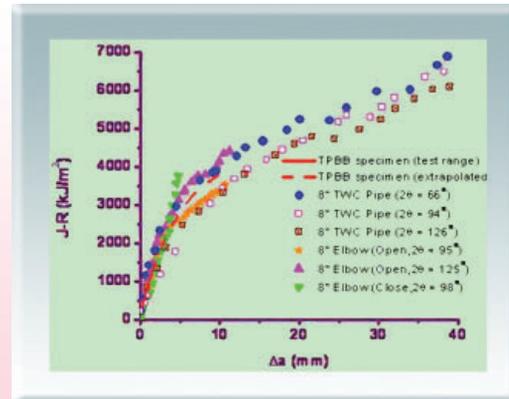


Fig. 12b: Comparison of component and specimen J-R curves

#### Experimental Validation: Fracture (Level 3 LBB)

Sixty fracture tests were conducted, on pipes and elbows to validate the fracture assessment procedure and resolve the important issue of transferability of fracture data, from specimen to components. The validation of fracture calculation procedure vis-à-vis experimental results is highlighted, for a typical case in Fig.12a. A new parameter " $A_{nq}$ " was proposed, which is the ratio of actual triaxility quotient to critical triaxility quotient. It was established that, if  $A_{nq}$  of specimen and component are the same then specimen and component J-R curves would also be the same. (Fig. 12b). Such studies, helped in enhancing the confidence level of analytical LBB assessment results.

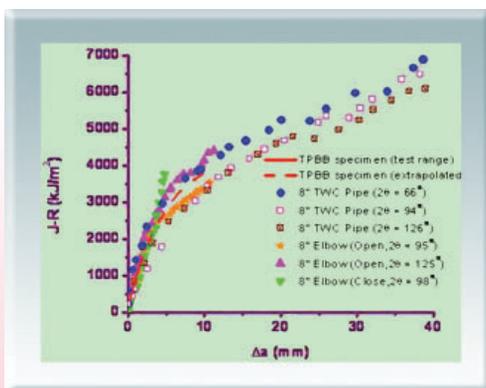


Fig. 12a: Experimental validation of pipe fracture evaluation

#### New Developments in Fracture: Cyclic Tearing

In Level-3 LBB, the safety margin is ensured against peak dynamic load and number of cycles of load application was generally not considered in the past. Some of the most referred IAEA documents or USNRC guides, on LBB, are also silent about cyclic fracture aspects under earthquake. In India, the earthquake load is one of the main design basis accident loads. The reversible cyclic loading, during the earthquake, has significant impact on fracture behavior. Experimental and analytical investigations on cyclic tearing for TAPS 3 & 4, were carried out on full scale (8, 12 and 16 inch NPS), cracked straight pipes made of C-Mn steel and subjected to cyclic loadings, as shown in Fig. 13a.

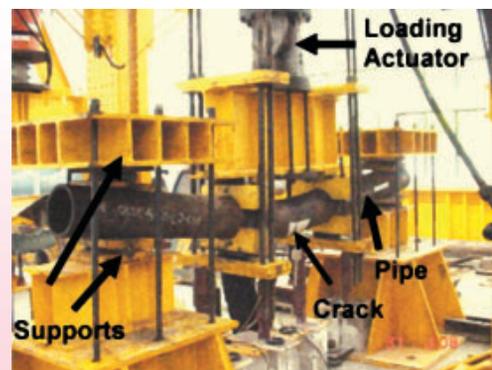


Fig. 13a: Cyclic tearing test set-up picture

The cyclic test results have been compared with the corresponding monotonic pipe fracture test results and shown in Figs. 13b and 13c. These figures show, that a significant drop occurs in energy absorbing capacity or fracture resistance of cracked pipes, under cyclic loads. A simplified master curve (Fig. 13d) has been generated directly from the pipe cyclic tearing and monotonic fracture experiment results. Based on these investigations, additional safety margins were suggested for accounting the damage due to reversible cyclic loading. Following are modified margins critical crack size and on critical load

$$M_{NOC+SSE} \times \beta_c < M_{crit} \text{ (at } 2 \times LSC)$$

and

$$M_{NOC+SSE} \times \sqrt{2} \times \beta_c < M_{crit} \text{ (at } LSC)$$

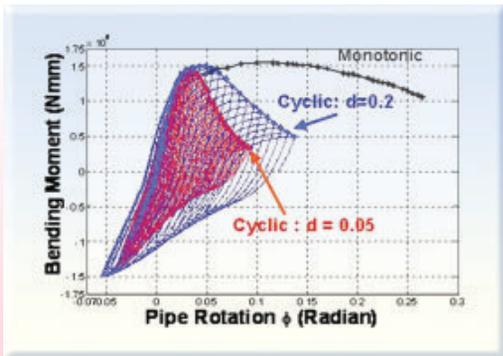


Fig. 13b: Moment-Rotations

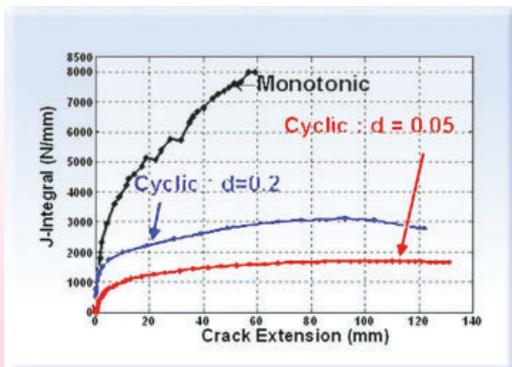


Fig. 13c: J-R curves -monotonic and cyclic tests

Where, LSC is leakage size crack and  $\beta_c = 4/3$  (for SSE) and  $3/2$  (for OBE) loading. Basis of additional margin is derived from Fig. 13d.

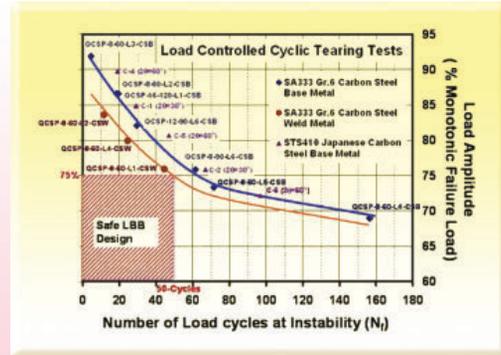


Fig. 13d: Master curve for cyclic tearing

### Integrity of Pressurized Piping Components under Seismic Loading: Fatigue-Ratcheting

The safety and integrity of PHT piping system, is of particular concern, especially when they are subjected to high strain, low cycle fatigue conditions as may be induced during earthquake. Here, the pressurized piping may fail due to excessive accumulation of plastic strain by ratchet action in addition to the low cycle fatigue damage. To ensure the necessary margins against ratcheting failure for TAPS 3 & 4 piping, ratcheting experimental investigations have been carried out on several pressurized elbows made of C-Mn steel.

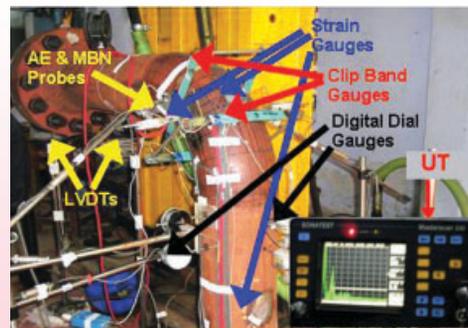


Fig.14a: Fatigue ratcheting test set-up picture

The test setup is shown in Fig. 14a and failure in Fig. 14b. It was observed that diametral growth of elbow occurs followed by crack initiation on inner surface of crown, which finally grows through thickness leading to complete failure.



Fig. 14b: Failed elbow pictures

It was concluded that in case of TAPS 3 & 4, for the worst pressure and seismic bending moment, failure by fatigue ratcheting is ruled out. It proves that seismic qualification of its pressurized primary components, based on plastic collapse as primary mode of failure, is valid.

### Structural Assessment of Primary Components:

Detailed stress analyses were done for several components of TAPS 3 & 4, for design qualification or optimization. Some of them are listed below:

- (i) Detailed stress analysis of End-Shield was carried out, to compute stresses under dead weight and thermal loads. During investigations, several design changes were recommended such as minimizing of extra thickness in bottom region, to reduce thermal stresses without affecting load carrying capacity under mechanical loads. These recommendations were finally implemented.
- (ii) A detailed program was pursued to optimize the dimensions of the

540 MWe seal disc because of its larger diameter and larger coolant pressure as compared to 235 MWe reactor seal disk. During the design, it was found that for two stages of installation, the stresses were more than the ASME limits. Hence, limit analysis were carried out for ASME qualification of the seal disc. Fig.15a shows the deformed shape of the seal disc, during collapse, to find out the limit load.

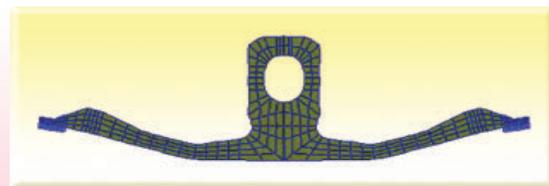


Fig. 15a Limit load analysis of seal analysis

- (iii) Stress analysis of different components i.e. pressure housing, End-cover, Gray lock coupling (Fig. 15b) of Fuelling Machine Head for design check and optimization. Three-Dimensional finite

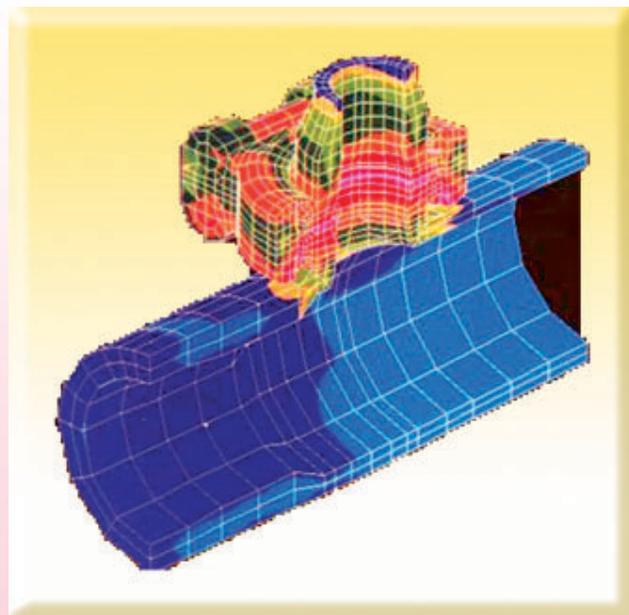


Fig. 15b: Stress analysis of Graylock Coupling

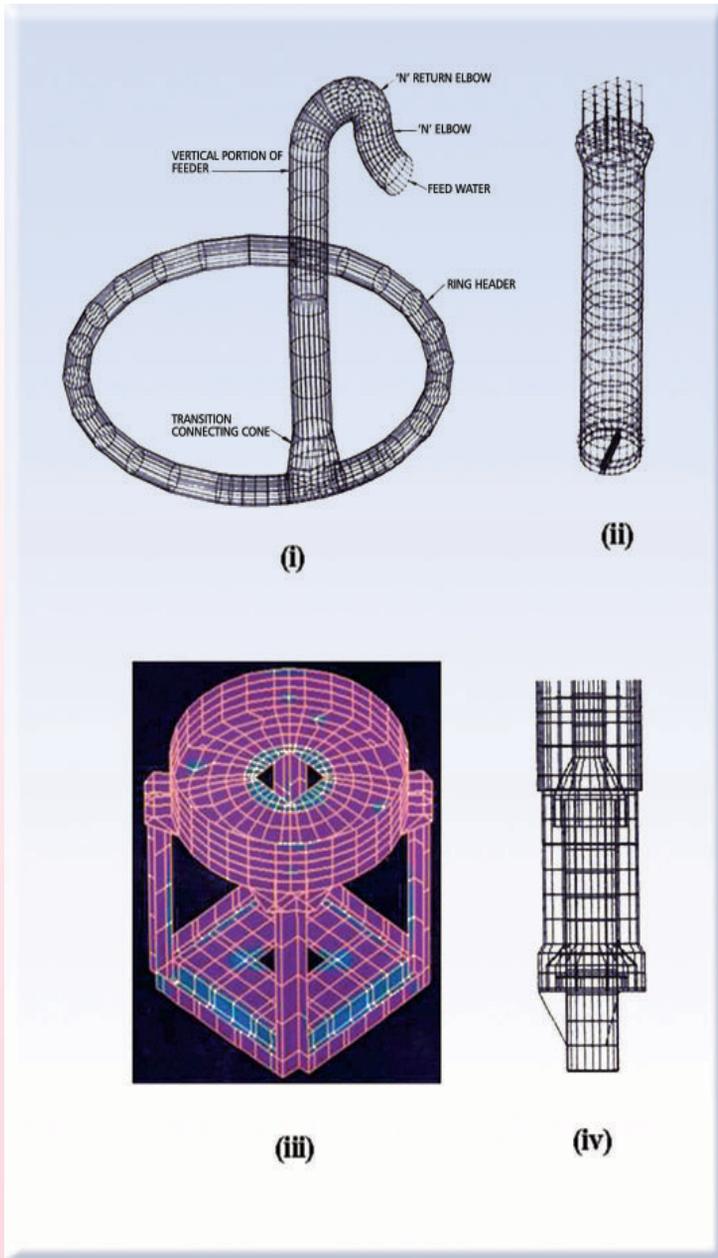


Fig. 15c: Finite Element Analysis Models for different internal components of TAPS 3&4 steam generator:  
(i) Feed Water Inlet Ring Header,  
(ii) Shroud Vessel with Deck Plate and Steam Separator,  
(iii) Steam Drier Assembly and  
(iv) Steam Separator Assembly

element models were prepared, for all the components separately and analyses were carried out, for different stages of loadings. Stresses were found to be within ASME limits.

- (iv) Stress analysis of Steam Generator Internals (Fig. 15c) under postulated Feed Water or Main Steam Line Break, to ensure integrity under severe loads.
- (v) Flow induced vibration and seismic assessment of Liquid Poison Injection tubes and Shutoff Rod Guide Tubes.
- (vi) Experimental residual stress measurements during inhouse (by Reactor Engineering Division) development of zero clearance rolled joints of pressure tubes.
- (vii) Structural assessment of calandria under postulated simultaneous pressure tube-calandria tube rupture. The analysis included, the pressure wave analysis to generate load time histories.
- (viii) Studies on sloshing of moderator, in calandria, during seismic events. It helped in generating sloshing loads on calandria internals.

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



Swati Kota

श्रीमती स्वाती कोटा, आणविक जैव प्रभाग को जनवरी 8-10, 2006 के दौरान ओस्मानिया विश्वविद्यालय, हैदराबाद के जनन विज्ञान विभाग में आयोजित फ्रंटियर्स इन जनिटिक्स एन्ड बयोटेक्नालोजी की अंतर्राष्ट्रीय परिचर्चा में “आइडेंटिफिकेशन ऑफ ए मल्टिप्रोटीन डीएनए मेटाबोलिक कॉम्प्लेक्स फ्रॉम ए

रेडियोरेजिसटंट बैक्टीरियम डीनोकोकस रेडियोडूरस” नामक शोध पत्र के लिए सर्वश्रेष्ठ पोस्टर पुरस्कार से सम्मानित किया गया।

Ms Swati Kota of Molecular Biology Division was awarded the Best Poster Prize for her paper “Identification of a multiprotein DNA metabolic complex from a radioresistant bacterium *Deinococcus radiodurans*” at the International Symposium on Frontiers in genetics and Biotechnology; retrospect and prospect”, held at the Dept. of Genetics, Osmania University, Hyderabad, from Jan. 8 -10, 2006.



Bhakti Basu

श्रीमती भक्ति बासु, आणविक जैव प्रभाग को जनवरी 17-20, 2006 के दौरान इंडस्ट्रियल -टोक्सिकोलोजी रिसर्च सेंटर, लखनऊ, में आयोजित डीनोकोकस रेडियोडूरस जीन टु जीनोम, एनविरोनमेंटल एन्ड केमिकल इंटरैक्शन की 29वीं अखिल भारतीय सभा एवं गोष्ठी में “प्रोटीन रिसाइक्लिंग इन दि पोस्ट

इरेडियेशन रिकवरी ऑफ डीनोकोकस रेडियोडूरस”, नामक शोध

पत्र की सर्वश्रेष्ठ प्रस्तुति के लिए प्रोफेसर वी.सी. शाह पुरस्कार से सम्मानित किया गया।

Ms Bhakti Basu of Molecular Biology Division was awarded the Prof. V.C. Shah prize for Best Paper presentation for her paper “Protein recycling in the post irradiation recovery of *Deinococcus radiodurans*”, at the 29<sup>th</sup> All India Cell Biology Conference and Symposium on *Deinococcus radiodurans*. Gene to Genome, Environmental and Chemical Interaction, held at the Industrial Toxicology Research Centre, Lucknow, between Jan. 17-20, 2006.



Roychowdhury

श्री एस. रॉयचौधरी, पदार्थ विज्ञान प्रभाग को “एनविरोनमेंटल इफेक्ट्स ऑन दि फ्रैक्चर टफनेस ऑफ स्टेनलेस स्टीलस” नामक सर्वश्रेष्ठ एम. टेक शोध पत्र के लिए इलेवंथ एन.ए.सी.ई (NACE) इंटरनैशनल इन्डिया सेक्शनस कोरोजन एवियरनेस पुरस्कार से

सम्मानित किया गया। यह पुरस्कार ओ एन जी सी के द्वारा संस्थापित किया गया था। श्री रॉयचौधरी ने जुलाई 2005 में आईआईटी मुंबई से एम टेक की पदवी प्राप्त की जिसमें इन्होंने पदार्थ विज्ञान एवं इंजीनियरिंग में विशेषज्ञता प्राप्त की।

Mr S. Roychowdhury for Materials Science Division was awarded the 11<sup>th</sup> NACE International India Section’s corrosion awareness award for the “Best M.Tech. Thesis”, entitled, “Environmental effects on the fracture toughness of stainless steels”. The award was instituted by ONGC. Mr. Roychowdhury completed his M.Tech. in July 2005, from IIT, Mumbai with specialization in materials science and engineering.



Edited & Published by : Dr. Vijai Kumar, Associate Director,  
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