सत्यमेव जयते

# Fast Breeder Test Reactor



October 2010

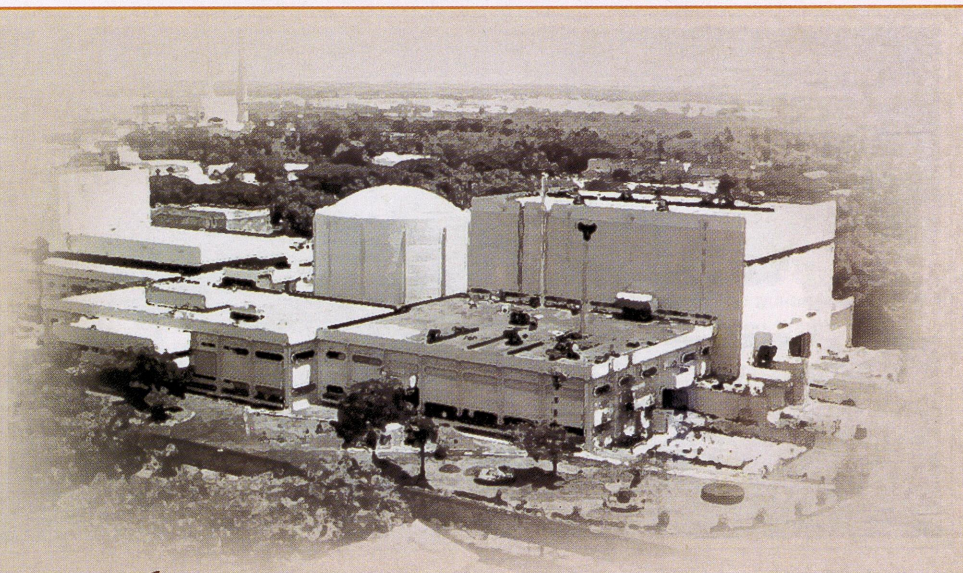
## *An Exciting Journey through a New Technology*

25 years
Since 1985
Celebration

Government of India
Department of Atomic Energy
INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH
Kalpakkam - 603 102

सत्यमेव जयते

# Fast Breeder Test Reactor



*An Exciting Journey through a New Technology*

25 years
Since 1985
Celebration

Government of India
Department of Atomic Energy
INDIRA GANDHI CENTRE FOR ATOMIC RESEARCH
Kalpakkam - 603 102



# *Fast Breeder Reactors*
# *Towards Energy*
# *Security for the Nation*

डॉ. श्रीकुमार बॅनर्जी
Dr. Srikumar Banerjee

भारत सरकार
Government of India

अध्यक्ष, परमाणु ऊर्जा आयोग
व
सचिव, परमाणु ऊर्जा विभाग
Chairman, Atomic Energy Commission
&
Secretary, Department of Atomic Energy

August 24, 2010.

MESSAGE

Twenty five years of successful and purposeful operation of Fast Breeder Test Reactor and Radiometallurgy Laboratory is a significant mega stone in the annals of fast breeder and closed fuel cycle technologies not only in India but indeed the world.

Fast reactors are essential for the energy security of our country. FBTR was built in the seventies with an indigenous component of more than 80% and stands out as a symbol of our national pride. FBTR uses a unique U-Pu carbide fuel with high Pu content. The excellent performance of this novel fuel has won acclaims of the global fast reactor community. The safe and successful operation of FBTR has provided the necessary inputs for the design of PFBR, 500 MW(e) and was one of the major factors which gave us the confidence for launching its construction. It is heartening to note that the fuel cycle of the unique carbide fuel of FBTR has been closed with success. It is also a good augury that the MOX fuel of PFBR composition has been irradiated to its target burn-up of 100 GWd/t without failure of any of the pin in the sub assembly. Both these milestones signify that fast reactor technology has come of age in India. With an estimated residual life of ten Effective Full Power (EFP) years, I am sure that FBTR will continue to contribute to testing of metallic fuels and advanced structural materials for the next generation of safe and economic fast reactors, build experience and more important; human resources in this important technology.

The Radiometallurgy Laboratory (RML) of IGCAR has played a commendable companion role for FBTR in conducting the stage-wise post-irradiation examination of the fuel and providing the inputs to the fuel designers & manufacturing personnel in modeling the fuel behaviour, tuning the achievable specifications and progressively enhancing its burn-up to approx. 160,000 MWd/tonne. The hot cells of RML are unique since they handle fuel pins and subassemblies with very high burn-up values under inert atmosphere. The handling of the carbide fuel pellets and pins covered with unavoidable sodium demands high standards of safety. RML has accomplished these precise and high technology assignments with dexterity and confidence.

It is essential that the knowledge gained over the years, especially in an evolving technology, is enhanced and further strengthened. The initiative taken by IGCAR in knowledge management in the form of books on FBTR and RML is commendable.

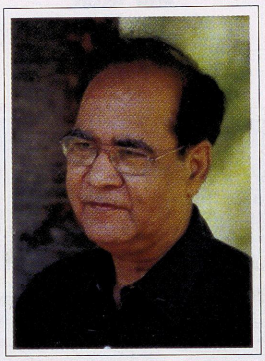
I wish FBTR and RML a safe and exciting journey ahead with many more milestones and laurels to be achieved, in future. I am sure that the experience gained in the operation of FBTR and RML will provide a strong foundation for the success of our fast reactor programme.

Srikumar Banerjee

(Srikumar Banerjee)

अणुशक्तिभवन, छत्रपति शिवाजी महाराज मार्ग, मुंबई - 400 001, भारत
दूरभाष : +(91) (22) 2202 2543 • फैक्स : +(91) (22) 2204 8476 / 2284 3888 • तार: एटमर्ग
Anushakti Bhavan, Chhatrapati Shivaji Maharaj Marg, Mumbai - 400 001, India
Phone: +(91) (22) 2202 2543 • Fax: +(91) (22) 2204 8476 / 2284 3888 • Grams: ATOMERG
E-mail: chmn@dae.gov.in

# Foreword



Twenty five years of safe and successful operation of the Fast Breeder Test Reactor (FBTR), a reactor which is the first of its kind in India, is a joyous occasion deserving commemorating, more so considering that this unique reactor has an equally long innings for fulfilling the intended missions. When FBTR was being built, there were no formal regulatory mechanisms. FBTR was started as a self-regulated project. The legacy left behind by the project leaders of FBTR has been inherited and improved by the Operation and Maintenance personnel. Safety is accorded first priority in operating FBTR. I have been a member of the IGCAR Safety committee and then a member of Safety Review Committee for Operating Plants (SARCOP). I have always appreciated the adequate care and judicious caution with which the safety issues were addressed by the team. The safe and successful operation of FBTR, the confidence with which challenges were faced and the wisdom acquired through years of painstaking experience are major inputs to the decision for launching the 500 MWe Prototype Fast Breeder Reactor. I take this opportunity to congratulate the FBTR team for taking the reactor through the various milestones and dreaming for the future.

Credit for the success of FBTR goes to the successive team leaders of FBTR who inspired confidence among the staff, trained them and honed their skills over the years. Late Shri.M.K.Ramamoorthy, the first Superintendent, led the team for commissioning the systems and was instrumental in the first criticality. He steered FBTR during the difficult days of the fuel handling incident in 1987. Shri.S.B.Bhoje, who took over from him, systematized the documentation system. It was during his tenure that the steam generators were commissioned and electricity from a Fast Reactor fed to the grid for the first time in India. Late Shri.R.P.Kapoor successfully steered to completion complex safety related experiments to prove the inherent safety of the reactor and obtained clearances for sustained operation at high power.

He was the main author of technical specification document- the rule book of operation- and played a major role as a safety committee member. He ensured that no safety relevant issue goes amiss. He was a professional with keen eyes and brain for detailing. He also had a flair for perfection in documentation, a legacy that is being continued. Shri.P.V.Ramalingam, with his initial training in CIRUS reactor at Trombay and subsequent training at Rapsodie (sodium cooled fast spectrum reactor), France, assiduously honed the operating skills of the personnel. He was instrumental in carrying out several modifications to improve the availability of the reactor and played a major role in handling the sodium leak incident in 2002. The current team under the leadership of Shri G. Srinivasan has inherited all the good legacies left behind by the earlier team leaders and is systematically making efforts to scale greater heights in terms of reactor power, operating temperatures and utilization of the reactor for experiments and tests.

I have seen the progress of FBTR all these twenty five years, in various capacities and responsibilities. The experience to achieve maturity and excellence in the route to mastering the technology is enriching and rewarding.

I am the chief architect of Radiometallurgy laboratory, designed primarily to be a robust companion of FBTR, as a Post-Irradiation Examination Laboratory (RML). The hot cells of RML have to handle subassemblies covered with sodium. The change over from oxide to the carbide fuel led to essential requirements wherein the facility was modified for operation in high purity inert gas. Fuel with high burn-up levels (~ 200 GWd/tonne) needs to be handled in the cells. All these objectives and more make the hot cells of RML a unique facility in India and indeed the world. Depending on the mission, need based dismantling, non-destructive and destructive techniques are developed and deployed. As a young science and technology manager of RML in June 1974, I dreamt of this facility as a world class one in infrastructure, techniques and technologies capable of sustaining the materials and fuels programme of fast reactors. It can safely be deduced that the facility has met this simple dream, in a comprehensive manner.

A perfect spirit of cooperation and synergy exists among the operating staff of FBTR and RML. This bond is a key to FBTR reaching many of the milestones. RML has played a pivotal role in the progressive enhancement of the unique high plutonium carbide fuel from a target burn-up of 25 GWd/t to 165 GWd/t. In recent times, RML has contributed to the life extension studies of FBTR by measuring the ductility of grid plate specimens, irradiated to low neutron doses. The Kamini reactor, located in the basement of RML, is being operated by FBTR staff for neutron radiography by RML. Perfect harmony between FBTR and RML is a pleasure to experience. The twenty five years of operation of FBTR and RML have been a period of learning and mastering of various facets of new technologies, facing challenges, setting demanding milestones and reaching those, not so easily with good measure of sweet but never wounded.

India has a rich legacy of knowledge management. The Indian ethos dictates that knowledge should have utility to mankind and every piece of knowledge should be preserved and transmitted to next generations. Before writing was invented, knowledge was transmitted by the Guru to the succession of disciples in Gurukulas; a fine tradition worth emulating even in the current context. With the invention of writing, knowledge was preserved in the form of palm leaf writings and stone inscriptions. Due to the difficulties in writing, much of the ancient wisdom was written in pithy capsule forms called `Sutras' or strings. It is in this context that I have been writing the history of each facility and programme in IGCAR in the IGC newsletters, under the category: Director's Column. We feel a need for more elaborate and useful records of the history of facilities like FBTR and RML, where knowledge has been acquired with huge investments of money and painstaking efforts, spread over a time period covering at a least three generations of professionals. The two books, being released on the occasion, fulfill the above mentioned objective. Like the sutras of yore, they describe, in a capsulated form, the long and exciting journey of FBTR and RML in pursuit of their mandates and missions. It is hoped that these two books shall serve as reference books for posterity. I compliment and appreciate the authors and the editorial committees of both the books for these publications for the benefit of the next generation of fast reactor technologists, enlightened public and policy makers.

Fast Reactors with closed fuel cycle are in the centre stage of the nuclear programmes to achieve sustainability of energy and give robustness to nuclear programmes of the world not only in the 21st century but a few centuries to come. In India, the success of three stage programme rests on the shoulders of fast reactor programme. We have met the challenges and prepared ourselves for meeting higher levels of challenges and aspirations in the coming years and decades. This celebration and commemorative function, full of eminence and wisdom, shall continue to inspire all of us and our future generations for the years to come.

I thank all of you, who have supported us in our efforts, sorrows and joys. We hope to have your confidence, in much greater measure, in the years ahead of us.

Baldev Raj

(Baldev Raj)
Director, IGCAR

*Honoured*

*and Inspired*

*by*

*their Visits.....*






*Honourable Prime Minister Dr. Manmohan Singh's visit (2004)*



*Former President Dr A. P. J. Abdul Kalam with Dr Srikumar Banerjee in RCB (2008)*



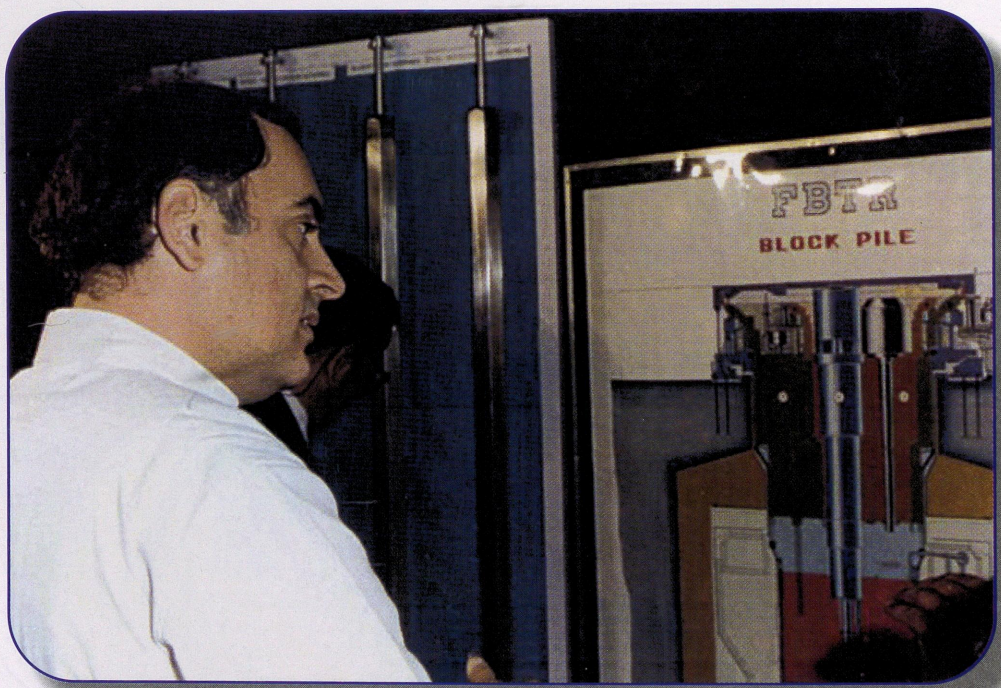
*Former President (Late) Shri. R. Venkataraman in RCB (1986)*



*Former President (Late) Dr. Giani Zail Singh being received at the portico (1983)*



*Former Prime Minister (Late) Smt. Indira Gandhi's visit (1983)*




*Former Prime Minister (Late) Shri Rajiv Gandhi inside RCB (1985)*



*Former Prime Minister (Late) Shri Rajiv Gandhi in Control Room with (Late) Shri M. K. Ramamurthy, Station Superintendent (1985)*



*Former Prime Minister Shri Atal Bihari Vajpayee in the lawns of FBTR for planting a sapling (1998)*



*Dr. Mohammed Elbaradei, Director General, IAEA visiting FBTR (2009)*




*Former Defence Minister George Fernandes at FBTR ( 2000)*



*Honourable Union Ministers Shri Pranab Mukherjee & Shri Prithviraj Chavan (2005)*




*Former Governor Ms Fathima Beevi planting a sapling (1998)*

## Editorial Committee


G. Srinivasan

K. V. Suresh Kumar

B. Anandapadmanaban

B. Babu

B. S. Panigrahi

M. S. Chandrasekar

Pankaj Mehta

M. Sai Baba

# Contents


| | |
|---|---|
| Preface | i |
| About Fast Reactors | v |
| Plant Description | 01 |
| The Project Years | 23 |
| Engineering R&D | 57 |
| Commissioning, Criticality and Start of Power Operation | 75 |
| Operational History since 1995 | 109 |
| Challenges on the way | 133 |
| Major Retrofitting and Modifications | 153 |
| Plant Life Extension | 169 |
| Carbide Fuel | 181 |
| Regulatory Experience | 193 |
| Radiological Experience | 203 |
| Summary | 213 |
| Acknowledgment | 222 |

# Preface

The indispensability of energy for the sustenance, and even survival, of mankind could not have been more cryptically expressed than by the famous statement of Dr. Homi Bhabha that 'No energy is costlier than no energy'. The visionary he was, he realised that the fossil fuel reserves within the bowels of mother earth, slowly built by nature over geological ages, would get squandered away in centuries, forcing mankind to look for alternate sources of energy. With the prospects of energy from fusion of hydrogen seemingly far away, he foresaw the assured energy from fission as an interim solution for mankind's ravenous need for power. As regards India, he recognised that our limited reserves of Uranium cannot sustain the nuclear energy programme for long and gave out the breeder route for the efficient use of Uranium. In the long run, Thorium was slated to play the major role in our nuclear programme. The Department of Atomic Energy has been actively pursuing the famous 'three stage nuclear programme' chalked out by Dr. Bhabha.

Dr. Bhabha laid the foundations for the first stage of our nuclear programme, by founding the BARC and assiduously promoting all the R&D activities for the indigenous heavy water reactor programme. When untimely death snatched away Dr. Bhabha, the mantle of pursuing Dr. Bhabha's vision fell on Dr. Vikram Sarabhai. He laid the seeds of our fast reactor programme. Realising the enormity of the R&D tasks for the fast reactor programme, he decided to set up a separate centre exclusively dedicated to the development of fast reactors in India. The Reactor Research Centre was started at Kalpakkam in 1970, with the Fast Breeder Test Reactor as its hub. The mandate of the Centre was to carry out comprehensive R&D on all aspects of sodium cooled fast reactor technology and promote the commercial deployment of fast reactors in India. RRC was renamed as Indira Gandhi Centre for Atomic Research in 1985, after FBTR went critical.

The excellent rapport between the French and Indian governments resulted in signing of an agreement between DAE and French CEA for the transfer of the design and drawings of Rapsodie reactor. Though FBTR design mostly followed the Rapsodie, some modifications were carried out vis-à-vis Rapsodie, the major being the incorporation of steam generators and a custom built turbine generator, thus making FBTR a full fledged small power plant. The French supplied the design, drawings, safety report and operational documents of the Rapsodie reactor

transferred the technology of fabrication of critical components and trained the Indian engineers in various aspects including component manufacture and reactor operation. The French side continued to fulfil all their obligations as per the agreement till the Rapsodie reactor was shutdown, and even beyond, up to 1987.

As per the Indo-French agreement, the responsibility of construction of FBTR was with India. For a first-of-its-kind reactor in India, the indigenous component of FBTR was about 80%, a fact that should make us proud, considering the technology available in the seventies. Concurrent with the construction of FBTR, Engineering Halls with sodium loops were built at IGCAR for training of personnel in the safe handling & disposal of sodium and testing of sodium instruments & control rod drive mechanisms. Water loops were put up for testing the sodium pumps. A facility was set up for purifying the commercial grade sodium to reactor grade purity before charging into the reactor systems. Hot cells and other laboratories were also taking shape.

The agreement between India and France, however, did not include the supply of fuel. With the changed international scenario after the Pokhran-I, it became clear that we had to fend for ourselves as regards the fuel. A bold decision was then taken by Dr. Raja Ramanna to go in for a plutonium rich carbide fuel. This decision, in retrospect, has proved to be a challenge converted into an opportunity. We have been able, in a single go, to perfect the technology of an advanced fast reactor fuel, including the closing of its cycle. This has put us in a good stead for managing the fuel cycle of the conventional MOX fuel being deployed for the first few commercial fast reactors, including the 500 MWe Prototype Fast Breeder Reactor.

FBTR went critical on 18$^{th}$ October 1985, putting India in the select list of countries with fast reactors. The twenty five years which followed that historic day have proved to be a period of intense learning, challenges and solutions, problems and come-backs. The reactor has completed 16 irradiation campaigns, most of them with the Turbo-Generator synchronised to the grid. Sodium has been maintained in liquid state and kept in continuous circulation with unimpaired purity. The four sodium pumps have been in continuous cumulative service exceeding 650,000 hours. The steam generators, which are critical for the success of sodium cooled fast reactors, have not had a single leak of water into sodium. Exciting safety related experiments have been carried out. Several planned and unplanned interventions have been made in the sodium systems, without any incident. The carbide fuel has performed far beyond the original expectations, reaching burn-up levels more than six times the original target burn-up, without a single pin failure. A few major and several minor challenges have been encountered, making FBTR operation an

exciting journey, but in the end, the right solutions have been found. These include a major fuel handling incident which warranted a 'surgical intervention' inside the reactor. There was another incident of a leak of 75 kg of primary sodium into an inerted cabin, but the system was normalised within two months, without any radiological consequences to the personnel or public. There had been problems of reactivity anomalies, mostly attributable to the small size of the core. These were extensively investigated, giving a deep insight into the core behaviour. The collective radiation dose to the personnel till date is less than what is allowed for a single individual for three years.

Like in any new human endeavour, there have been moments of joy, elation and creativity, alternating occasionally with moments of concern and dejection. But there have never been moments of helplessness, due to the ruggedness of the design and the synergy within the Department. The collective wisdom and the diverse expertise available in DAE were always available when any problem could not be solved in-house.

Two major milestones were reached by FBTR during the silver jubilee year. One is the reaching of the target burn-up of 100 GWd/t by the MOX fuel subassembly simulating the fuel composition of PFBR fuel, which validates the design and fabrication of the fuel for PFBR. The other is the loading back into the FBTR core, of a fuel subassembly using Pu recovered by reprocessing the spent fuel from FBTR, formally closing the carbide fuel cycle. If the first criticality of FBTR heralded the arrival of fast reactor in India, the silver jubilee year may well announce the coming of age of the fast reactor technology in India and the mastery of all the facets of this new technology.

FBTR is the mother of fast reactors in India. Operational feedback from FBTR went into the design of PFBR. The safe operation of FBTR was instrumental in the sanction for the PFBR project. Just as CIRUS reactor in BARC was the cradle of manpower for our PHWR programme, so is FBTR for our fast reactor programme. The O&M personnel of PFBR are being trained in FBTR as a prelude to operating the bigger reactor. FBTR has provided inputs for formulating the regulatory framework for fast reactors in India.

Plant life extension studies indicate that FBTR can continue to be the work horse of our fast reactor programme for ten more Effective Full Power Years. The residual life will be effectively utilised for testing the more advanced metallic fuels on a large scale and irradiation of the core structural materials of the next genre of fast reactors with higher breeding ratios and also with enhanced safety & economics.

Almost all the personnel involved in FBTR project and all the O&M personnel who were trained in Rapsodie have now retired. Hence, there has been a pressing need to archive the consolidated history of FBTR up to the current stage as a part of knowledge management for posterity in the form of a book. It was also felt that the book should be none too technical, in order that the enlightened public too come to know of the travails of a new technology and appreciate the time it takes for any nuclear technology to reach a stage of maturity. It was felt that the book should also be of appeal, at least in part, to the science student community, with a view to expose them to the challenges of nuclear technology and inspire and motivate them to pursue careers in science and technology. Explanatory notes have been given for the technical terms at appropriate places. This book has been prepared, keeping in view the diversity of readership.

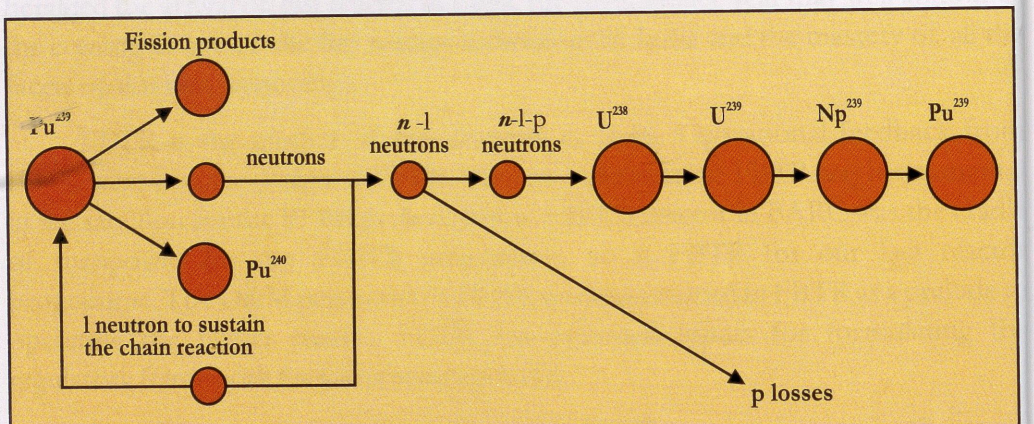
***G. Srinivasan***

Chairman
Editorial Committee

## About Fast Reactors

In nuclear reactors, energy is produced by the splitting of the atomic nucleus; this process is called fission. When a neutron enters heavy atoms ($U^{235}$, $Pu^{239}$ or $U^{233}$), their nuclei become unstable and split into two fragments called fission products which are radioactive in nature. (Fortunately, all but a few of these fission products decay fast and only a small fraction of the fission products have long lives and have to be managed in the reprocessing of fuel and waste disposal). $U^{238}$, which has an abundance of 99.3% in natural uranium, can capture a neutron and gets transmuted to $Pu^{239}$. Similarly, $Th^{232}$ can also get transmuted to $U^{233}$.

Each fission is also accompanied by the release, on an average, of about 2.5 neutrons. Most of the neutrons either escape or get absorbed in the non-nuclear structural / coolant materials in the core. When one neutron out of those produced by each fission is made available to cause the next fission, a sustained 'chain reaction' is possible and we get continuous energy from fission. In a reactor, control of the fission chain is effected by using neutron absorbers called control rods. By adjusting the position of the control rods, the chain reaction rate can be maintained as constant, divergent or convergent. The state where the chain reaction rate is constant is called 'critical state' and the reactor is said to be 'critical'. The divergent state is called 'super-critical' and the convergent state 'sub-critical'.



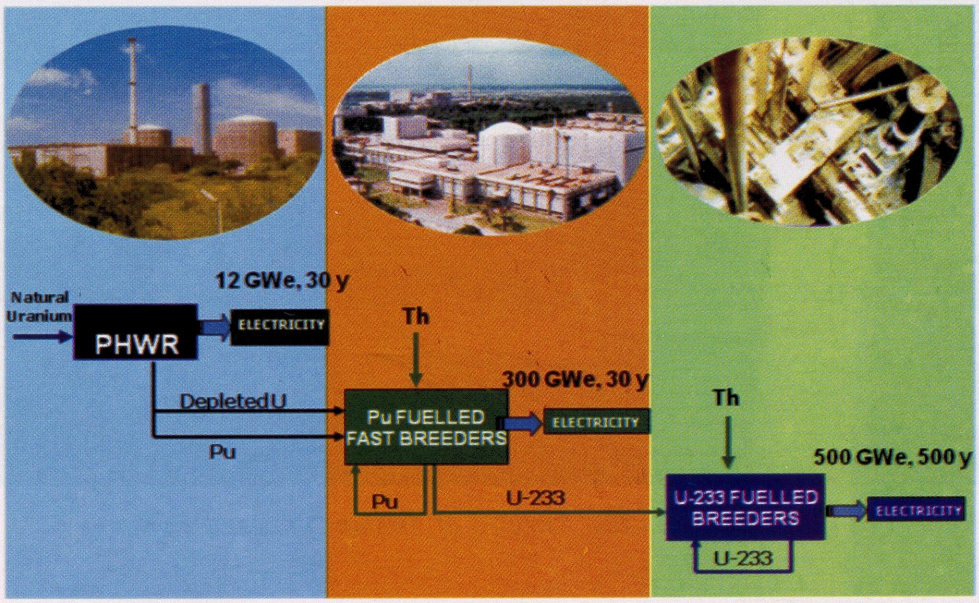

For breeding to be possible, the number of $Pu^{239}$ nuclei produced must exceed the number of $Pu^{239}$ nuclei consumed.
The diagram shows that for one plutonium nucleus consumed, ***n***-p-1 are produced.
It is consequently necessary that ***n***-1-p> 1 and ***n*** >2+p>2.

***Breeding***

The probability of a neutron getting absorbed in a nucleus is higher when the neutron velocities are low. Natural uranium contains only 0.7 % of the fissile $U^{235}$. In order to make natural uranium fuelled reactors critical, it is required to deliberately slow down the neutrons released by fission, using moderators like heavy water, light water or graphite. Reactors operating with moderated neutrons are called 'Thermal reactors'. Depending on the coolant / moderator combination, they are fuelled with natural uranium or slightly enriched uranium. It may be noted that only a small fraction of the total fuel is utilised in thermal reactors. The bulk of the residue is $U^{238}$ and is practically unutilised, but a small fraction gets converted into $Pu^{239}$.

It is possible to convert the $U^{238}$ into useful $Pu^{239}$ in a fast reactor more effectively. The term 'fast' refers to the high energy of the neutrons. Fast reactors do not use any moderating material. The coolant has to be a heavy metal like sodium, in order to avoid moderation. A fast breeder reactor uses a mixture of Pu and natural uranium (or depleted uranium from thermal reactors). While Pu undergoes fission, $U^{238}$ gets transmuted into Pu. By a proper design, it is possible to have a core in which more Pu is produced than consumed; this process is called breeding. The ratio of the quantity of Pu bred to that consumed is called breeding ratio. When breeding ratio

## Three Stages of Indian Nuclear Power Programme



exceeds 1.0, we get more Pu than required for operating the reactor which can go as feed for another reactor. The recovery of unburnt and bred Pu from the spent fuel from a fast reactor, its fabrication into fuel pins and loading the fuel back into the reactor is called 'closing the fuel cycle'. Thus, breeder reactors with closed fuel cycle enable not only effective utilisation of natural uranium, but also the rapid rate of growth of nuclear power. For a country like India, with limited reserves of natural uranium and abundant reserves of Thorium (which can be transmuted to $U^{233}$), fast reactors are a big boon for assured energy in future.

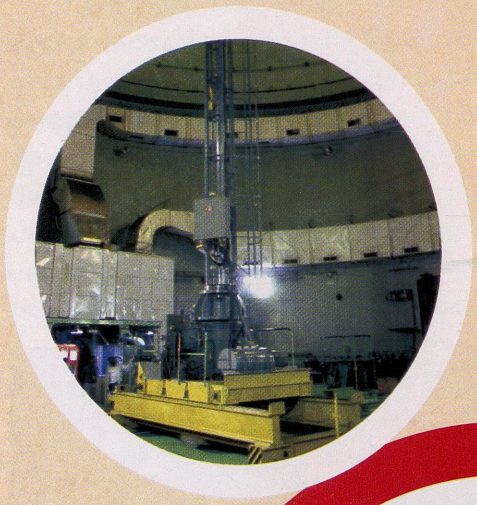
Doubling time refers to the time it takes for getting enough inventory of fuel from a reactor for loading another reactor core of equal size. The higher the breeding ratio, the lower is the doubling time. Ideally, the doubling time of any reactor should commensurate with the time required to build another reactor. With the oxide fuels, the doubling times are quite long (25 to 30 years), whereas a new reactor of standard design can be built in about 5 to 6 years. Hence, advanced fuels are required for rapid deployment of fast reactors. Among the various types of fuels, the metallic fuel is currently considered the best. In the next two decades, the emphasis of our R&D will be on the metallic fuels with advanced core structural materials.

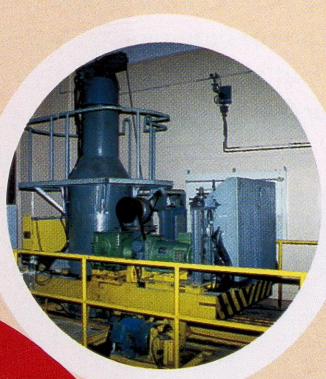


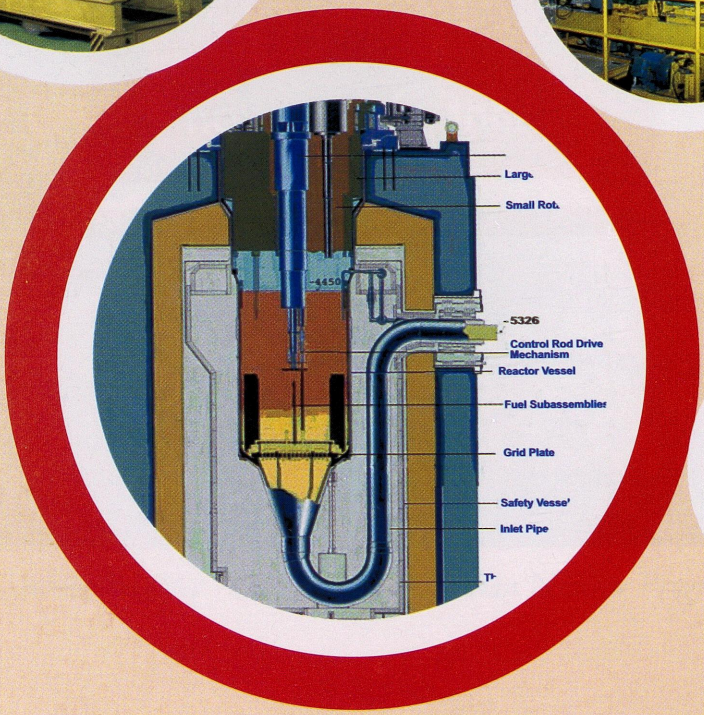
*Dr. Homi Bhabha - the Architect of Indian Nuclear Programme*

Chapter - 1

# Plant Description





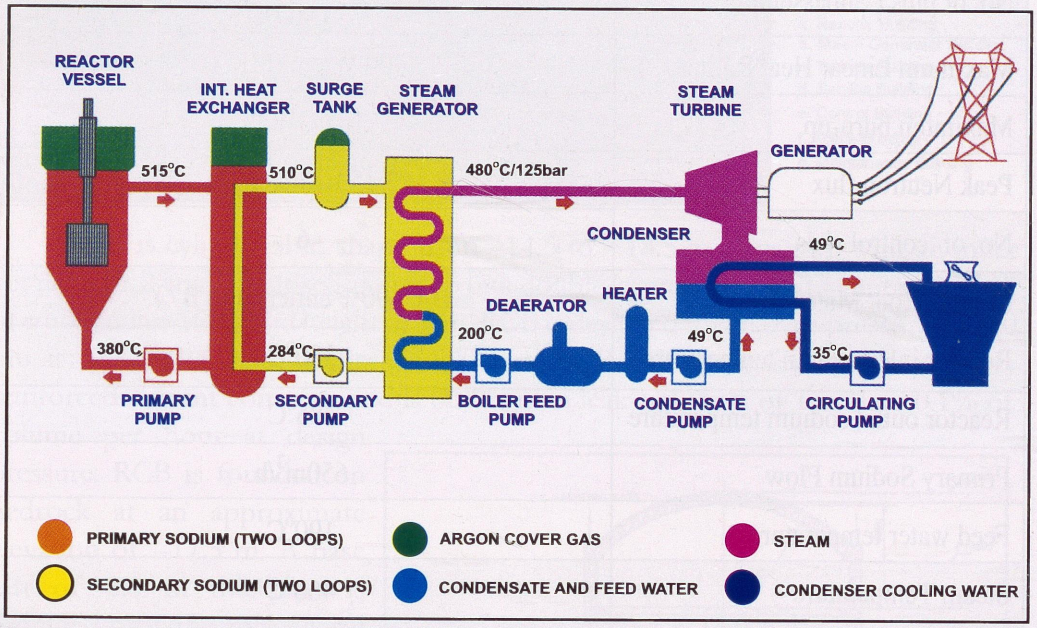

Large
Small Rot.
4450
5326
Control Rod Drive Mechanism
Reactor Vessel
Fuel Subassemblies
Grid Plate
Safety Vessel
Inlet Pipe



# Plant Description

## Heat Transport Circuit

The heat generated in the reactor is removed by two primary sodium loops and transferred to the corresponding secondary sodium loops through Intermediate Heat Exchangers as shown in the schematic flow sheet. Each secondary sodium loop is provided with two once-through steam generator modules. Steam from the four modules is fed to turbine-generator. A Dump Condenser of 100% capacity is also provided in the steam water circuit for continued reactor operation when TG is not available.




*Simplified schematic of FBTR Heat Transport System*

FBTR was originally designed for a power of 40 MWt / 13.6 MWe and the heat transport systems were designed accordingly. However, FBTR started its operation with a small carbide core rated for 10.2 MWt. The excellent performance of the carbide fuel resulted in a rather slow expansion of the core. The reactor has been operated to a power of 18.6 MWt so far.

**Present characteristics of FBTR**

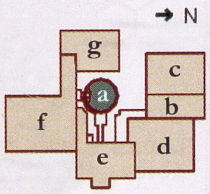
| | |
|---|---|
| Reactor type | Sodium cooled, loop type |
| Reactor Power | 18.6 MWt / 3.2 MWe |
| Fuel | Mark I (70% PuC + 30% UC)<br>Mark II (55% PuC + 45% UC)<br>MOX (44% $PuO_2$ + 56% $UO_2$) |
| No. of Fuel Subassemblies | 27 Mark I + 13 Mark-II + 8 MOX + 1 PFBR Test + 1 Experimental |
| Fuel Pin diameter | 5.1 mm (MK-I & II)<br>6.6 mm (MOX) |
| No. of pins / Subassembly | 61 (MK-I & II)<br>37 (MOX) |
| Maximum Linear Heat Rating | 400 W/cm (MK-I) |
| Maximum burn-up | 165 GWd/t |
| Peak Neutron flux | 3.15 E15 $n/cm^2/s$ |
| No. of control rods | 6 |
| Control Rod Material | $B_4C$ (90% enriched in $B^{10}$) |
| Reactor inlet sodium temperature | 380°C |
| Reactor outlet sodium temperature | 485°C |
| Primary Sodium Flow | $650m^3/h$ |
| Feed water temperature | 190°C |
| Steam Temperature | 460°C |
| Feed water flow | 30 t/h |
| Steam Pressure | 125 $kg/cm^2$ |
| Sodium Inventory | 150 t |
| Steam Generator | Once-through type, 7 tubes in shell, serpentine shaped |
| Turbine-generator | 16 stages, condensing type, 16.4 MWe air cooled. |
| Bypass circuit | 100% Dump Condenser |

**Plant layout**

The Reactor Containment Building is flanked on the north by the Steam Generator Building, on the north-east by the Service Building, on the north-west by the Turbine Building, on the east by the Control Building and on the south by the Active Building.

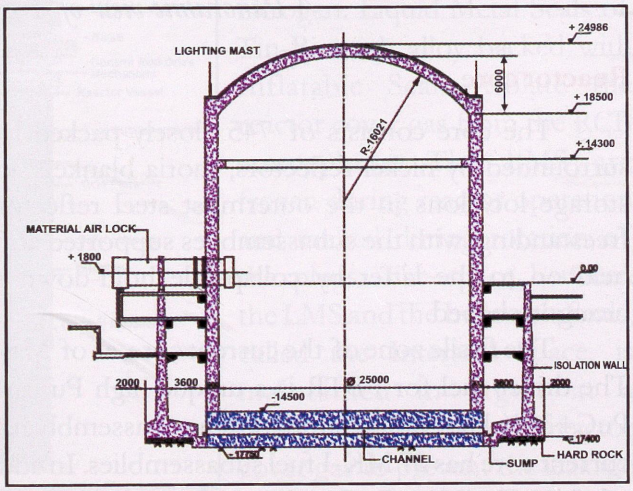
*FBTR layout*






a. Reactor Building
b. Steam Generator Building
c. Turbine Building
d. Service Building
e. Control Building
f. Active Building
g. Maintenance Building

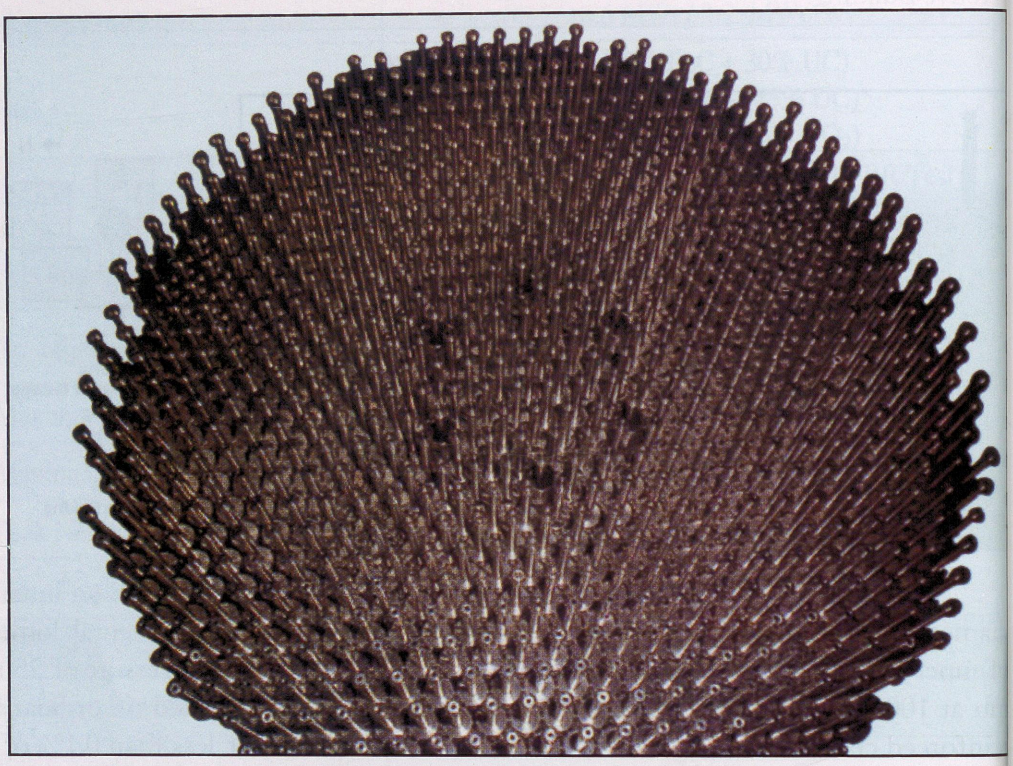
RCB is cylindrical in shape from –14.5 to +18.5 m elevation, with an inner diameter of 25 m. It is closed at the top by a double layered spherical segmental dome of inner radius 16 m, and height of 6 m. RCB is designed for an over-pressure of 250 mb at 100C. The wall thickness is 900 mm. RCB wall is constructed of ordinary reinforced cement concrete, and is built with a leak-tightness of less than 0.1% of volume per hour at design pressure. RCB is founded on bedrock at an approximate elevation of –17.5 m. A base raft of 2.5 m thickness is provided over the rock. A set of radial drains is provided below the base slab and connected to the drain channel in the peripheral gallery to collect any water seeping from the rock. RCB is surrounded by an isolation wall, to arrest lateral seepage of ground water into the building.




*Reactor Containment Building*

## Plant Description

RCB houses the entire primary sodium system in concrete cells below ground level. SGB houses the secondary sodium system, including the steam generators. Turbine Building houses the steam water system. This philosophy is followed to minimise the interface among the active primary & inactive secondary sodium systems and the water systems.



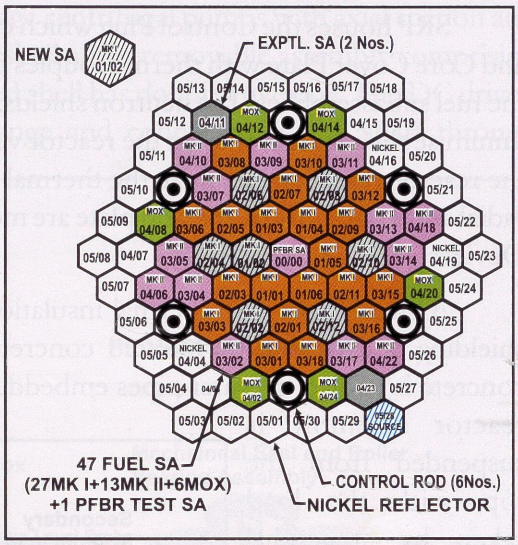
*A Panoramic view of FBTR core*

### Reactor core

The core consists of 745 closely packed locations, with fuel at the centre surrounded by nickel reflectors, thoria blankets and steel reflectors. There are 2 storage locations in the outermost steel reflector row. The core is vertical and freestanding, with the subassemblies supported at the bottom by the Grid Plate and held on to the latter by collapsible hold-down springs. The subassemblies are hexagon shaped.

The fissile zone of the current core, as of May 2010, has 47 fuel subassemblies. The driver fuel for FBTR is a unique high Pu carbide (called MK-I fuel, of 70% PuC+30% UC composition). Each subassembly has 61 fuel pins of Φ 5.1 mm. The current core has 27 MK-I fuel subassemblies. In addition, there are 13 subassemblies of MK-II (55% PuC+45% UC) and six subassemblies of high Pu MOX

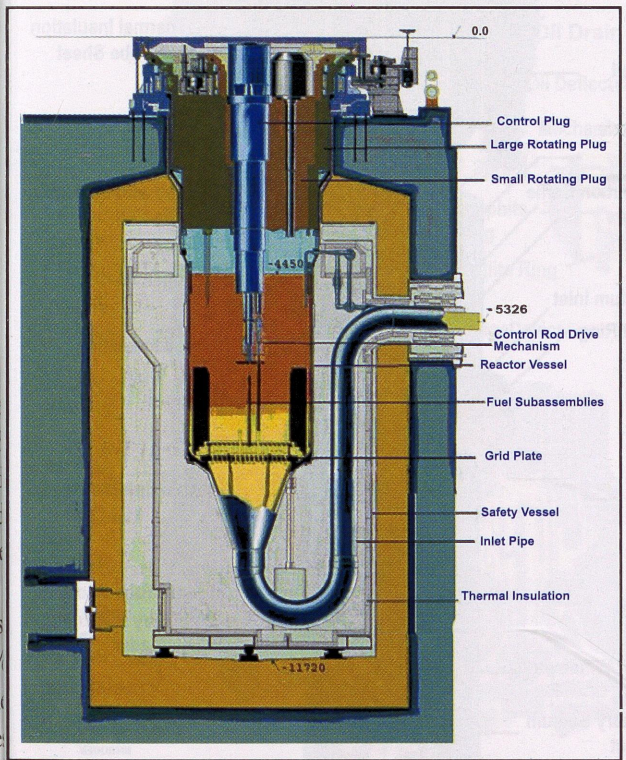
(44% $PuO_2$). There is one test subassembly simulating the PFBR fuel composition. Reactor control is through six control rods of 90% enriched boron carbide.




*The core as of May 2010 (47 fuel subassemblies)*

## Primary system

The major components of the primary system are the reactor assembly, two IHX and two sodium pumps. The entire primary system including the piping is provided with a concentric outer jacket called `Double Envelope' designed to keep the level of sodium in the reactor above the core in the event of any possible leak from the main walls. The Reactor Vessel houses the core. The sodium inlet pipe joins the reactor vessel at the bottom and two sodium outlet pipes radially branch out of the vessel above the core. Charging and discharging of subassemblies are done from the reactor top with the reactor in shutdown state. The reactor is closed at the top by LRP & SRP, which serve as top shields and also enable access to the core locations for fuel handling. The Rotatable Plugs are cooled by nitrogen. Liquid Metal Seals of Tin-Bismuth alloy backed with Inflatable Seals, isolate the reactor cover gas from the RCB atmosphere. The LMS are frozen during reactor operation and melted during rotation of the plugs. The space between the LMS and the Inflatable Seals, called the Interseal Space, is maintained in argon at a pressure higher than the reactor cover gas to prevent the leakage of active cover gas into the RCB.



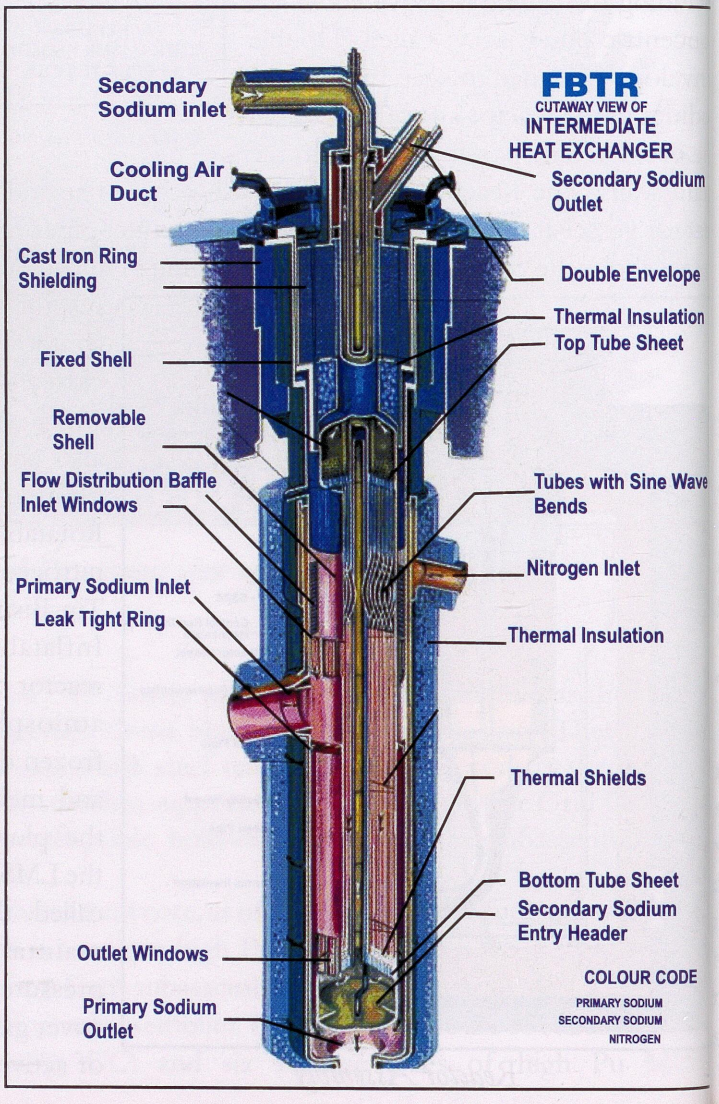

*Reactor Assembly*

## Plant Description

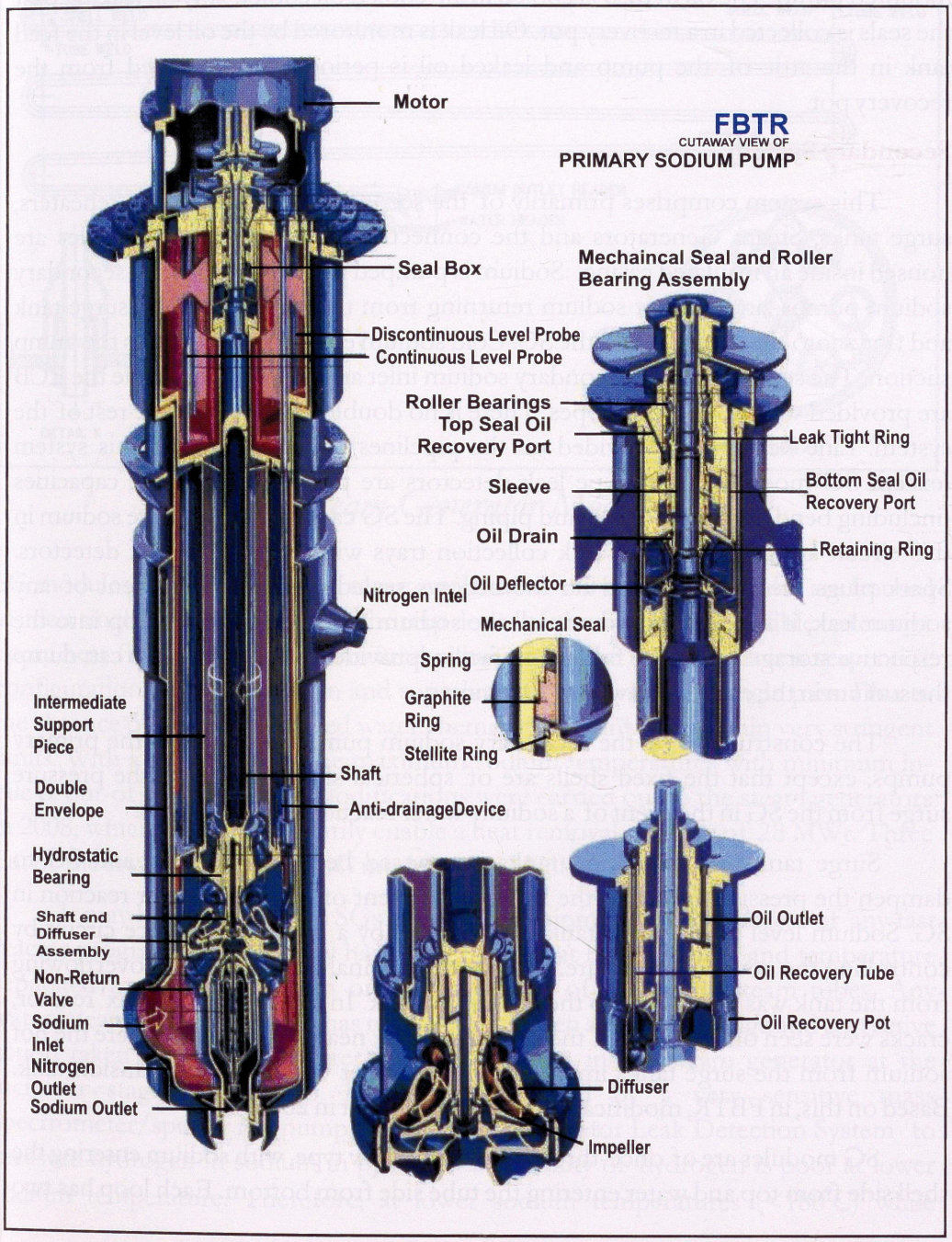
SRP houses the Control Plug which carries six Control Rod Drive Mechanism and Core Cover Plate with thermocouples for monitoring the outlet temperatures of the fuel subassemblies. Ten neutron shields, each 25 mm thick, surround the core and minimise the incident flux on the reactor vessel. Thermal shields are provided inside the reactor vessel to minimize the thermal stresses due to cold and hot shocks. The radial and axial shifts of the grid plate are monitored by two Displacement Measuring Devices.

A steel vessel with thermal insulation surrounds the Reactor Vessel. Radial shielding is provided by borated concrete and structural concrete. The borated concrete is cooled by water pipes embedded close to its inner periphery. The entire reactor assembly is suspended from the top, with the load taken by structural concrete. The reactor is closed at the top by the Anti-Explosion Floor, which is bolted to tie-rods anchored on the structural concrete.

IHXs are vertical, counter-flow heat exchangers and transfer heat from the active primary sodium to the inactive secondary sodium. Primary sodium flows on the shell side and secondary sodium on the tube side. The tube-bundle assembly is removable and has its own shell. It is housed in a fixed vessel with its double envelope.



The sodium pumps are vertical, single stage centrifugal pumps with axial suction and radial discharge. Each pump has a fixed shell and a removable assembly comprising the impeller, diffuser and shaft. The fixed shell has double envelope. The DC driven motors are mounted on the pump flange and connected to the shaft through



couplings. The shaft is supported by taper roller bearings at the top and guided by hydrostatic bearings at the bottom. A float type non-return valve is provided at the bottom to avoid bypassing of the core in the event of one pump trip. Top sealing is by mechanical seals. The seals are lubricated and cooled by oil through a shaft mounted pump. The oil in turn is cooled in air-cooled oil cooler. Any oil leaking past the seals is collected in a recovery pot. Oil leak is monitored by the oil level in the feed tank in the attic of the pump and leaked oil is periodically recovered from the recovery pot.

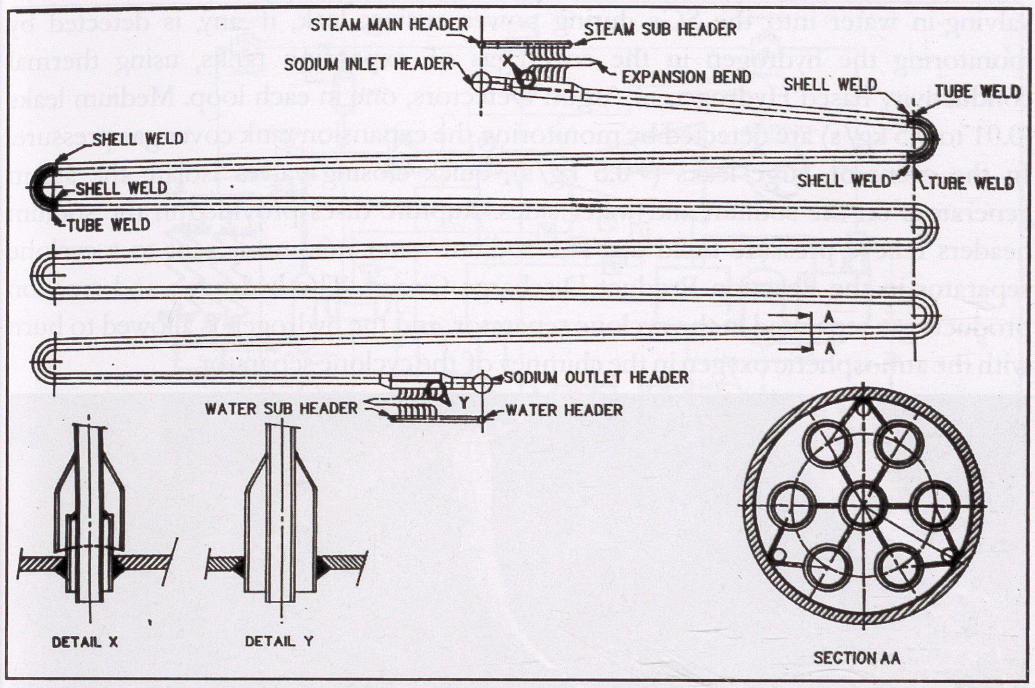
## Secondary System

This system comprises primarily of the secondary sodium pumps, reheaters, surge tanks, Steam Generators and the connecting piping. The SG modules are housed inside an insulated casing. Sodium is pumped into the IHX by the secondary sodium pumps, and the hot sodium returning from the IHX enters the surge tank and flows into the shell sides of the SG. Cold sodium exiting the SG enters the pump suction. The sections of the secondary sodium inlet and outlet pipes inside the RCB are provided with double envelopes. There is no double envelope for the rest of the system. Line heaters are provided on the pipelines and capacities of this system (except SG modules). Wire type leak detectors are provided on all the capacities (including bends in the SG shell) and piping. The SG casing and the entire sodium in this circuit are provided with leak collection trays with spark plug leak detectors. Spark plugs are also provided on the bellows sealed valves. In the event of any sodium leak, it is possible to dump all the sodium from the affected loop into the respective storage tank. The fast dump facility provided for this purpose can dump the sodium in the entire loop within 10 minutes.

The construction of the secondary sodium pumps is similar to the primary pumps, except that the fixed shells are of spherical shape to absorb the pressure surge from the SG in the event of a sodium-water reaction in SG.

Surge tanks are cylindrical tanks interposed between the IHX and SG to dampen the pressure surges to the IHX in the event of a sodium-water reaction in SG. Sodium level in the surge tank is controlled by a level maintenance circuit by controlling the cover gas pressure. As per the original design, sodium overflowing from the tank was being let into the expansion tank. In the French `Phenix' reactor, cracks were seen on the walls of the expansion tank near the locations where the hot sodium from the surge tanks impinges on the cooler walls of the expansion tank. Based on this, in FBTR, modifications were carried out in 2008.

SG modules are of once-through, counter-flow type, with sodium entering the shell side from top and water entering the tube side from bottom. Each loop has two




*Steam Generator Module*

SG modules. Each module has seven water tubes. All the four modules are housed inside a single casing. By opening the trap doors of the casing, it is possible to remove decay heat from the reactor by natural convection. The modules have a serpentine configuration with evaporation and superheating occurring in a single pass. Due to the absence of blow-down, feed water chemistry is maintained within very stringent limits. With a view to reach the maximum sodium temperatures with minimum in-pile & out-of pile inventory, modifications were carried out to the steam generators in 2008, which would temporarily enable a heat removal capacity of 28 MWt. Three out of the seven tubes in each SG module were blanked.

It may be noted that the SGs are critical equipment for the success of any fast reactor programme, since they have water/steam at high pressure and temperature separated from liquid sodium only by the walls of the water/steam tubes. Any water/steam leak into sodium has to be detected even at incipient stage and corrective action taken immediately. Water leak into sodium in the steam generator at the incipient stage (50 mg to 10 g/s) is detected by a very sensitive mass spectrometer/sputter ion pump based Steam Generator Leak Detection System to measure hydrogen in sodium in ppb levels. Solubility of hydrogen is poor at lower sodium temperature. Therefore, at lower sodium temperatures (<180$^0$C) while

valving-in water into the SGs during power raising, leak, if any, is detected by monitoring the hydrogen in the cover gas of expansion tanks, using thermal conductivity based Hydrogen in Argon Detectors, one in each loop. Medium leaks (0.01 to 0.5 kg/s) are detected by monitoring the expansion tank cover gas pressure. In the event of large leaks (>0.5 kg/s), quick closing valves isolate the steam generators on the sodium and water sides. Rupture discs provided in the sodium headers relieve pressure build up. The sodium-water mixture is sent to a cyclone separator in the Reaction Product Discharge Circuit. The hydrogen and reaction products get separated in the cyclone separator, and the hydrogen is allowed to burn with the atmospheric oxygen in the chimney of the cyclone separator.

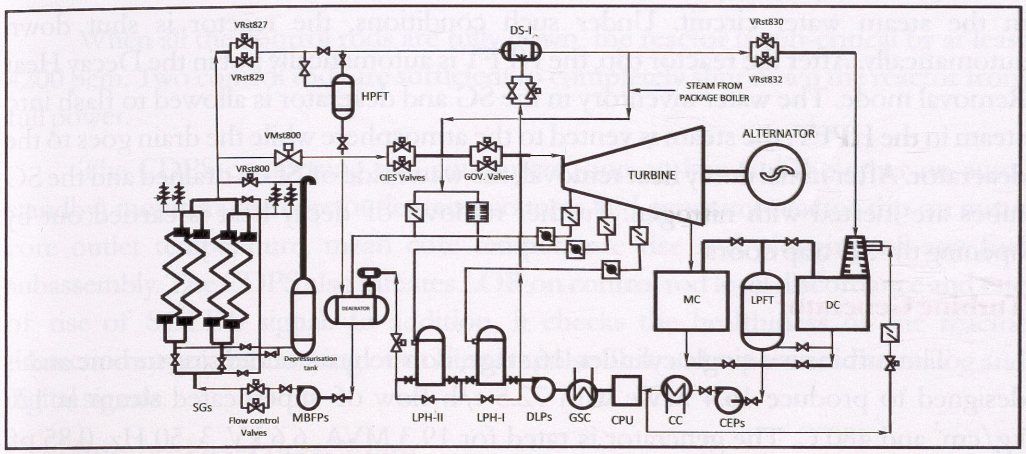



*Steam header of SG Modules showing blanked tubes*

## Steam Water Circuit

Rapsodie did not have a steam water circuit. In order to have a full feel and experience of operating a fast reactor power station, it was decided at the project stage itself to incorporate a full fledged steam water circuit with TG in FBTR.

The steam water circuit has a deaerator, High Pressure Flash Tank , two de superheating stations (DS-1 & 2), turbine, condenser, Dump Condenser, Low




*Schematic of Steam Water System*

Pressure Flash Tank, two Condensate Extraction Pumps, two Deaerator Lift Pumps, two Main Boiler Feed Pumps, two LP heaters (LPH-1 & 2), Condensate Cooler, Condensate Polishing Unit, Depressuriser tank, piping and valves. The high pressure control valves in this circuit are hydraulically operated. One each of CEP, DLP and MBFP are in service; the other respective pumps are on auto-standby. If any of the operating DLP or CEP trips and the auto-standby does not take over in 3 seconds, the reactor is shut down by lowering of control rods (LOR). In case of MBFP trip, if the auto-standby does not take over in 10 seconds, the reactor is shut down by LOR. Condensates from the Main condenser and Dump condenser mix and equilibrate in the LPFT. The CEP takes the condensate from the LPFT and passes it to the CPU through the condensate cooler. The polished condensate is preheated by gland steam from the turbine and admitted to the DLP. The DLP pumps it to the deaerator through two non-contact type heaters where the water is heated by steam extracted from two intermediate stages of the turbine. The Deaerator is located in the mezzanine floor above the operating floor of the TB, to provide enough NPSH for the MBFP. The MBFP takes feed water suction from the deaerator and pumps it through the SG to generate high pressure, high temperature steam. This steam is sent to the turbine and the dump condenser. During TG operation, about 2% of the steam is sent to the DC to keep it hot. Steam admission to the DC is after reducing the steam pressure in two stages, and also de-superheating in two stages. When the TG is not available, entire steam is dumped into the DC.

The HPFT is located on the roof of the Turbine Building. During initial valving-in of SG with water, the wet steam is flashed in this tank, and the steam alone is sent to the DC while the drain goes to the deaerator. After steam conditions improve, the HPFT is bypassed. The HPFT also serves to remove decay heat when the steam water system becomes unavailable due to the tripping of any of the pumps

in the steam water circuit. Under such conditions, the reactor is shut dow automatically. After the reactor trip, the HPFT is automatically put in the Decay Hea Removal mode. The water inventory in the SG and deaerator is allowed to flash int steam in the HPFT; the steam is vented to the atmosphere while the drain goes to th deaerator. After initial decay heat removal, the water side of SG is drained and the SG tubes are inerted with nitrogen. Further removal of decay heat is carried out b opening the SG trap doors.

## Turbine Generator

The turbine is a single cylinder, 16 stage, non-reheat condensing turbine and i designed to produce 16.4 MWe with 72.5 t/h flow of superheated steam at 12 $kg/cm^2$ and 480°C. The generator is rated for 19.3 MVA, 6.6 kV, 3, 50 Hz, 0.85 pf 3000 rpm. The rotor is air-cooled with a closed circuit; the air is cooled by water in heat exchanger. The generator field is powered by a shaft driven exciter rated for 11 kW & 220 V (dc). The exhaust hood of the turbine is connected to the Mai condenser. The turbine has its own separate foundation, separated from th operating floor. The low power turbine using steam at such high pressure an temperature was specially designed for FBTR by M/s BHEL, Bhopal India.

## Condensate Cooling Water System

The terminal heat sink is a cooling tower of induced draft, counter-curren type, with four compartments and four fans. The water from the cooling tower basi returns to the circulating water pit by gravity through an octagonal concrete tunne The evaporation losses in the system are made up from underground raw wate storage tanks located west of the TB. Two circulating water pumps take suction fro the circulating water pit and pump the cooling water to the main condenser, dum condenser, condensate cooler and all other heat exchangers of auxiliary systems i the plant. The evaporation losses in the system are made up with Palar river wate from underground raw water storage tanks fed by gravity from the reservoir o adjacent Madras Atomic Power Station.

## Reactor Protection

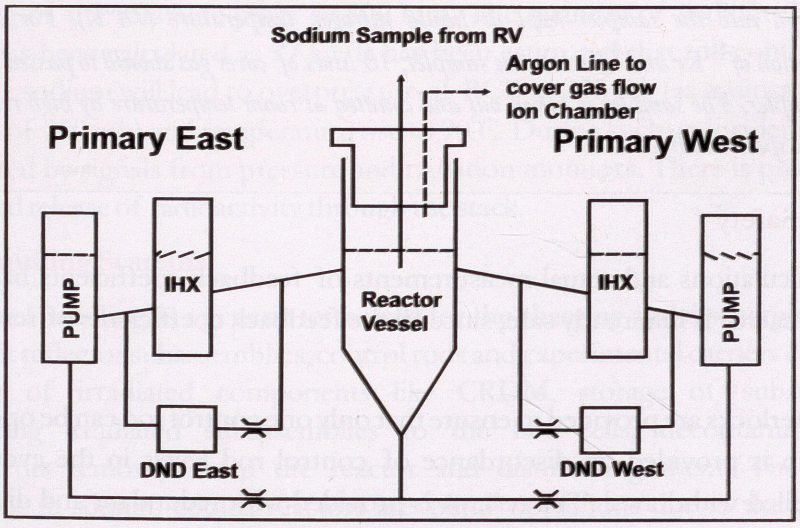
The reactor power control and shutdown are by six control rods. Fo shutdown, the rods are inserted into the core in two modes, viz. LOR, wherein all th rods are driven down by the respective drive mechanisms at a speed of 1 mm/s, an scram, wherein the rods drop down by gravity in less than 400 ms. Scram is initiate by neutronic parameters (power, period & reactivity), core thermal parameters an Delayed Neutron Detection system. LOR is ordered by thermal hydraulic parameter of the plant, which, because of the thermal inertia, do not call for fast shutdow Power control is manual.

When all the control rods are fully down, the reactor is sub-critical by at least 4200 pcm. Two control rods are sufficient to completely shut down the reactor from full power.

The CDPS comprising two computers (one on-line and the other on auto-standby) monitors the core outlet temperatures and generates reactor trip on mean core outlet temperature, mean core temperature rise and plugging of any fuel subassembly. The CDPS also initiates LOR on control rod level discordance and rate of rise of SGLDS signal. In addition, it checks the healthiness of the reactor protection system by carrying out a Fine Impulse Test and supervises the analog and digital signals.

## Clad Rupture Detection Circuit

The fuel pellets are enclosed in a SS clad. During the course of reactor operation, fission products (solid and gaseous) are generated inside the cladding of a fuel pins. When a clad failure takes place, these fission products are released in the coolant and the cover gas. To avoid contamination of the coolant by the fission products, it is essential to detect fuel clad failure and locate the failed fuel and discharge it at the earliest. In FBTR, there are two methods used for detecting clad failure.




*Failed Fuel Detection Systems*

## Delayed Neutron Detection in sodium

In case of fuel clad failure, delayed neutron precursors like bromine and iodine called are released to primary sodium. In order to measure the delayed neutron activity in primary sodium, sodium is sampled from reactor vessel outlet line. This

sodium is passed through the delayed neutron detector blocks, situated in th shielded cells and finally join suction pipe of the primary sodium pumps. The tw Delayed Neutron Detector blocks are provided for east and west loops which ar situated in two cells called CRD pits. It consists of a shield block in lead, followed by moderator block in polyethylene. The moderator blocks house six boron counter Reactor undergoes scram when the DND signals cross twice the background.

### *Clad Rupture Detection in reactor cover gas argon*

*When fuel clad ruptures, radio active fission gases such as Xenon and Krypton are released from the fuel pins and collect in the reactor vessel cover gas space. Thus, increase in the cover gas activity indicates a possible clad rupture in the core. The circuit make use of a gas flow ion chamber to detect the fission product activity in the cover gas. The cover gas sample is drawn continuously from the reactor vessel and monitered by the ion chamber.*

*In addition to this, a failed fuel localisation system has been installed in the CRD argon experimental circuit. This system is intended to establish the $^{85}Kr$ / $^{87}Kr$ ratio in the cover gas to dermine the age of failed fuel. The system consists of an activated charcoal adsorber, an activated charcoal sampler, a flow meter and a vaccum pump. The adsorbed cover gas is eluted by argon into the sampler kept at liquid nitrogen temperature (78°K). For complete adsorption of $^{85}Kr$ and $^{87}Kr$ into the sampler, 10 litres of cover gas should to passed through the sampler. The sampler is taken out and counted at room temperature by high resolution gamma spectrometry.*

## Reactor Safety

Calculations and actual measurements of feedback coefficients have show that the reactor is inherently safe, since all the feedback coefficients of reactivity a negative.

Interlocks are provided to ensure that only one control rod can be operated at time. Trip is provided on discordance of control rod levels in the event of a uncontrolled withdrawal. The reactor is provided with redundant and diverse trip from neutronic and core thermal parameters i.e. lin-P, log-P, period, reactivity, mea core outlet temperature and mean core temperature rise. These features avoi Transient Over Power Accidents.

The Double Envelope of the primary sodium loops safeguards against Lo Of Coolant Accident. Leak-before-break principle enables early detection befo any leak escalation.

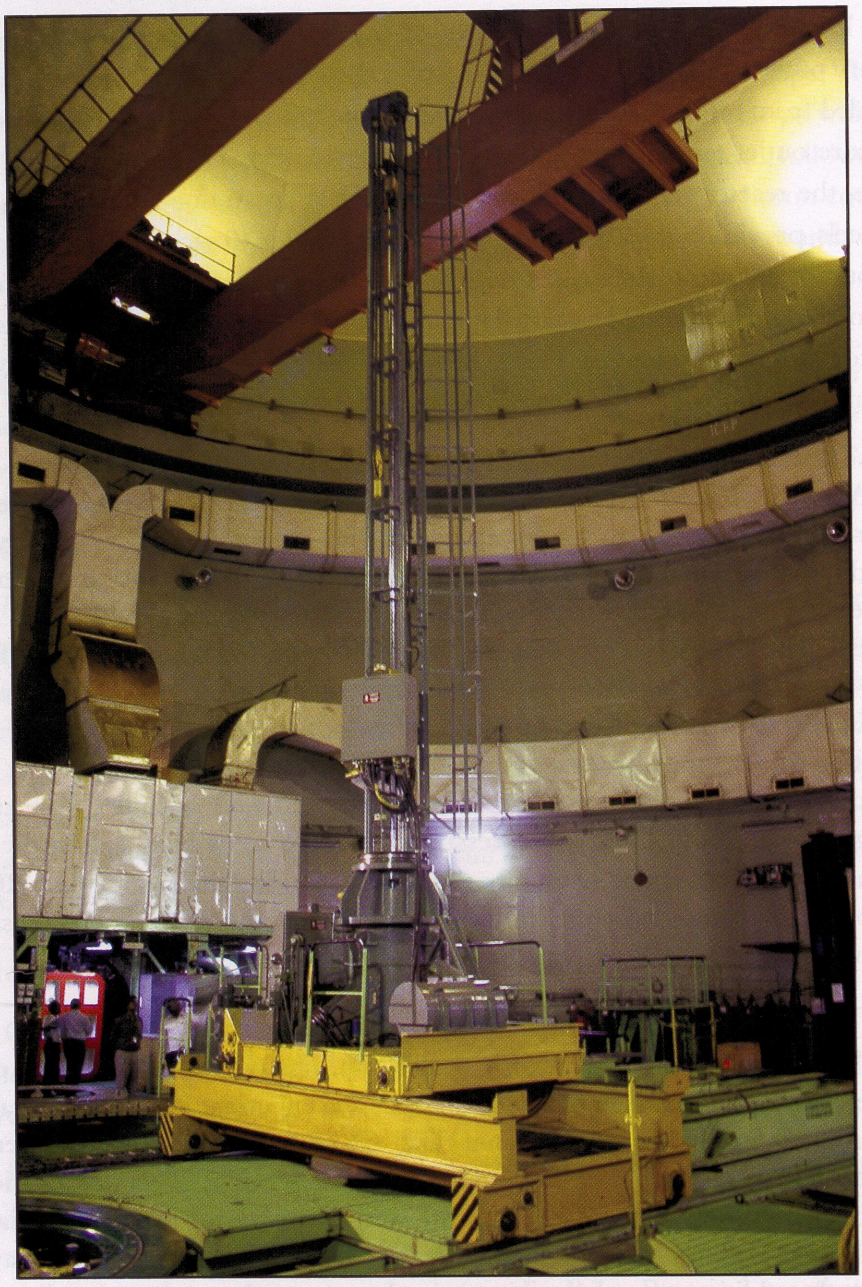
Loss Of Flow Accident is avoided by the parallel operation of both the primary pumps and the absence of any valve in the primary sodium loops. Trips are provided from total flow through the reactor, individual loop flow, individual pump trip, core outlet temperature and core temperature rise. During any offsite power failure, the reactor trips and primary pumps coast down due to the inertia of the flywheels, providing at least 30% of the rated flow after 40 seconds of power failure. The pumps are meanwhile taken over by the emergency diesel power. In case of a station blackout, dedicated battery banks take over and run the primary pumps at 150 rpm for at least 30 minutes. Decay heat removal beyond this time is possible by natural convection by opening the trap doors of the steam generator casing.

The three scenarios taken into account for Design Basis Accident are TOPA, LOCA and LOFA without the actuation of the shut down system. The resulting scenario under any of these conditions is partial core melting, rapid rise of power due to additional reactivity caused by slumping of core, fuel vaporisation and core disassembly. Theoretical calculations indicate the maximum mechanical energy release will be 14.5 MJ or less than 3.5 kg of Tri-Nitro Toluene equivalent. Reactor vessel can safely contain 8.8 kg of TNT equivalent. The energy released through the top of the reactor will be absorbed by the tie rods of the anti-explosion floor to limit the lifting of the rotatable plugs to 800 mm. The quantity of sodium that will be ejected has been calculated as 27 kg. It has been estimated that full combustion of 100 kg of sodium will lead to overpressure of RCB to 235 mb (as against the design pressure of 250 mb) and temperature rise to 90$^0$C. During such an incident RCB will get isolated by signals from pressure and radiation monitors. There is provision for controlled release of radioactivity through the stack.

## Fuel Handling Systems

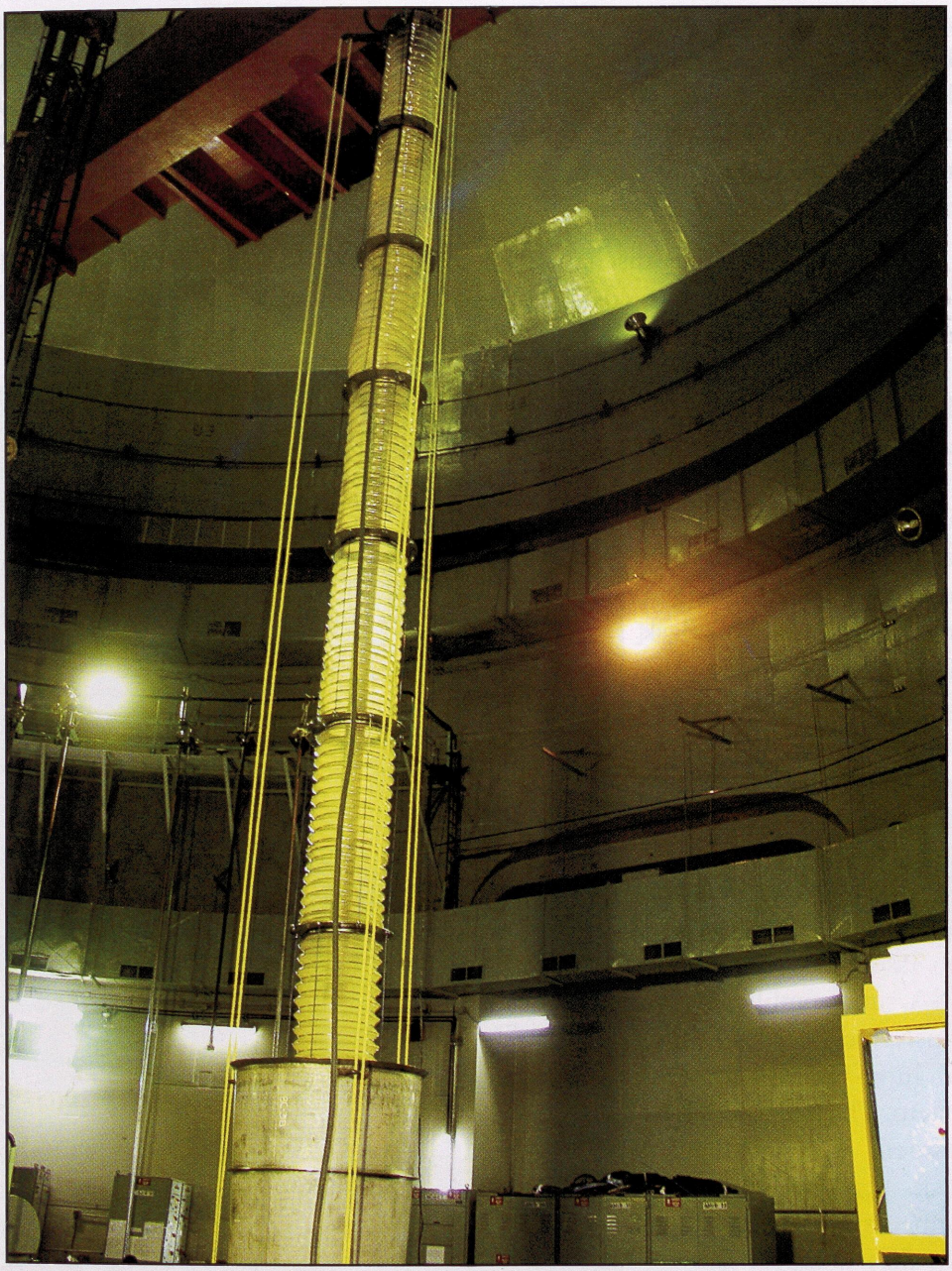
'Fuel handling' is a generic term used for the charging and discharging of fuel, blanket & reflector subassemblies, control rods and experimental carriers & capsules, handling of irradiated components like CRDM, storage of subassemblies, transferring irradiated subassemblies to the hot cells, decontamination of components removed from the reactor and dismantling them. The handling operations in sodium cooled reactors are 'blind' operations due to the opacity of sodium. In addition to radioactivity, it is also essential to carry out all handling operations under inert cover to avoid the reaction of sodium sticking to the components with air. All the handling mechanisms are long, slender components, requiring precise alignment.

Fuel handling is carried out with the reactor in shut down state and sodium at 180°C. It is a group of operations consisting of charging of fresh fuel, in-pile



*Charging Flask*

transfer to storage locations inside the reactor for decay heat removal and discharging of irradiated and sufficiently cooled fuel from the reactor. The charging and discharging operations are done with the help of charging and discharging flasks. Transfer operations are done with the SA fully immersed inside the sodium. For this



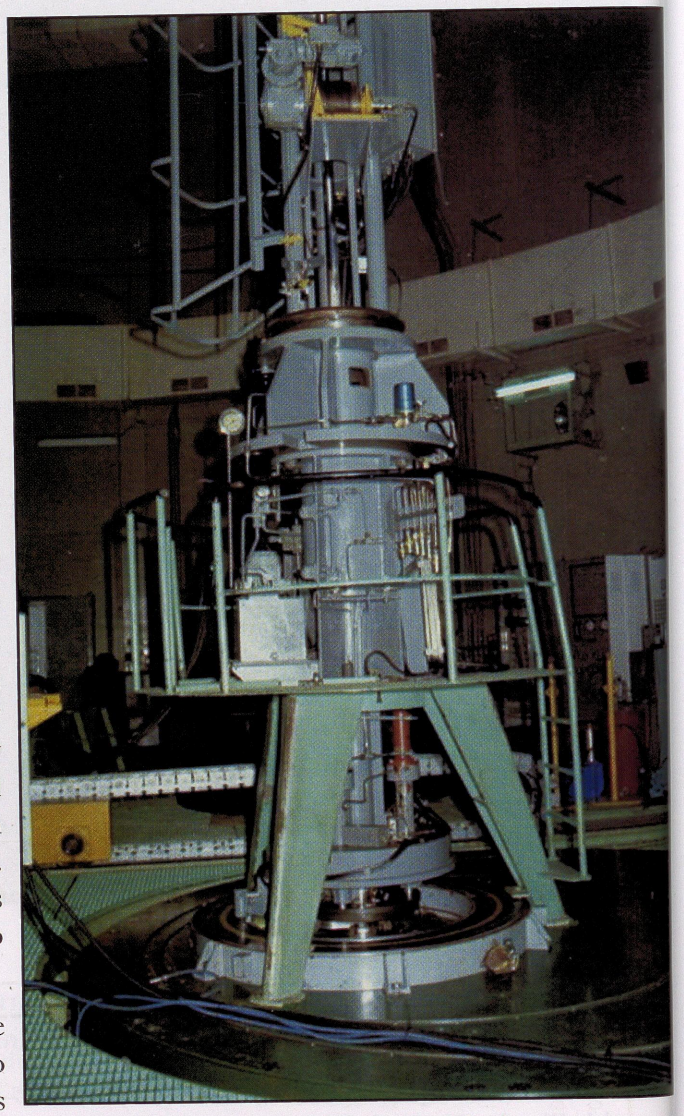
*Special Flask - 5*

purpose, the Charging Flask can be removed from its carriage and installed on a stool on the Fuel Handling Floor, thus converting it into a Transfer Flask.

The reactor top is closed by a Large Rotatable Plug within which a Small Rotatable Plug is mounted eccentrically. The fuel handling canal is mounted eccentric

to the SRP. Fuel Handling Canal can be aligned over any location in the core by rotating the LRP & SRP about their axes. The FH canal is closed with a shield plug which has to be removed for accessing the core. In order to guide the subassemblies into and out of the reactor during handling, a long sturdy guide tube of about 6 m length is installed inside the canal. The guide tube is handled by a Special Flask called SF-5.

The subassemblies discharged from the reactor are immediately stored in inerted pots in the air-cooled Discharge pits inside RCB. Subsequently, they are shifted to the air-cooled Irradiated Fuel Storage Area for longer storage till they are ready to be shifted to the adjoining RML. A short flask called Secondary Flask is used to transport the discharge pots / transfer pots containing subassemblies discharged from the reactor. This flask can travel through the Material Airlock on rails in a carriage called Transfer Carriage. Adequate shielding is provided both during transportation by the flask and during the storage of the irradiated sub-assemblies. From the IFSA, subassemblies can be transferred to RML using an inter-facility transfer system. The SA is vertically charged into a shielded flask kept in a trench connecting FBTR & RML. The flask is moved on rails in the trench and lifted up into the hot cells of RML.



*Transfer Flask*

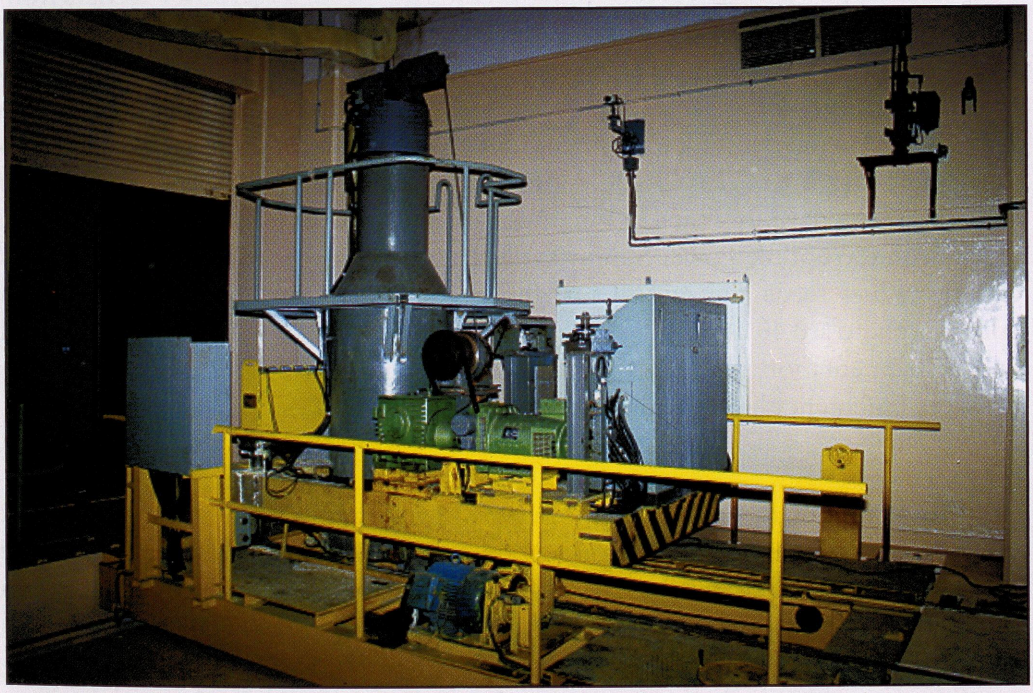
Infrequently, there arise requirements to discharge long components e.g. primary sodium pumps,

IHX, the lower parts of CRDM (with or without the control rods), radioactive experimental / irradiation devices, level probes, etc. A Special Flask called SF-1 is used for handling primary sodium pumps & IHX. Long components from the reactor are removed using a heavily shielded 30 T flask called SF-2. The components removed from the reactor are immediately stored in inerted decay pits inside the RCB. They can then be transported to the decontamination area in the Active Building. The components are decontaminated of sodium using a mixture of nitrogen and steam and then cleaned with a spray of water. There is also an adjoining facility for cutting the components for disposal or retrieval of parts.

## Radiation and Air Activity Monitoring system

The Radiation and Air Activity Monitoring System consists of 10 particulate activity monitors, 12 area gamma monitors, 6 gas activity monitors, 3 in-duct gas activity monitors, and one discharge flask high range gamma monitor. Continuous monitoring of RCB air for particulate and gas activity by Particulate Activity Monitors and Gas Activity Monitors respectively inside RCB is unique for FBTR.

Special purpose radiation monitors such as $Na^{24}$ activity monitor were introduced in FBTR for primary sodium leak detection, fuel clad failure detection and decay heat measurements. Also, four numbers of sodium fume activity monitors are installed in B-cell ventilation ducts, meant to detect the leak of active primary



*Secondary Flask*

sodium from the pipelines into the shielded concrete cells. A gas activity monitor to detect sodium aerosol in the double envelope (filled with nitrogen gas) of reactor vessel and a particulate activity monitor for detecting primary sodium leak in purification cabin were introduced later. Reactor cover gas activity is monitored continuously using two gas flow ionization chambers known as CRD monitors for the early detection of fuel pin failures. The Discharging Flask is provided with a high range gamma monitor to prevent any inadvertent removal of fuel sub-assembly with more than 400 W of decay power. Two portal monitors, plastic scintillator based ones, are installed at the Fresh Fuel Storage Area and at the final exit point of FBTR.

Generation of solid waste in FBTR is very minimal. Liquid effluent from decontamination facilities and condensates from air handling units located in potentially active areas are stored in separate SS tanks and pumped to Centralized Waste Management Facility for disposal. The major and only radioactive component expected to be present in the gaseous effluent of FBTR under normal operating conditions is $Ar^{41}$. Because of the retention of iodine by the coolant Na, special equipment for removal or control of iodine is not required for FBTR off gas system. Efficient iodine filter system provides adequate control of the release of radioiodine, even in the event of substantial clad failure or during an incident of sodium leak.

## Electrical power supply

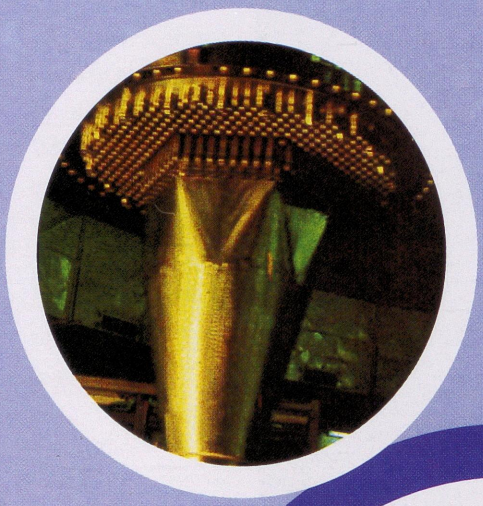
Offsite power supply is received from the grid through two 33 kV feeders from the adjoining MAPS and stepped down to 6.6 kV by two 10 MVA transformers. The 6.6 kV buses are interconnected, so that the tripping of any single offsite feeder does not affect station power supply. The turbine-generator can be synchronized to the 6.6 kV bus. Station power is distributed at 415 V through two emergency buses and two non-emergency buses. Two 1000 kVA, quick-starting emergency diesel generators provide emergency power at 415 V in the event of offsite power failure. 220 V, single phase Uninterruptible Power Supply system, comprising rectifier-charger, batteries and invertors, feeds the critical instrumentation loads including the CDPS. In addition, there are 48 and 24 V dc power supplies with battery back-up for control and instrumentation and 220 V dc supplying control power supply for circuit breakers, emergency lighting and dc flushing oil pump of TG.

## Auxiliary systems

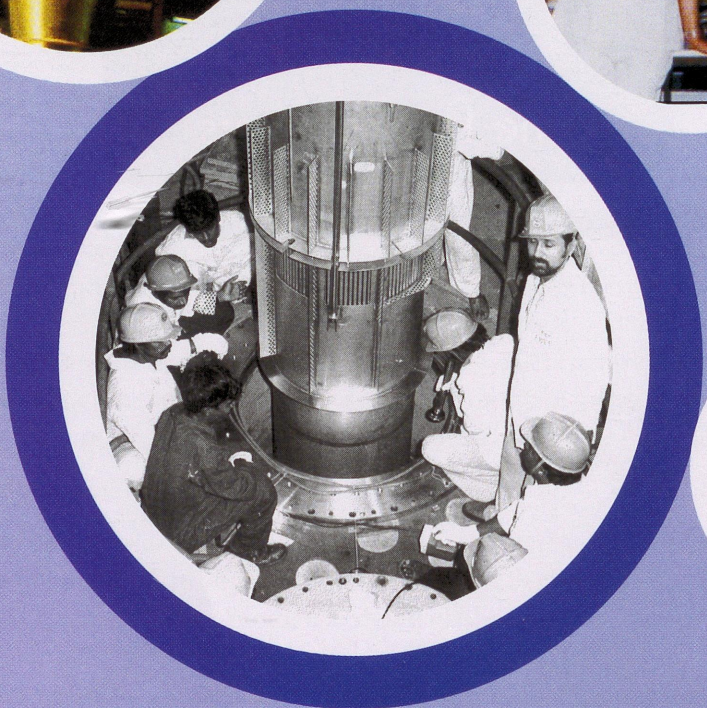
Station auxiliaries comprise the Service and Circulating Water System for heat removal from main and dump condensers and other heat exchangers, fire water system, chilled water system, biological shield cooling system, compressed air system to provide service, insrument and mask air, service argon system, service nitrogen system with a Pressure Swing Adsorption nitrogen plant and Horten sphere, Air-conditioning and Ventilation System, make-up Demineralised plant and auxiliary boiler.

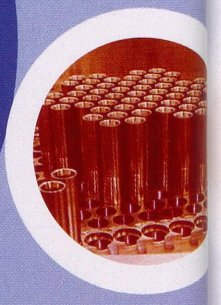
Chapter - 2

# The Project Years









# The Project Years

## Beginnings at BARC

Even when the major thrust of R&D activities in BARC was on the first stage of Indian nuclear programme, studies on the second stage had been initiated in parallel at the Reactor Engineering Division of BARC. FBTR had its beginnings in 1968 in the Fast Reactor Section of this Division. The Section carried out theoretical studies on several design options for fast reactors in the Indian context. They studied different combinations of coolants (sodium, NaK, mercury, molten salt & helium gas) and fuels (MOX, carbide, nitride & metal). The focus was on identifying the design of a safe reactor with the highest breeding ratio for eventual commercial deployment. It was concluded that sodium cooled reactor with carbide or metallic fuel will have to be the ultimate choice in the Indian context. Before launching on a large scale indigenous programme on fast reactors, however, there was an obvious need for a test reactor aimed at providing experience in fast reactor operation and large scale sodium handling. The reactor was also required to serve as a test bed for irradiation of fast reactor fuels and materials. The need for comprehensive R&D on all aspects of fast reactors, up to the closing of the fuel cycle, was also recognized. The Fast Reactor section initiated the design of a 10 MWt test reactor.

## Reactor Research Centre

Dr. Vikram Sarabhai, recognising the magnitude of the mission, decided to set up a centre exclusively devoted to the development of sodium cooled fast reactor technology in India. The location chosen was Kalpakkam, where construction of the first indigenous twin units of 235 MWe Pressurised Heavy Water Reactors, christened as Madras Atomic Power Project, was in progress. The availability of land within the exclusion zone of MAPP, low population density, assurance of water supply, low seismicity and proximity to MAPP which would facilitate sharing of infrastructure and exchange of knowledge base were the major reasons for the choice of Kalpakkam as the site for the proposed research centre. The centre was planned with the test reactor as its hub. It would have engineering halls for sodium loops and component testing, hot cells for Post-Irradiation Examination and laboratories with advanced facilities for materials and metals research, safety studies, sodium and fuel chemistry studies, development of instrumentation, R&D on reprocessing, etc. Thus the entire gamut of R&D for the second stage of our programme was included in the mission of the centre. The Reactor Research Centre

was started in 1971. It was renamed as Indira Gandhi Centre for Atomic Research in 1985.

In retrospect, the vision and foresight of Dr. Sarabhai were astounding, since the site chosen has been able to accommodate not only the twin units of MAPS and the full fledged Research Centre, but also the site for the 500 MWe PFBR, the Fast Reactor Fuel Cycle Facility, several BARC facilities and the Safety Research Institute of Atomic Energy Regulatory Board. It is also the proposed site for two more 500 MWe fast reactors, thus making it a veritable fast reactor park.



*Dr. Vikram Sarabhai sowed the seed for the second stage of Indian Nuclear Programme by setting up the Reactor Research Centre at Kalpakkam in 1971*

## Rapsodie and its Indian Avatar

A few experimental fast reactors were then in operation around the world. The major of these were EBR-II in USA, BOR-60 in USSR, DFR in UK and Rapsodie-Fortissimo in France. It was decided that the design of one of the reactors could be adopted, instead of designing a test reactor ab-initio. The bonhomie between French and Indian governments was conducive to have an agreement for transfer of the design and drawings of Rapsodie-Fortissimo reactor, training of Indian team in design, construction & operation and transfer of manufacturing technology of critical components.



*The Rapsodie-Fortissimo Reactor, Cadarache, France*

The French Rapsodie reactor was built between 1962 and 1966. It went critical on 28th January 1967 and reached the design power of 20 MWt two months later. In 1970, the Rapsodie was upgraded to 40 MWt and was renamed as Fortissimo. Rapsodie-Fortissimo had a sodium-to-air heat exchanger as the terminal heat sink. France was then building the 250 MWe Phenix reactor, with steam generators of modular design. By installing four of these modules in place of the sodium-air heat exchanger, it would be possible to convert the Indian test reactor into a power station. This modification had one major technical edge over the option of sodium-air heat exchanger. In the steam generators, high pressure water / steam is separated from low pressure sodium only by the walls of the tubes. Hence reliable operation of the steam generators is critical to the success of any fast reactor. The proposed modification from Rapsodie would thus give us operating experience with steam generators. The French agreed with the proposal and also provided us the design of the Phenix steam generator modules. The secondary sodium system was modified accordingly.

Electricity generation is not the prime agenda of a test reactor. Nevertheless, it was decided to install a turbine generator also, with a dump condenser as a back-up in order to continue the reactor operation if the TG becomes unavailable. To some extent, the incorporation of the TG also got justified by the major energy crunch faced by the world later, in 1973, due to the oil crisis. Electricity generation from all possible avenues was the order of the day. In the Indian context, the 13 MWe which could be generated from a 40 MWt reactor was not small; enough to meet the loads of the whole research centre. Thus, the Indian avatar of Rapsodie became a small power

station. The reactor was named 'Indian Fast Test Breeder Reactor' and prelimina project report was prepared. It was renamed as 'Fast Breeder Test Reactor' in 1971.

## The Project

As envisaged by the Indo-French agreement, a joint team comprisi personnel from DAE, CEA and the French industries was formed in May 1969 wi the objective of preparation of a design report. The design team went to France 1969. The team included reactor physicists, engineers & draughtsmen. Work relat to the nuclear portion of the plant was carried out in Cadarache while some of t preliminary studies related to the turbo-generator and steam water system were do in India through consultants. The first batch of design engineers came to Kalpakk in 1971 and prepared the Project Report of FBTR. The project cost, as of April 19 was estimated as Rs. 452.5 millions, excluding the cost of the fuel. A Technical Liais Mission was also set up in Paris.

The project organisation had the Project Design Wing, responsible for release of drawings & specifications; the Project Engineering Wing, responsible the procurement and fabrication of all components; the Project Construction Wi responsible for all site construction & erection activities; and the Quality Contro Inspection Wing, responsible for all inspection activities. During the course of project execution, some of the activities were executed in the Task Force mode, w each Task Force having system engineers from all the wings. The mandate of the T Forces was to get all components fabricated and installed at site as per the respect designs, specifications and quality requirements. The designer was made the leade the Task Force. The Task Force mode had the major advantage that the designers



*The project leaders discussing w the French team. Mr.N.L.Ch Principal Project Engineer (first the right) steered the project completion. Mr.S.R.Paranj Principal Design Engineer (seco on the right), fondly called as Father of Fast Reactor program in India, was fully responsible the design, including the choice the fuel. Mr.N.Srinivasan, (th from the right and leaning on table) is the Founder Director RRC and all the programmes this Centre were initiated by hi*

a feel of the actual fabrication & construction difficulties and the procurement & quality control engineers had a feel of the design. It is to the credit of the Task Force leaders that in spite of having the responsibility of timely delivery of components, there was no compromise on quality.

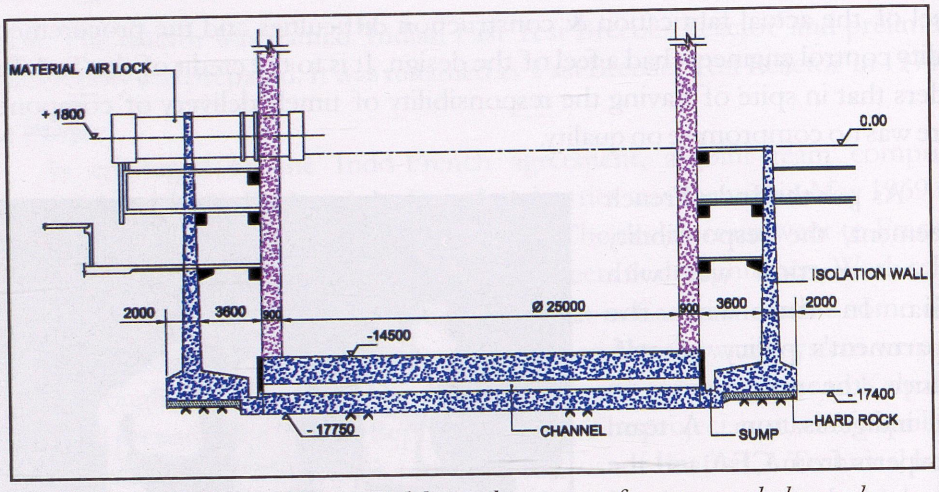


*Shri S. R. Paranjpe, Shri N. L. Char and (Late) Dr Placid Rodriguez during discussion in RCB*

As per the Indo-French agreement, the responsibility for construction was with India. In line with the Department's policy of self-reliance, the accent was on total indigenisation. A team of experts from CEA and the French industry associated with the French fast reactor programme was invited to India to assess the potential of the Indian industries to undertake FBTR jobs. The team submitted a favourable report and identified those components which would be difficult to make in India. With an indigenous component of more than 80 %, FBTR is in fact a symbol of our self-reliance and indigenous capability. The imported components of the project which include raw materials for the sodium systems and Nuclear Steam Supply System, borated graphite, control rods, one control rod drive mechanism, one sodium pump, grid plate, bellows-sealed valves, sodium instrumentation, MI cables and Main Boiler Feed Pumps are from France. The ball bearings of the Large Rotating Plug and Small Rotating Plug and the quilted stainless steel insulation surrounding the reactor are German products. The French side also gifted the neutron shields and DU blanket subassemblies which were removed from the Rapsodie reactor mock-up. All the other components were made in India. The blankets were never loaded in FBTR, but they found their application, much later, in the shielding experiments for PFBR at Apsara reactor, BARC.

## Civil Structures

Rapsodie had a steel containment for the reactor. From the cost considerations, it was decided to go in for a concrete containment for FBTR. The inner cells in RCB were retained as in Rapsodie except the shielding around the reactor, which was nitrogen cooled, high temperature concrete in Rapsodie, was

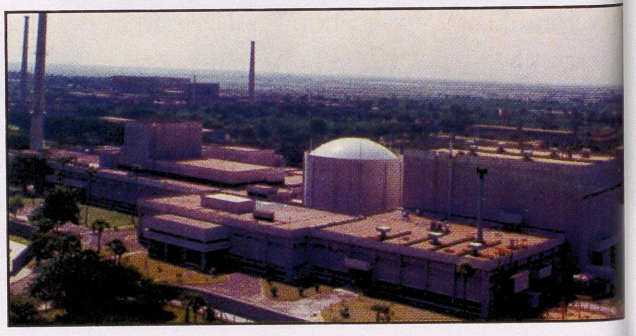



*RCB Isolation wall to avoid lateral seepage of water and channels in the raft to collect seepage from base rock*

changed over to water cooled, borated concrete. The addition of steam generators and TG called for a revised layout vis-à-vis Rapsodie layout.

Rapsodie was located in a highly seismic zone and was designed for high seismicity. Considering low seismicity at Kalpakkam, no detailed analysis was warranted for FBTR. This was indicated to the safety authorities (DAE-SRC) and clearance was obtained for construction of FBTR.

The civil structures were designed by Development Consultants Private Limited, a Calcutta based consultancy firm with rich experience in project design. The design of the RCB dome with ceramic heat shield was the first of its kind in India. RCB is supported on hard rock which was struck at a depth of ~15 metres. Being located in an area where water table is always high, the design had to ensure no water logging around the primary sodium cells. The RCB was hence designed with an annular isolation wall to avoid water seepage from the sides. Criss-cross channels were provided between the base raft of the RCB and the supporting rock to continuously collect any water seeping through the rock and pumping out.



*The ceramic heat shield on the RCB dome was the first of its kind in India*

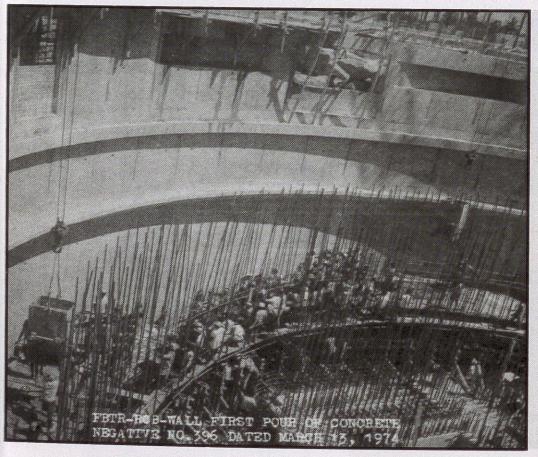


*The foundation stone was laid on 31st January 1972*

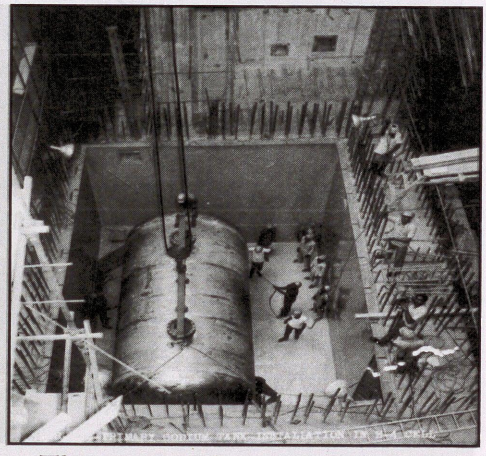



*Excavation for isolation wall Most of the construction was done manually*

The first breaking of ground was done on 31st January 1972. The first pour of concrete was done on 13th March 1973. Those were times when large scale mechanised equipment for excavation, earth moving and concrete pouring were yet to find their place in the civil construction industry in India. Most of the construction had to be done manually. The primary sodium storage tank, which had to be installed in its designated inner cell in RCB at -14.5 m elevation before casting its roof slab, arrived at site in March 1975. It came with a cracked nozzle weld due to improper support during transportation and was an eye-opener to the project team that due care should be taken during transportation. It was installed on 9th April 1975 and construction of cells was continued. Civil construction of all the buildings was completed by May 1977.

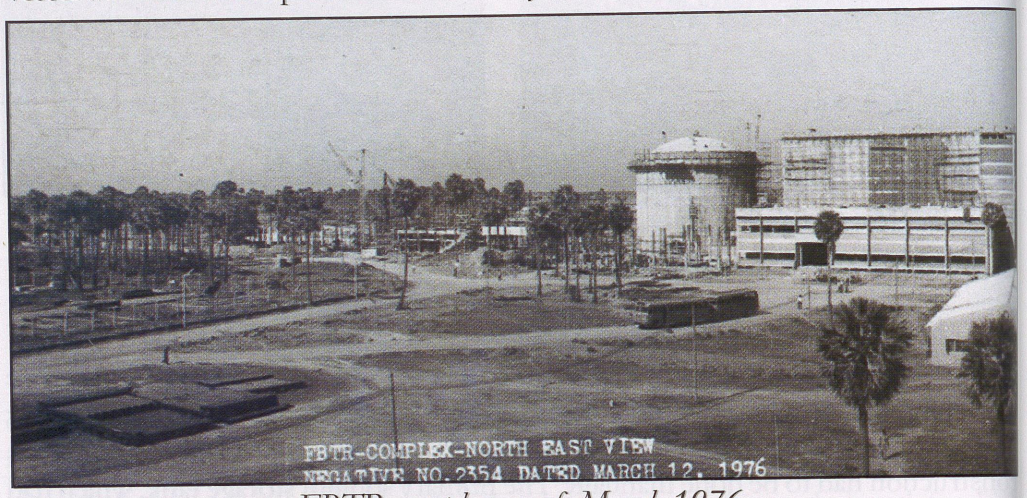



*The first pour of concrete on 13th march 1974*



*The primary sodium storage tank was installed in B-4 cell before building the superstructure*

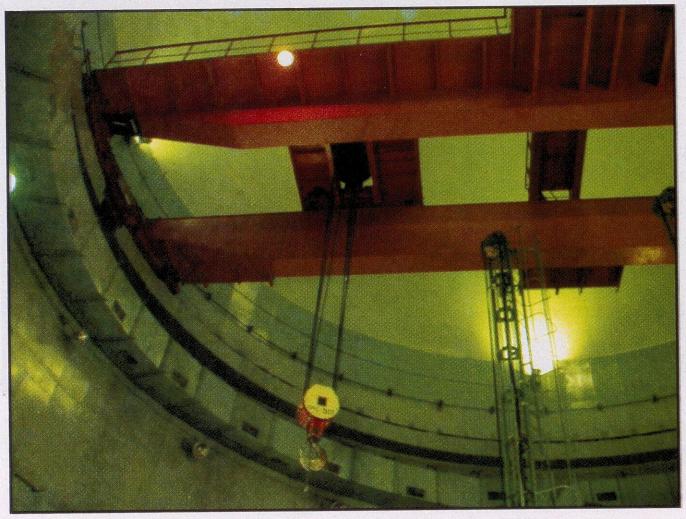
This was followed by a long wait for the arrival of the components, which were being fabricated in the shop floors of industries spread all over India. The RCB polar crane, considered the first of its kind in Asia, was installed in Feb 1978, and facilitated the erection of the steel vessel inside the A1 cell housing the reactor assembly. The steel vessel was an in-house product fabricated by the Central Workshop of RRC.




*FBTR complex as of March 1976*

**RCB Polar crane**

*In RCB a unique rotary electrically operated over head crane, which rotates on a circular track of 24 meter, top surface of the track being at an elevation of +14.5m, is provided with two trolleys of capacities 60 T (main) and 30 T (secondary) and each has two speeds. The command for the operation of the crane is possible from any of the five points at +1.00 m level in RCB.*

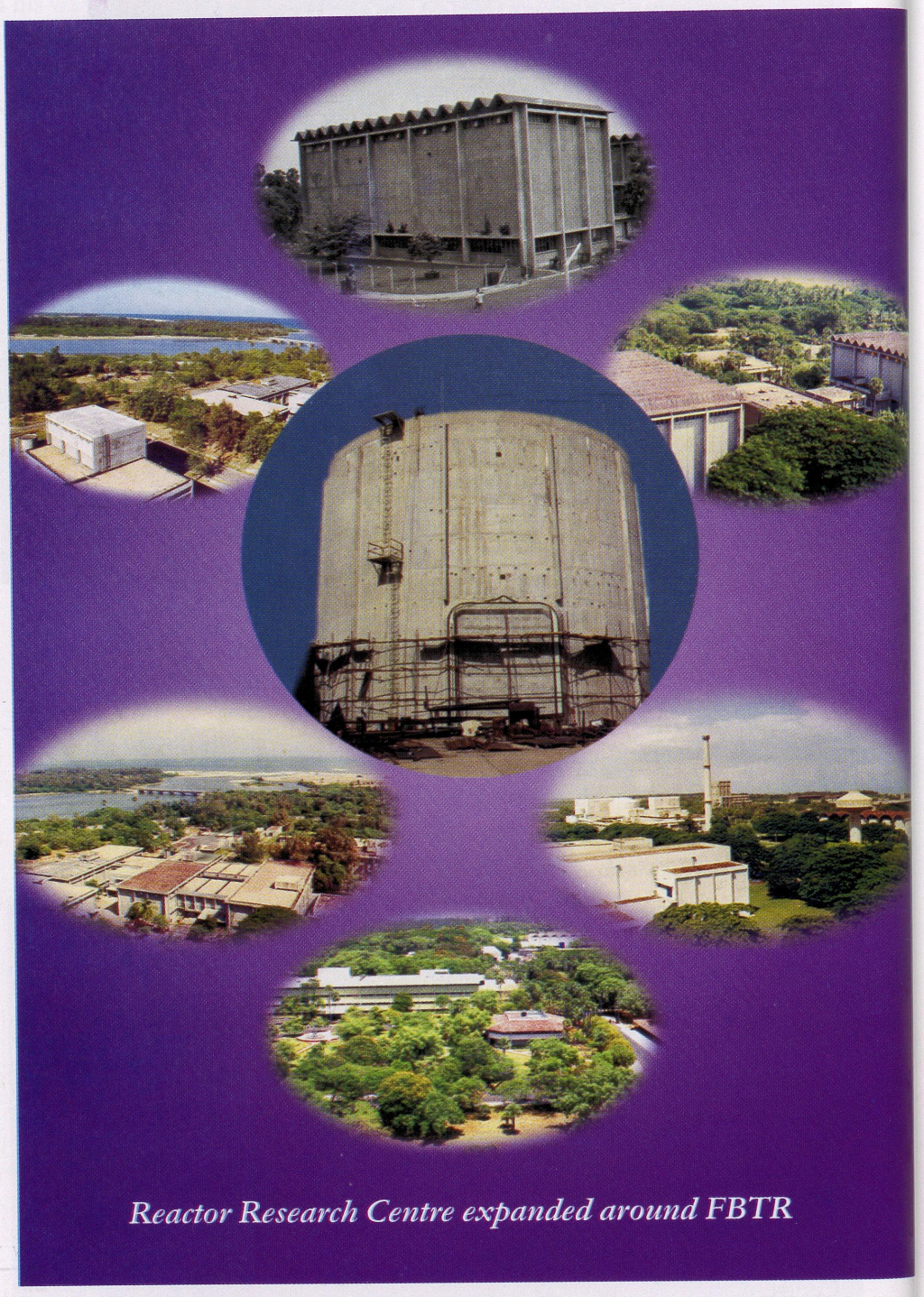


*The Polar (Rotary) Crane of RC*
*The first of its kind in India*

## The Growth of the Centre

Parallel to the construction of FBTR, the other facilities in the Centre were taking shape. Sodium loops were set up at the engineering halls for training the personnel in sodium handling and for testing the components and instruments in sodium. Water loops were set up for testing of sodium pumps, pressure drop devices etc. A facility for purifying commercial grade sodium to reactor grade was also put up. Concurrently, the other facilities in the Centre were also taking shape, which included the radio-metallurgy & radiochemistry laboratories and central workshops with necessary infrastructure. The radiochemistry laboratories were provided with facilities for the chemical analysis of metal samples and sodium to meet the immediate needs of the project. Work on the reprocessing development laboratory was also in progress.

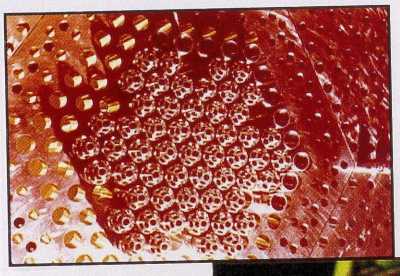
Since FBTR was provided with a steam water circuit, the safety analysis of Rapsodie was not fully applicable to FBTR. This resulted in an impetus to the in-house development of codes for thermal hydraulics and incident analyses. The reactor physics codes given by the French were fine-tuned. Shielding calculations were made afresh as the concrete used for shielding in FBTR was different from the high temperature concrete used in Rapsodie. Thus, FBTR provided the impetus for the growth of the analytical strength of the Centre. In those days, PCs were unknown. The nearest main frame computer available was at IIT, Chennai. The programmes had to be punched on cards, in card punching machines in the Centre and sent to IIT through the Despatch Rider of the Centre and the outputs of the previous day's run were collected. Relief came when the two TDC-316 computers meant for the Central Data Processing System of FBTR arrived at site. Till their installation at FBTR at a later stage, they were the work-horse for the computational activities of the Centre. These activities were later taken over by new main frame computers of the Computer Centre. As envisaged by the founding fathers, FBTR proved to be the catalyst for the overall growth of the Centre.

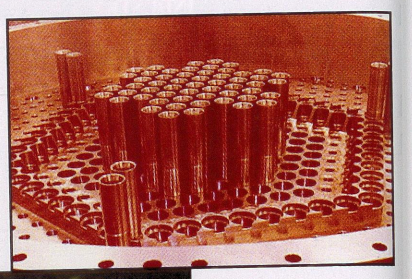


*Reactor Research Centre expanded around FBTR*

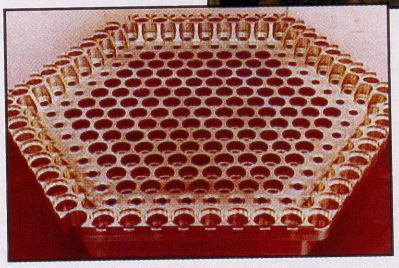
| INDUSTRY | SUPPLY |
|---|---|
| Alkali Metal Limited, Hyderabad | Sodium |
| Bharat Heavy Electricals Limited, Bhopal | Turbo Generator |
| Bharat Heavy Electricals Limited, Hyderabad | Reactor Vessel<br>Thermal Shields<br>Large Rotating Plug<br>Small Rotating Plug<br>Liquid Metal Seals<br>Vessel and Plug Support |
| Bharat Heavy Electricals Limited Tiruchirapalli | Steam Generators<br>Intermediate Heat Exchangers<br>Stainless Steel Piping (Primary)<br>Carbon and Alloy steel Valves |
| Development Consultants Pvt. Ltd. | Structural, Electrical and Ventilation Design |
| Electronics Corporation of India Ltd | Consultancy Services for Control & Instrumentation<br>Control and Instrumentation Packages<br>Central Data Processing System |
| Engineering Construction Corporation, Madras | Civil Works Main Plant Buildings |
| Everest Engineering Works, Coimbatore | Mixers and Pull-outs<br>Decontamination Vessels<br>Material Air Lock Bellows |
| Heavy Engineering Corporation, Ranchi | Special Steel Forgings for Block pile |
| Keltron, Trivandrum | 220 Volt Uninterrupted Power Supply |
| Kirloskar Brothers Limited, Kirloskarwadi | Charging and Discharging Flasks<br>Flask for Control Rod Drive Mechanisms and Level Detectors |
| KSB Pumps India Ltd. | Main Boiler Feed Pumps |
| Machine Tools Aids and Reconditioning, Hyderabad | Machined parts of Core Sub assemblies<br>Canal Valve Assemblies, Coupling & Sealing |
| Mukund Iron and Steel Works, Bombay | Piping |
| New Standard Engineering Co. Ltd. Bombay | 60T Polar Crane for Reactor Building |
| NGEF Ltd., Bombay | High Voltage and Medium Voltage Switch-gear<br>High Voltage and Medium Voltage Motors<br>Medium Voltage MCCs |
| Transformers & Switch Gears Ltd., Madras | 10MWA Main Transformers |
| Triveni Structural Ltd., Naini | Overflow Tank<br>Secondary Sodium Storage Tanks<br>Cyclone Separator<br>Cold Traps<br>Secondary Flask<br>Carriage Special Flask<br>Decay Pits<br>CI Blocks for Discharge Pots |
| Variety Engineers, Baroda | Shielding Blocks<br>Transformer Carriage No.2<br>Capsule Transfer Gripper<br>Steel Shielding Doors |
| Walchandnagar Industries Ltd., Machine Tool Division, Pune | Control Rod Drive Mechanisms<br>Core Cover Plate<br>Plug Drive Mechanisms |
| Walchandnagar Industries Ltd., Walchandnagar | Control Plug<br>Anti-explosion Floor<br>Fuel Handling Canal<br>Sodium Pumps |

*Major Indian industries spread all over India participated in the construction of FBTR. Indigenous component, even in the seventies, was ~80%.*







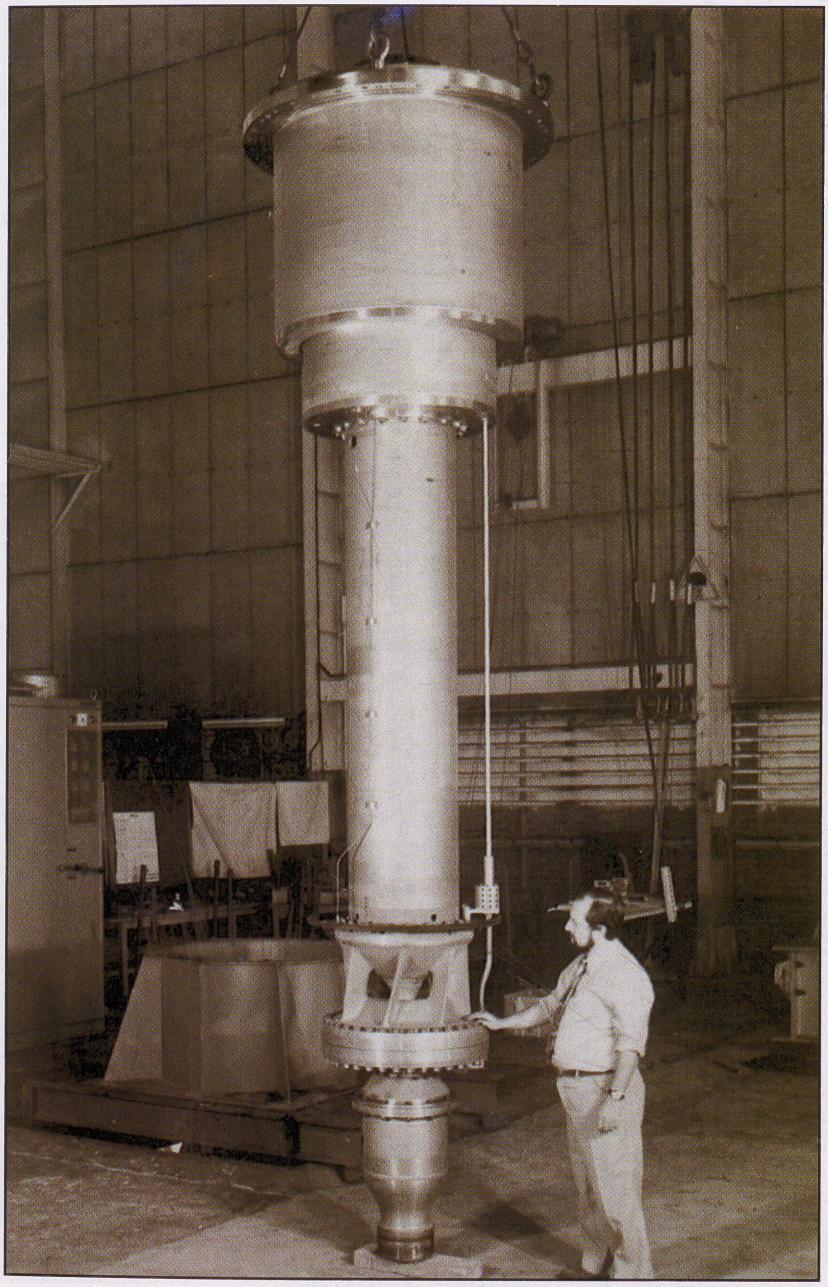




*The Grid Plate Came from France*

## Delays in the Fabrication of Components

The steel vessel of reactor assembly was fabricated at Central Workshop of RRC. Material and Personnel Airlocks were fabricated at Central Workshops, BARC. The fabrication of all other components had been outsourced.

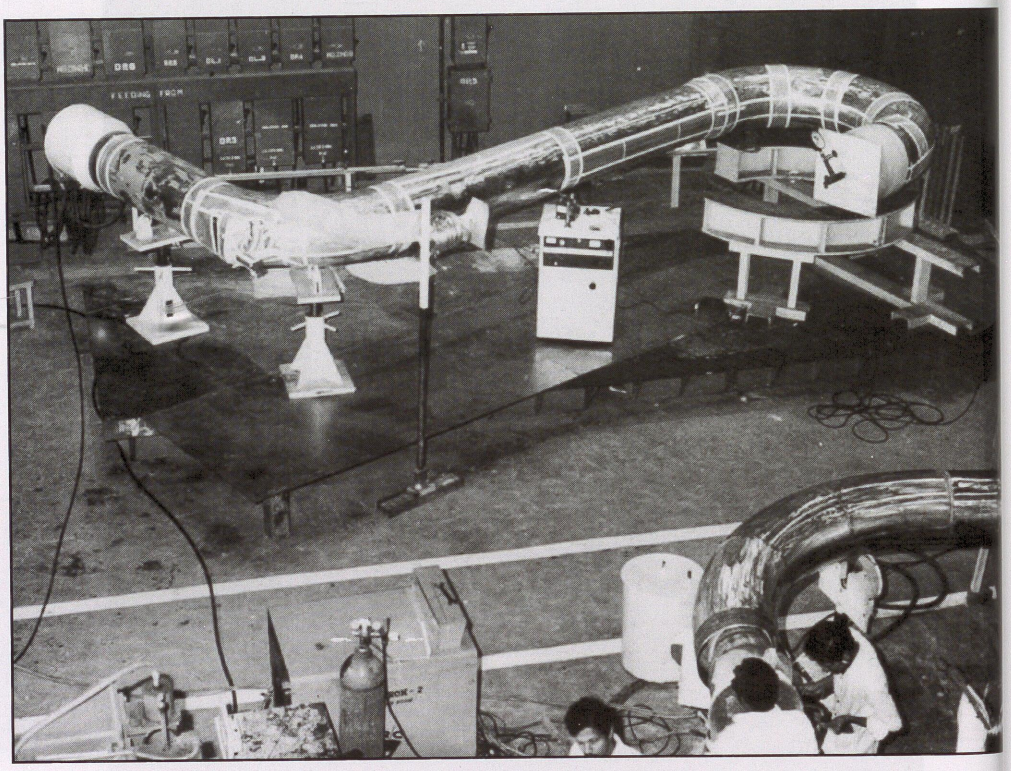


*One primary sodium pump came from France*

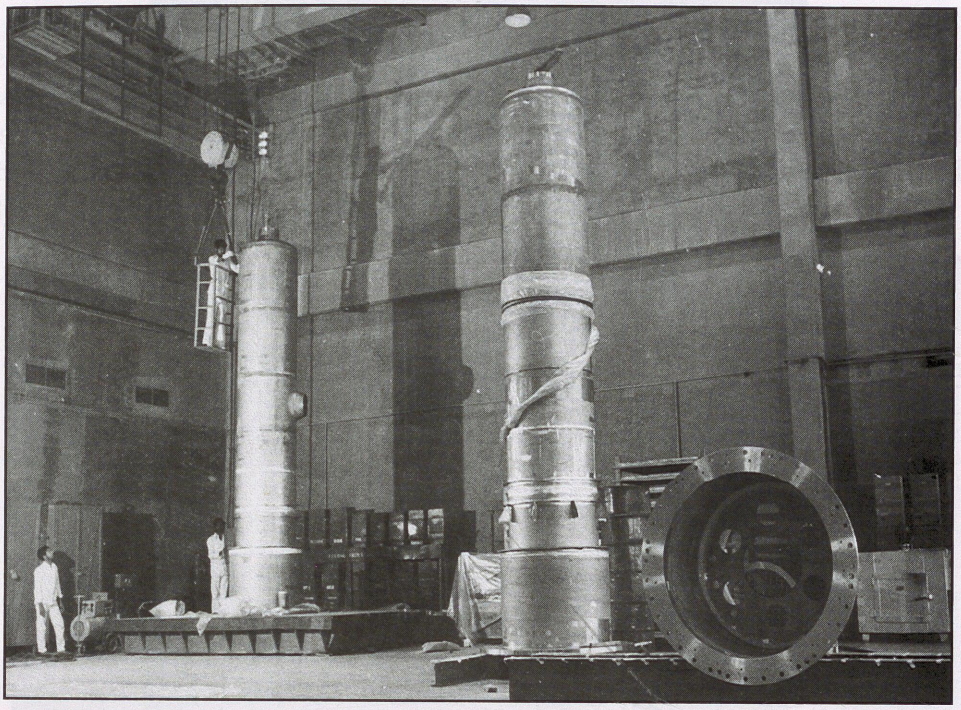
## The Project Years

Indian industry in the seventies was just coming to terms with ASME Sectio III (Rules for Construction of Nuclear Power Plant Components). FBT specifications being more stringent than ASME-Sec III, there was reluctance on th part of many industries to undertake manufacture of FBTR components.

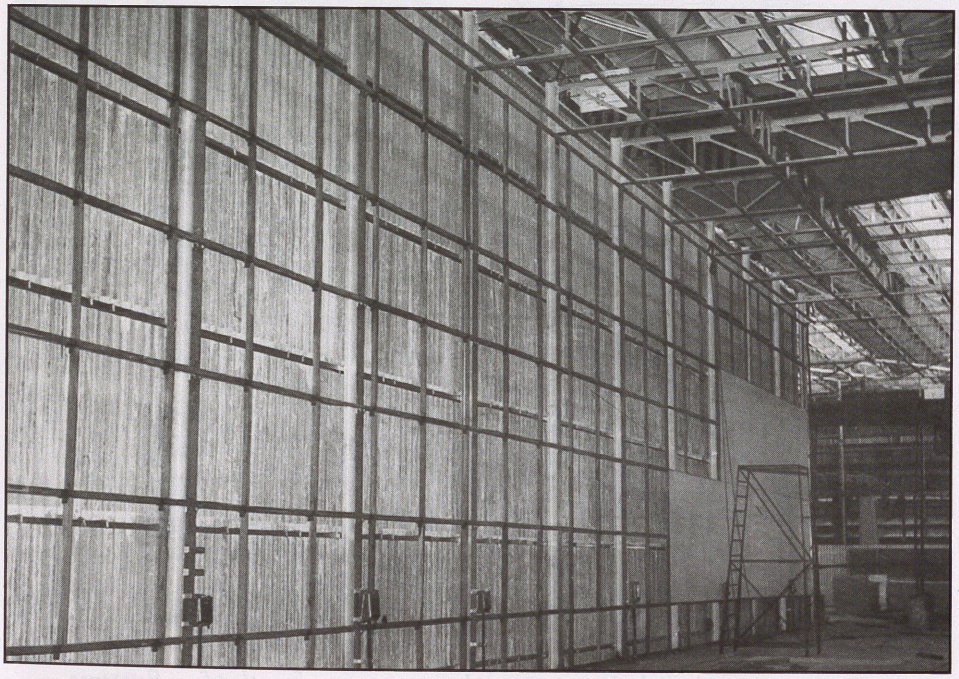
Most of the major components were entrusted to public sector undertaking by persuasion. Unlike pressure vessels, fast reactor components are thin walle slender and long with stringent shape tolerances. Distortion is known to be one o the major problems in stainless steel welding. The long, thin and slender nature of th FBTR components aggravated the problem. Control of local and global distortion was a major agenda for the fabricators. Unlike conventional systems, fast reacto circuits cannot be flushed with water during commissioning. Hence, cleanlines during manufacture had to be strictly enforced. Most of the shop floors had to b equipped with nuclear clean halls. During welding, the purge gas 'dams' had to b meticulously numbered and inventory maintained to ensure that nothing is left insid the systems.



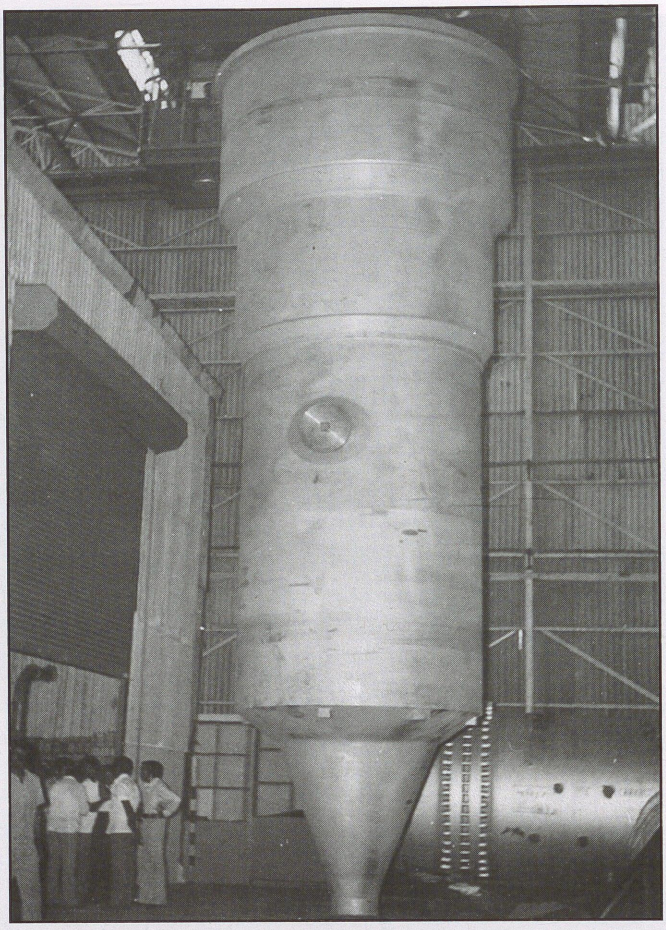
*Sodium Inlet Pipe*

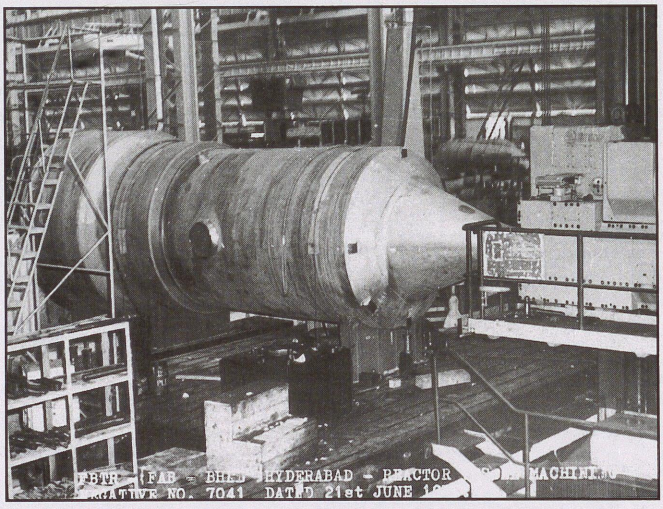


*IHX shell inside nuclear clean hall*



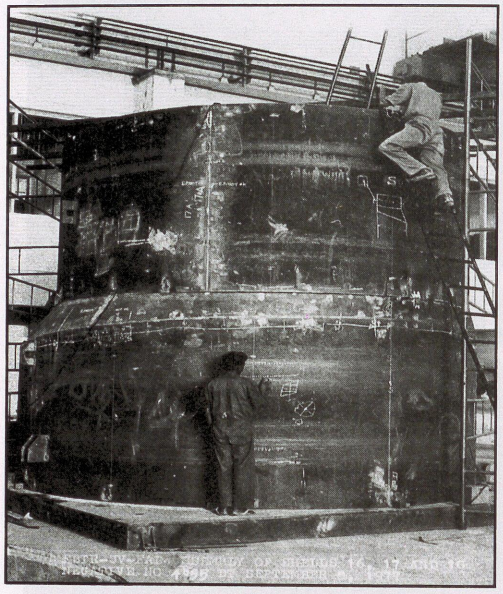
*Nuclear Clean Hall*

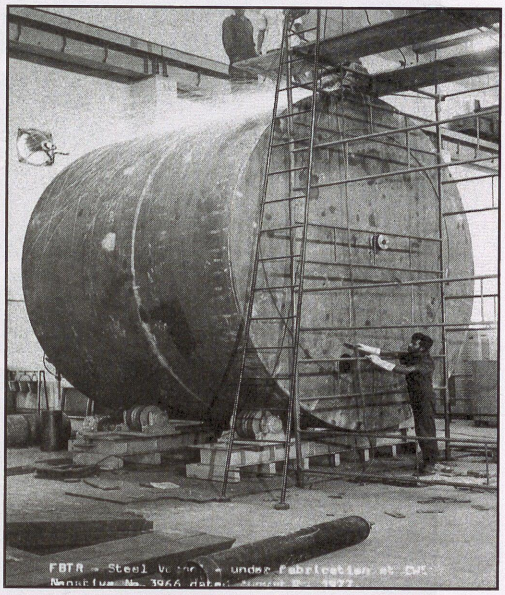


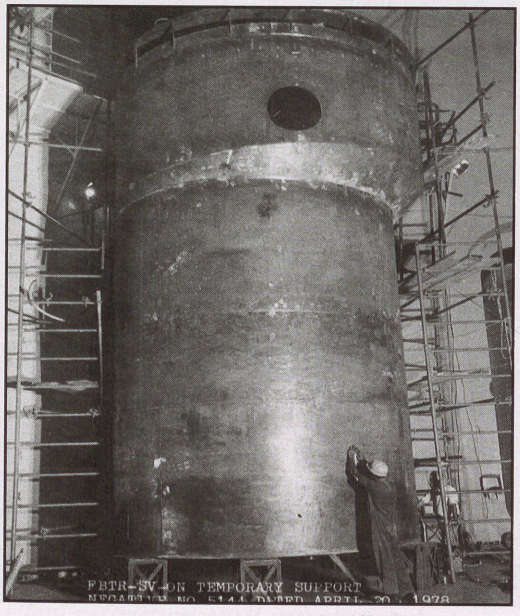


*Reactor vessel with 1 mm straightness over a length of ~8metres*

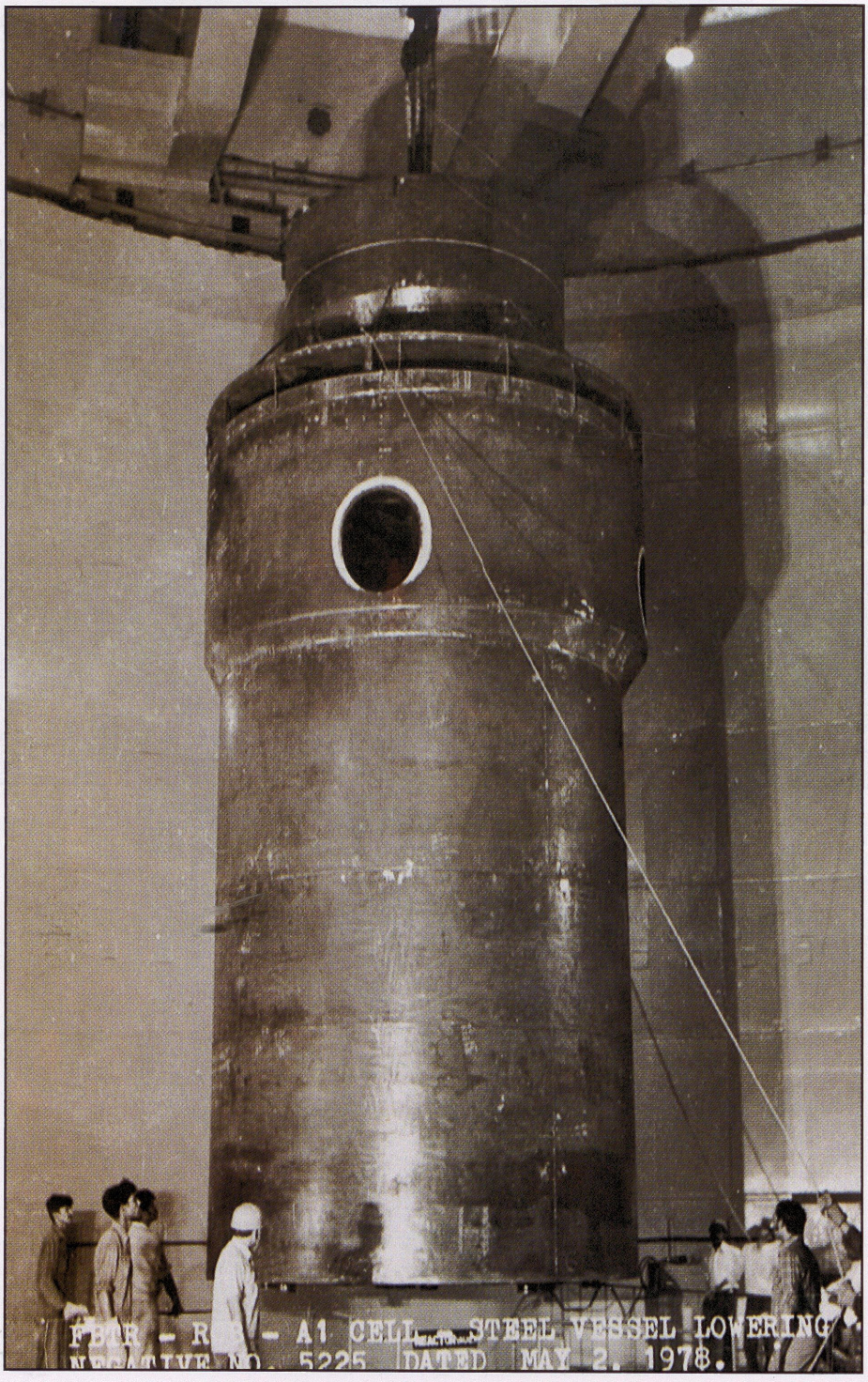
The psychological backdrop under which the project team had to work was no less daunting. The embargo in the wake of Peaceful Nuclear Explosion in 1974 was casting a shadow on the imported supplies. Large scale exodus of trained craftsmen – especially welders, fitters, electrical tradesmen and engineers from the industries in the wake of the Gulf boom was causing a lot of delays. All these resulted in long slippages in the scheduled delivery dates of almost all the components.







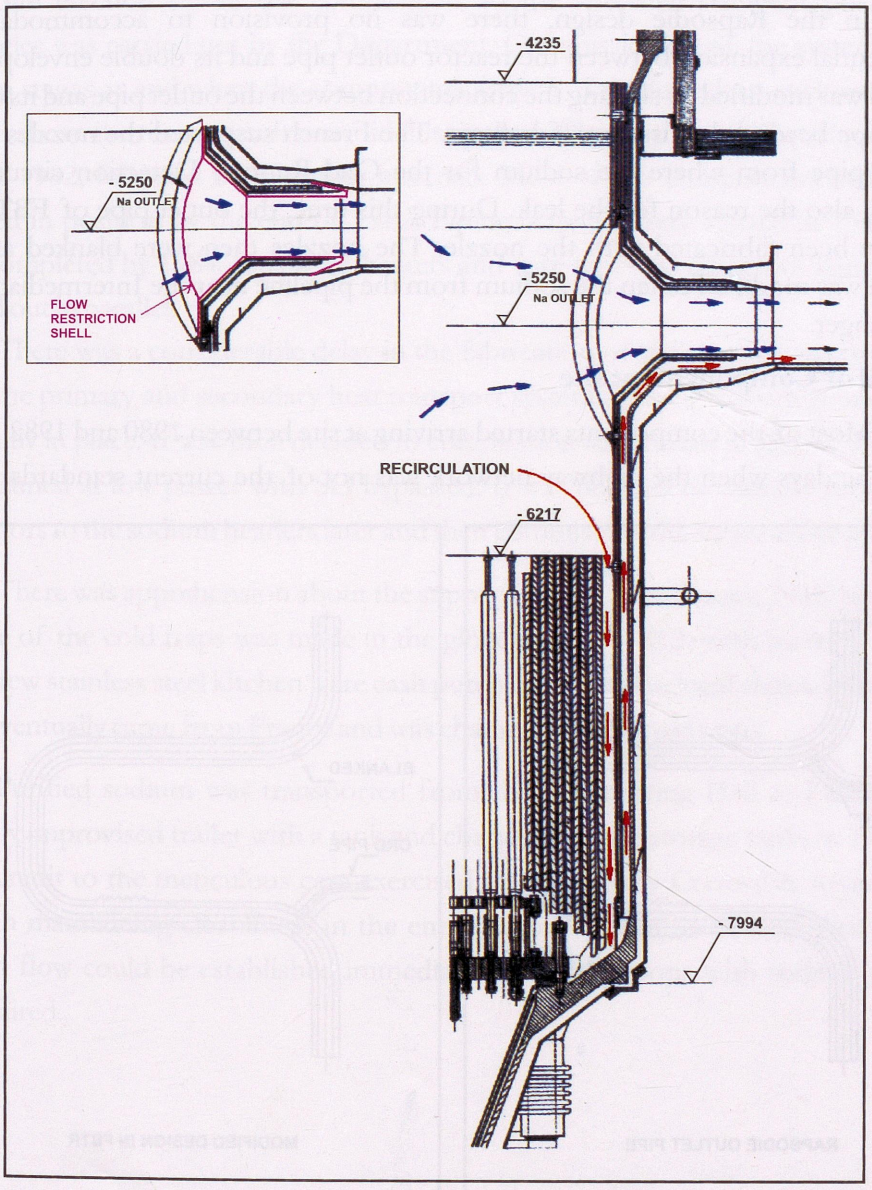

*The steel vessel was fabricated at Central Workshop of RRC*




*The steel vessel erection inside the A1 cell*

## Design changes

In 1978, when most of the reactor components had been manufactured, came the news of sodium leak in Rapsodie and the decision to decommission the Rapsodie reactor. This called for several last minute design modifications in the primary sodium piping.



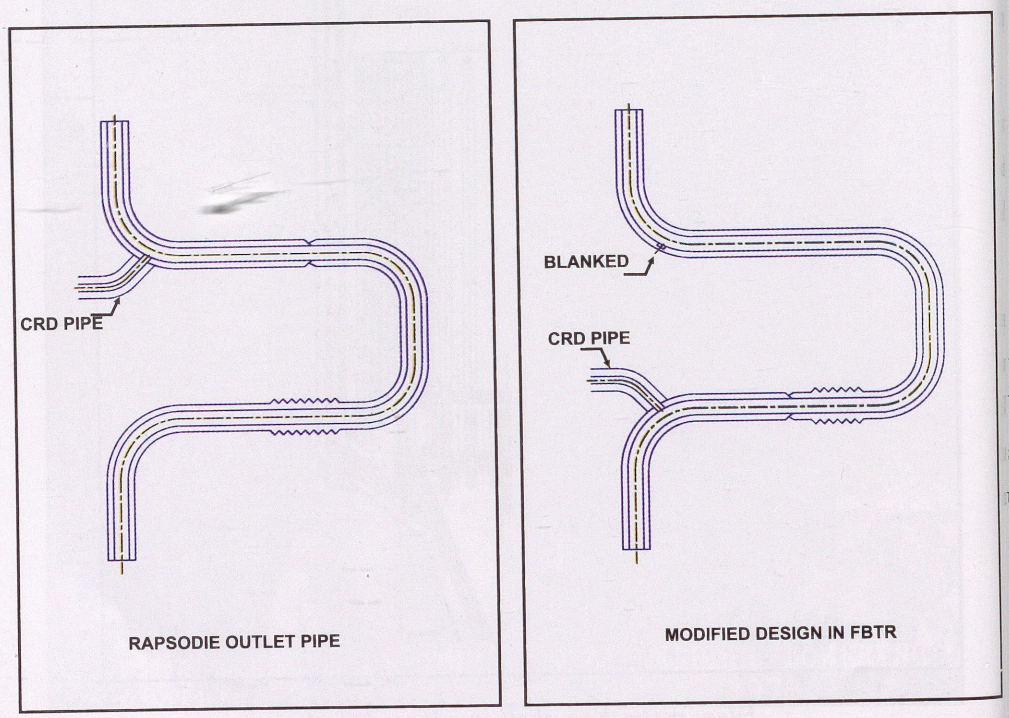

*Flow restriction shell incorporated at reactor vessel outlet in FBTR*

In Rapsodie, the main flange of the reactor vessel, which supports the core was found to be close to the outlet temperature of 515°C during operation though it was designed for an inlet temperature of 400°C. The hydraulic mock-up studies showed this was caused by re-circulation of hot sodium between the reactor vessel and thermal shield due to ejector action near the outlet pipes. In consultation with the French, a flow restriction shell was added in FBTR design to block the ejector action.

In the Rapsodie design, there was no provision to accommodate the differential expansion between the reactor outlet pipe and its double envelope. The design was modified by shifting the connection between the outlet pipe and its double envelope beyond the first set of bellows. The French suspected the nozzles of the outlet pipe from where the sodium for the Clad Rupture Detection circuits was tapped also the reason for the leak. During this time, the outlet pipe of FBTR had already been fabricated with the nozzle. The nozzles then were blanked and the design was modified to tap the sodium from the pipeline near the Intermediate Heat Exchanger.

## Arrival of Components at site

Most of the components started arriving at site between 1980 and 1982. Those were the days when the highway network was not of the current standards. Road




*Outlet pipes of Rapsodie and FBTR*

were narrow and bumpy. Many bridges and culverts were weak. Many over-dimensional components had adventurous journeys. All the critical components were transported from the manufacturers' works by departmental trailers, following the route-maps, safety and security restrictions which were personally decided by the Principal Project Engineer, with speed limitations decided by the design group.

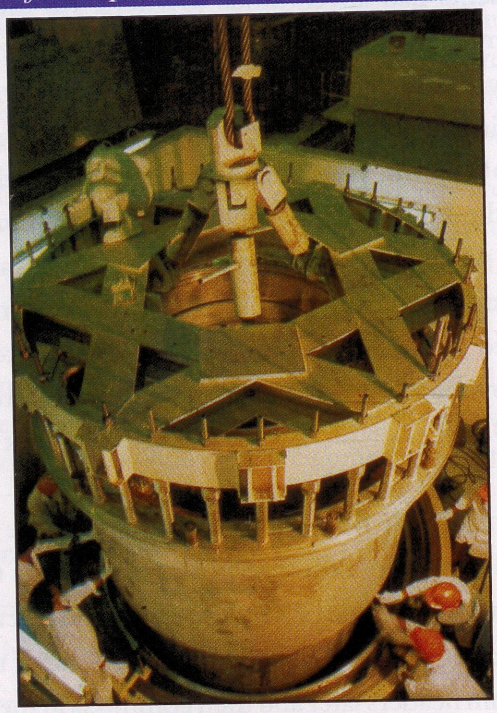
Site erection of all the components to the specified stringent positional accuracies was carried out by the Department. Commissioning of the systems was done in stages as and when the components arrived. A nuclear clean enclosure was erected above the reactor vault to avoid dust ingress. The reactor vessel arrived at site in June 1982, followed by the other internals. Most of the components had been installed in position by the end of 1982. Piping activities were started in 1980 and were completed by 1984. Laying of heaters and wire type leak detectors were being carried out in parallel.

There was a considerable delay in the fabrication of the steam generators. By 1984, the primary and secondary heat transport systems, sans the steam generators, were fully in place. It was then decided to commission the reactor in phases. Phase-I was planned at low power with SG bypassed. It was decided to connect the steam generators to the sodium headers later and then commission the steam water system.

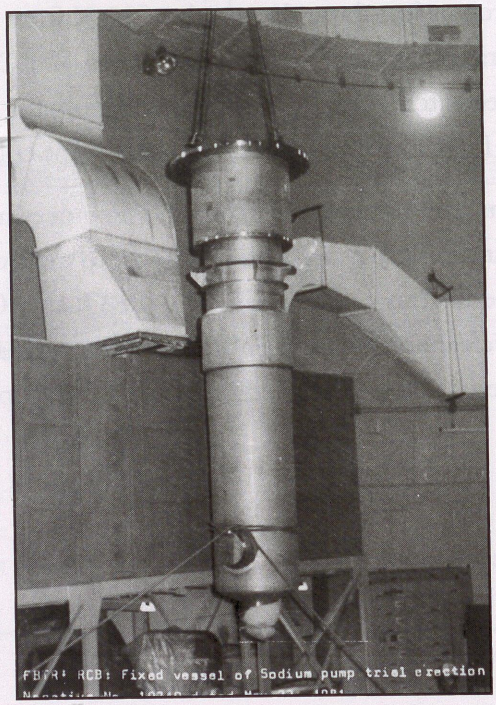
There was apprehension about the supply of NaK from France. NaK required for one of the cold traps was made in the glove boxes of RCL with no tools other than a few stainless steel kitchen ware cash purchased from the local shops. However, NaK eventually came from France and was charged into the cold traps.

Purified sodium was transported from the Engineering Hall to FBTR in a specially improvised trailer with a tank and charged into the storage tanks in 1984. It was a credit to the meticulous care exercised by the Quality Control & Inspection Wing in maintaining cleanliness in the entire piping system so that uninterrupted sodium flow could be established immediately in all the loops with sodium purity unimpaired.

*Erection of components was done departmentally*

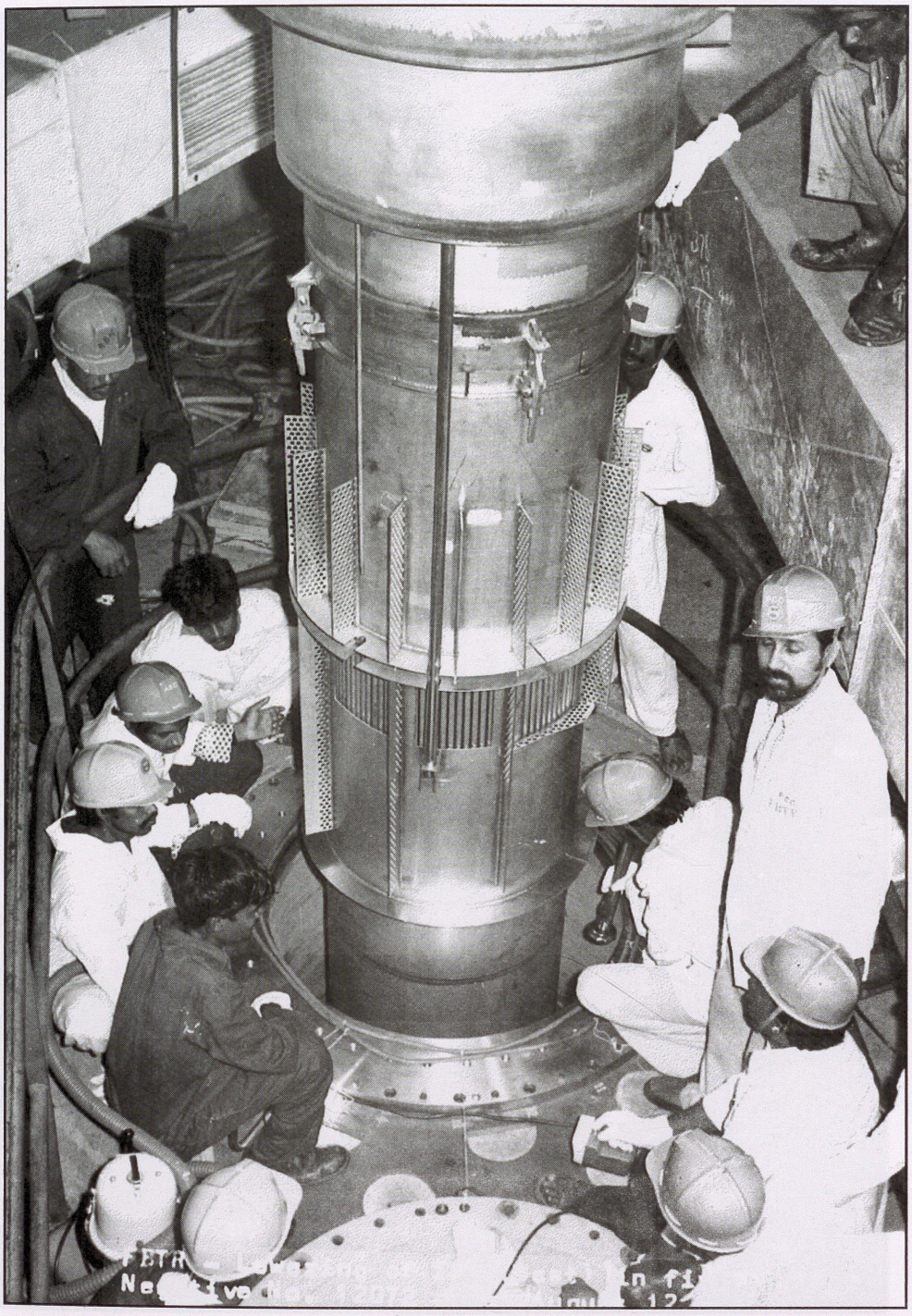


*LRP*



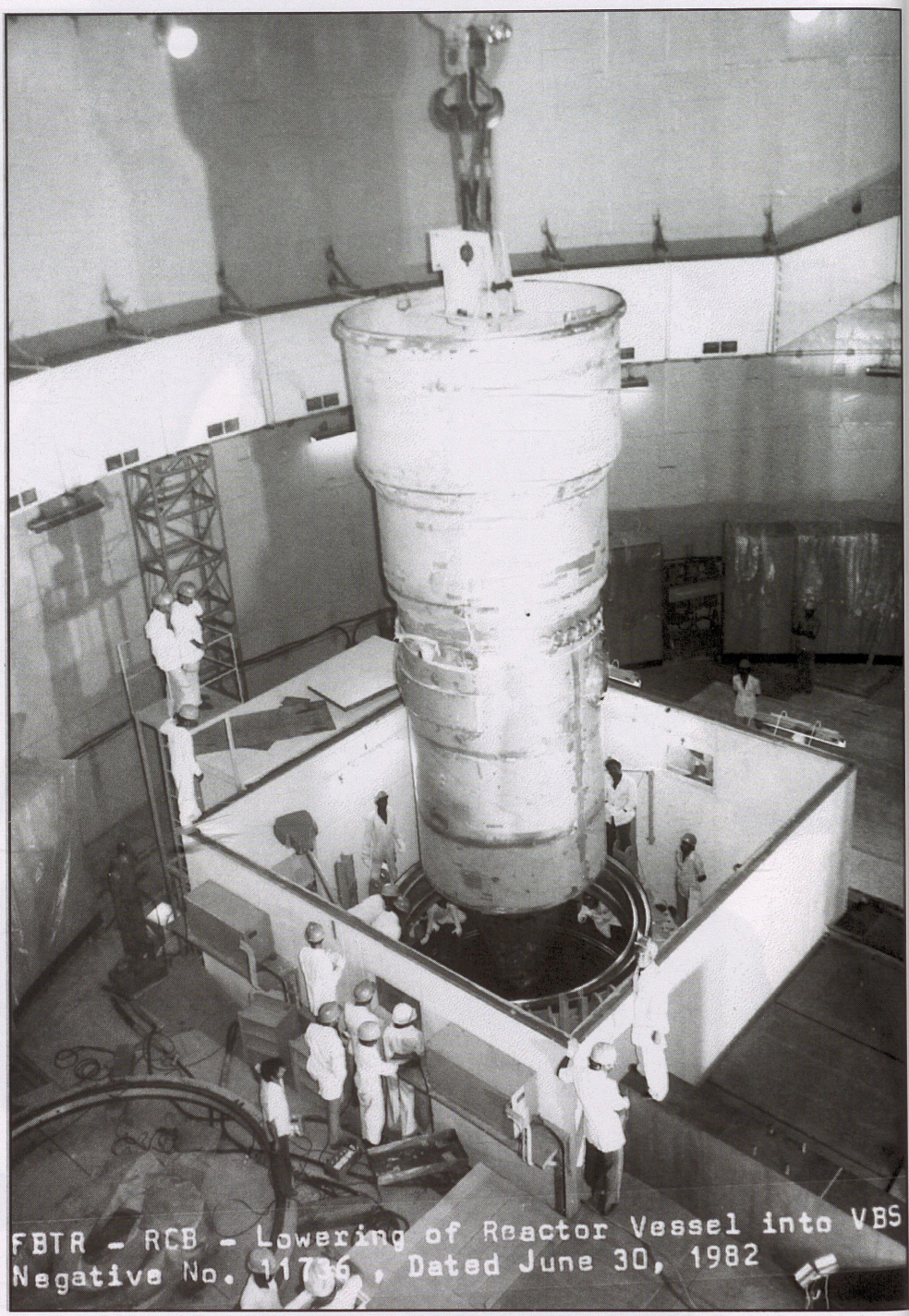

*Pump*

*Erection of components was done departmentally*



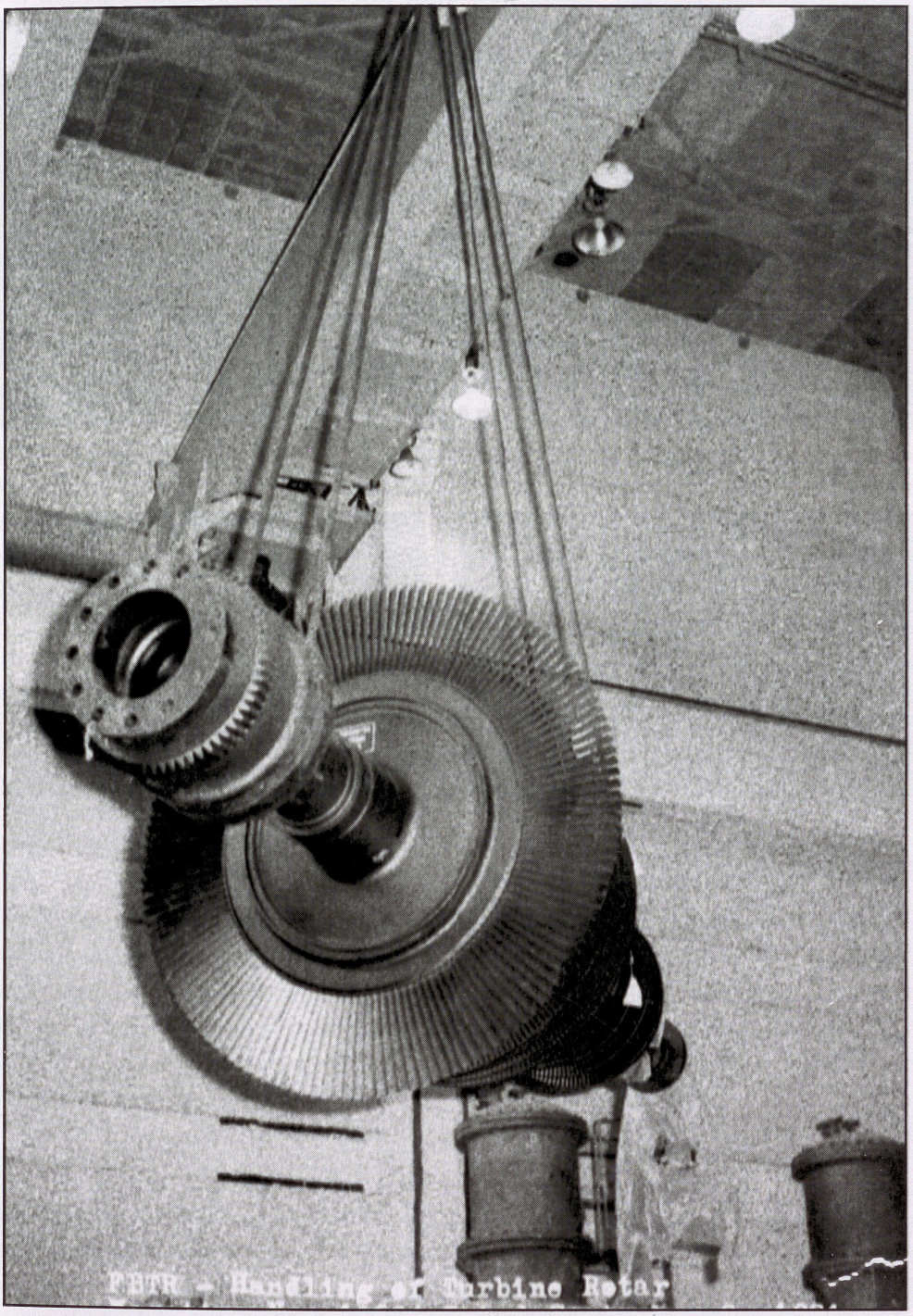
*Intermediate Heat Exchanger*

*Erection of components was done departmentally*



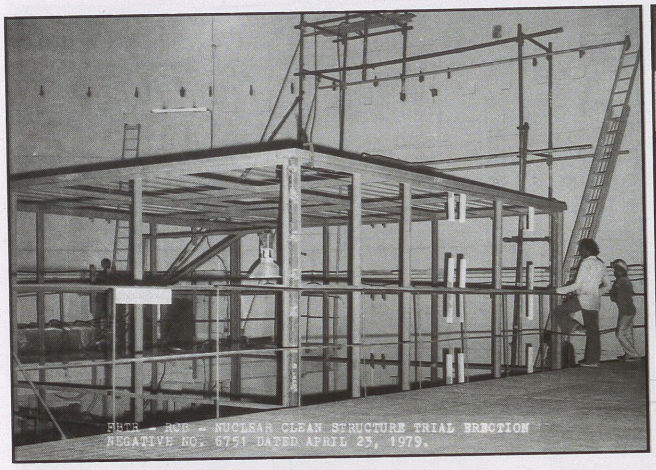

*Reactor vessel*

*Erection of components was done departmentally*

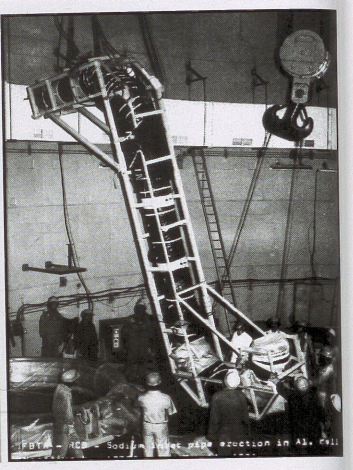



*Turbine rotor*

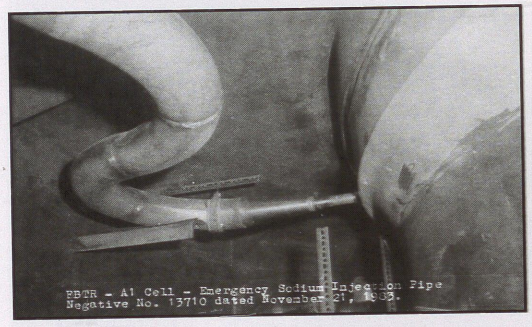
## *Erection of components was done departmentally*




*Nuclear Clean Enclosure above A1 cell*



*Inlet pipe*






*Complicated welding within the A1 cell*

### The Indo-French Agreement

The French side continued to fulfil all their obligations as per the agreemen The visit of the design team in 1969 to Rapsodie was followed by engineers fro operations and construction. During the fabrication of components, engineers fro the concerned Indian industries also visited the respective works in France and g trained. Engineers from Creusot Loire, the French industry which manufactured t major Block Pile components of Rapsodie, visited the works of BHEL at Trichy Hyderabad and clarified the doubts of shop floor engineers.

French engineers also had periodic stints at site as consultants. The excelle training manuals of FBTR owe their credit to some of them. Till the Rapsod reactor was shutdown, the weekly operation bulletins were received from Cadarach The French side cautioned us about the possible design weaknesses that could ha led to the sodium leak from the Rapsodie reactor block. They invited the Indian tea

to witness the investigations which were being carried out in Rapsodie and reviewed our proposed design modifications to overcome these in FBTR. During the commissioning stage of FBTR in 1984, some problems were referred to them and they had responded. After the 1987 fuel handling incident in FBTR, an inspection tool called 'Tube de visee' was supplied by them.

*Rapsodie specifications were meticulously followed by the project engineers to the letter. There were, in those days, cross-word wizards in the project. When confronted by any deviation from specification which needed a decision, the solution was 'a dose aspirin'- the anagram of 'as in Rapsodie'.*

## The Switchover to Small Carbide Core

The agreement between India and France did not include the supply of fuel. Rapsodie used MOX fuel with 30% $PuO_2$ and 70% $UO_2$, with the latter enriched to 85% in $U^{235}$. In order to avoid dependence on imports, it was proposed to go in for a MOX fuel with natural uranium and more Pu. Even when the design team was in Cadarache, studies were conducted on MOX fuel with 70% $PuO_2$ & 30% $UO_2$ (natural uranium). These studies indicated poor performance in terms of the Linear Heat Rating and compatibility with sodium. BARC had the capability to make MOX fuel and therefore, it was felt that fuel could be made in India with indigenous Pu and imported enriched U, pending a decision on importing enriched uranium.

### *Carbide Fuel*

*In nuclear reactors, the fuel (Pu, U) is deployed in various chemical forms- metal, metal alloy, oxide, carbide, nitride, carbo-nitride, silicide, etc. In a fast reactor, the fissile atom density (number of heavy atoms per unit volume) plays a major role in the breeding ratio. The higher the fissile atom density, the higher is the breeding ratio.*

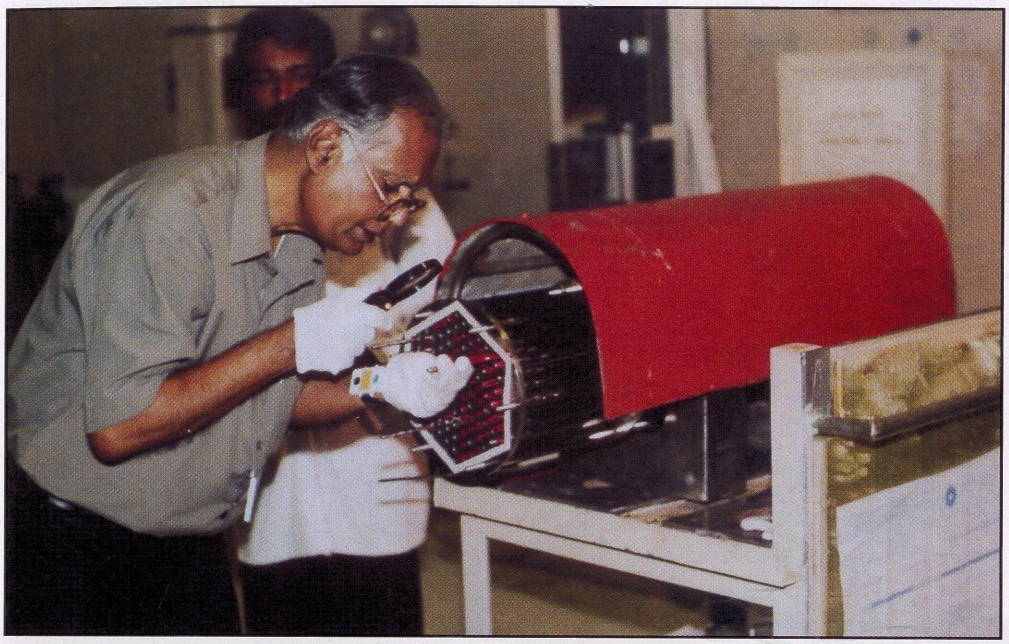
*Fuels in the form of oxides are relatively easy to fabricate, but they give low breeding ratios. Carbides have higher fissile atom density and hence better breeding ratios, but are difficult to fabricate. They are highly pyrophoric, i.e. the tendency to catch fire in air. The fuel is fabricated by the powder metallurgy route, i.e. the powder is pelletised, sintered and the pellets are encapsulated within the clad tube. The entire operation has to be done under inert gas atmosphere. During the cutting of the fuel, as during Post-Irradiation Examination in hot cells or during reprocessing, inert cover has to be maintained. The hot cells of Radio-Metallurgy Laboratories adjacent to FBTR are hence provided with high purity nitrogen cover. The fuel chopping in the reprocessing facility at IGCAR is also done under inert cover.*

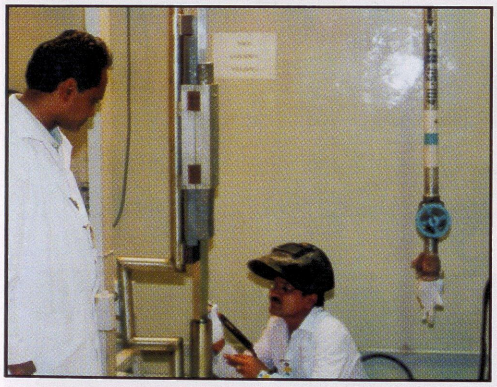
## The Project Years

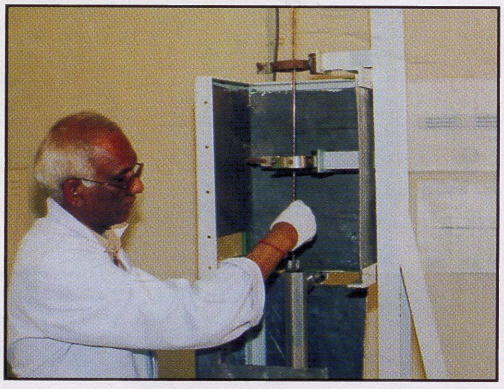
With the changed international scenario after the Pokhran-I, it became clea that we would not be able to import enriched uranium. Carbide fuels had bee tested, on a limited scale, in many fast reactors all over the world e.g. Rapsodie Fortissimo and Phenix in France, DFR in UK and EBR-II & FFTF in USA. I Russia, a Uranium carbide core was in operation in BR-5 reactor from 1965 to 197 and achieved a maximum burn-up of 6.2 at%. In the BOR-60 reactor in Russi several test subassemblies were irradiated to high burn-up levels. Mixed carbide fue however, had not been used in any of the reactors as a driver fuel. A bold decision wa then taken by Dr. Raja Ramanna to go in for a Pu rich carbide fuel as the driver fuel fo the first core of FBTR. It was decided to redesign the core for carbide fuel and wit minimum Pu inventory, considering that reprocessing experience with carbide fue was not then available. Reactor physics calculations indicated that the critical size fo the core with minimum inventory will need 21±2 subassemblies with a fuel of 70% PuC + 30% UC composition (designated as MK-I).

Studies conducted at Radio Metallurgy Division, BARC and Radio Chemistr Laboratory, IGCAR concluded that this fuel is compatible with sodium. However, i would not be possible to operate at the high LHR levels typical of carbide fuels as i has a lower melting point. An operating LHR of 250 W/cm was conservatively se based on out-of-pile studies. With this LHR value the achievable reactor power wa 10.2 MWt for the small core. It was decided that the MK-I core would b progressively replaced by fuel of composition of 55% PuC+45% UC (MK-II) t give a larger core capable of closer to the original design power of 40 MWt. A facilit for the fabrication of carbide fuel pellets and encapsulating them within clad tube was set up at RMD, BARC. All the hardware required for the core was manufacture by Nuclear Fuels Complex, Hyderabad. Fuel pins of MK-I composition wer fabricated by RMD, BARC. By April 1985, fuel pins for the first core were available site. Fuel pins were assembled into subassemblies at FBTR by a team from NFC.

The decision to go in for a carbide core, in retrospect, has proved to be challenge converted into an opportunity. We have been able, in a single go, to perfe the technology of an advanced fast reactor fuel, including closing its cycle. It may b recalled that right from the beginning, in the Indian context, the emphasis has bee on fuels with high breeding ratios. MOX fuels have been considered only as a sto gap measure. The rich experience with the carbide fuel has also put us in good stea for managing the fuel cycle of the conventional MOX fuel being deployed for th first few commercial fast reactors, including PFBR.







*Fuel pins being assembled into subassemblies in FBTR*

## Commissioning

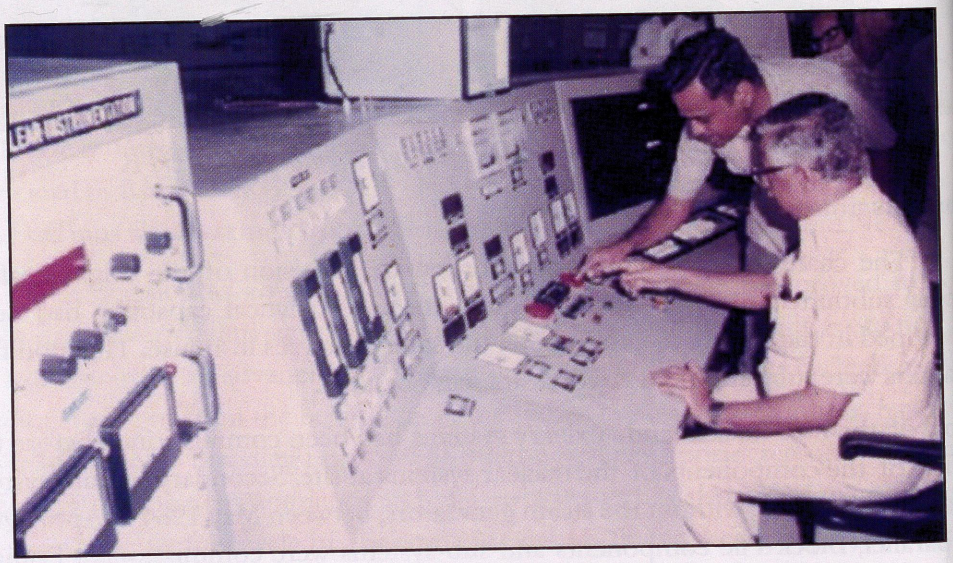
The change over to the small core required revision of the original safety report submitted to DAE-SRC. By then, enough analytical capability had been developed in the Centre to carry out all the safety analyses in-house. The modified chapters were submitted and safety clearances obtained.

Most of the non-nuclear and auxiliary systems had been commissioned before the arrival of the components of the nuclear systems at site. Secondary sodium system was commissioned, without the steam generators, between May 1984 & April 1985. In parallel, Block-Pile components and sub-systems were commissioned. Primary sodium storage tanks were filled in December 1984. Non-fissile SA and dummy fuel

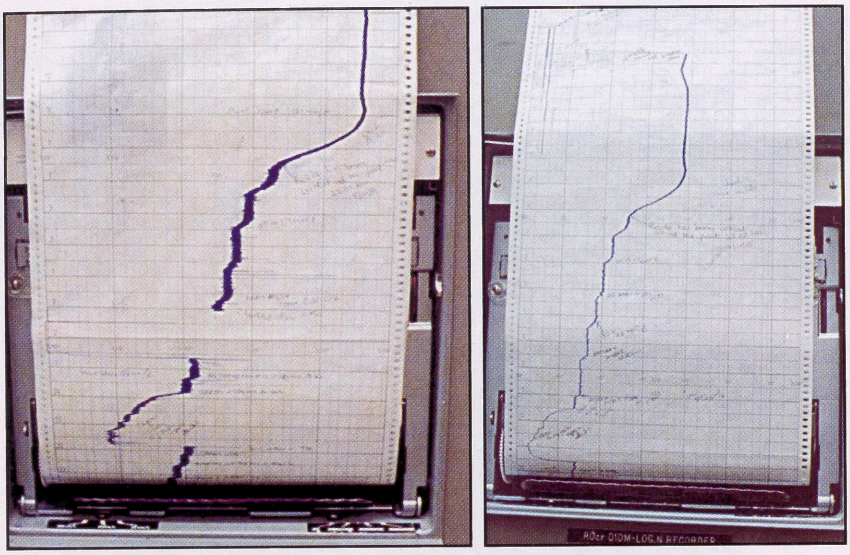
SA were manually loaded in the core and the primary system was filled with sodium in July 1985. The count-down for criticality began with the start of loading of fuel subassemblies on 2nd October, 1985. Dr. Raja Ramanna, the then Chairman, AEC was personally at the console when the first criticality was achieved at 2210 hours on 18th October, catapulting India into the select few countries with fast reactors. The completed cost of the project was 791.2 millions.



*Awaiting the critical moment Dr. Raja Ramanna & Dr. M.R. Srinivasan. Seen along are Mr. Bhoje, Dr. Shankar Singh, Dr. Lee*



*Dr. Raja Ramanna was at the console for the First Criticality*



*The First Criticality Charts*



*Briefing at the Conference Room Mr. C.V. Sundaram, the then Director, RRC, Dr. Raja Ramanna, Mr. N. Srinivasan (the first Director of RRC), Dr. M.R. Srinivasan & Mr. S.R. Paranjpe. Among those seated behind are Mr. Bhoje (Design engineer and Task Force Leader for Reactor Assembly), Dr. Placid Rodriguez and Dr. P.K. Iyengar*

*"The rigidity with which our scientists struck to specifications and commissioned them, comparable in quality to any in the world, showed that given the drive, our scientists are capable of reaching great heights. By splitting the atom, we have released boundless energy. It must be our task to see that this energy is used for the furtherance of our civilisation and not misused for the destruction of human kind. Our abhorrence to nuclear weapons is total" said Rajiv Gandhi, the then Prime Minister of India, while dedicating FBTR to the nation.*

THE HINDU
India's National Newspaper
Printed at Madras, Coimbatore, Bangalore, Hyderabad and Madurai

# FBTR goes critical

MADRAS, Oct. 18.

The 14 MW Fast Breeder Test Reactor (FBTR) at Kalpakkam went critical at 10.02 p.m. on Friday demonstrating emphatically the nation's success at a technology that promises to dominate power generation in the next century.

India thus became the seventh nation in the world to get a fast breeder reactor working, which Dr. Raja Ramanna, Chairman of the Department of Atomic Energy (DAE), who was present on the occasion, said, was especially creditable because India had done it alone, without help from others.

The progress towards criticality was generally smooth save for a false alarm which extended the countdown perhaps by an hour.

All the indigenously developed plutonium-uranium carbide fuel rods—there were 22 of them—were loaded on Wednesday night itself and the neutron activity within the reactor was kept in check till today, as per the mandatory procedure, with the help of control rods made of boron carbide, a neutron-absorbing material.

This evening, as top officials of the DAE joined the scientists at the Reactor Research Centre (RRC) to witness a historic moment, the control rods were raised in a deliberate and slow procedure. In response, the pace of neutron activity rose first gradually and then in quicker bursts to a level that produced sufficient number of atomic fissions to sustain a nuclear chain reaction.

Fission or splitting of the plutonium atom provides the heat that will be eventually harnessed for electricity generation. Initially the FBTR will run on low power to enable scientists to test the various systems and to perform physics experiments. Full power is likely to be reached some time next year.

The fast reactor is different in many ways from the conventional heavy water reactors such as the two 235 MW units next door at Kalpakkam. Plutonium is the fuel used here rather than natural uranium which is the staple for a heavy water reactor. The higher fissile content of the plutonium fuel helps engineers dispense with the heavy water that is required in a conventional reactor to moderate the speed of neutron production so as to ensure an adequate number of fissions. And because of the higher neutron yield from plutonium there is more fresh plutonium generated than is consumed during reactor operation.

The fast breeder reactor is reckoned as the logical follow-up to the heavy water reactor since its fuel, plutonium, is only a by-product of power generation at the heavy water reactor, which essentially converts the non-fissile content of the natural uranium fuel—the isotope U-238—into plutonium.

## Persistent research

For the team of scientists and engineers led by Mr. C. V. Sundaram, Director, Reactor Research Centre, Kalpakkam, this evening's performance was the culmination of some 13 years of persistent research and development of a technology that the six other nations in the world which already have it were hardly willing to divulge.

Admittedly, India did have the benefit of a French design when the construction of the FBTR started in 1972. But a significant number of modifications and adaptations were required and these were carried out indigenously.

One of the major departures from the French design was the nature of the fuel itself. The enriched uranium-plutonium oxide fuel that France used was not suitable, for India lacked an uranium enrichment facility. Scientists therefore opted for a plutonium-uranium carbide combination. No other country has yet tested out a fuel core entirely made of carbide, and as such the FBTR's performance is expected to be studied world wide with interest.

Dr. Raja Ramanna spoke of the DAE's plan to set up a 500 MW prototype fast breeder reactor. Preliminary designs had been drawn up and with the benefit of the experience of the FBTR, scientists would have the confidence to build the large reactor, which would be ready by the turn of the century, he said.

He noted that the Reactor Research Centre, which had developed the FBTR, would be renamed the Indira Gandhi Centre for Atomic Research by the Prime Minister, Mr. Rajiv Gandhi, when he visits Kalpakkam on December 16 this year. The Prime Minister is also expected to dedicate formally to the nation the FBTR and the second unit of the Madras Atomic Power Station.

Our Tiruchi Staff Reporter writes:

BHEL, Tiruchi has made its own contribution to the fast breeder test reactor by manufacturing and supplying the following equipments, said Mr. E. S. Chandrasekharan, Group General Manager in an interview in Tiruchi: intermediate heat exchangers; steam generators; block pile piping; and circuit pipes and special bends, including 'Y' bends.

# All eyes on Kalpakkam

From G. K. Reddy

NEW DELHI, Oct. 18.

After the Pokhran explosion in 1974, no single event in India's nuclear development has evoked such great excitement in the foreign diplomatic community as the successful tests that are being carried out by the country's atomic scientists with the fast breeder test reactor at Kalpakkam.

The scientific attaches of various missions have been seeking details from the departments concerned about this remarkable achievement by a developing nation, while the political experts are assessing the wider implications of this spectacular breakthrough by India in nuclear development.

**Ultimate intention:** What is considered really creditable is that the Indian scientists have succeeded in developing indigenously mixed carbide fuel with a plutonium and uranium base, instead of enriched uranium which India proposed to obtain from France at one stage for this experimental fast breeder programme. The ultimate intention is to use thorium which can be extracted from monazite sand as fuel when fast breeder reactors go into commercial production in due course.

There are only six countries—the U.S., Soviet Union, France, Britain, West Germany and Japan—that are ahead of India in fast breeder technology which is going to be the main source of nuclear power in future. But the Indian success in reaching criticality at the experimental reactor in Kalpakkam is considered quite creditable.

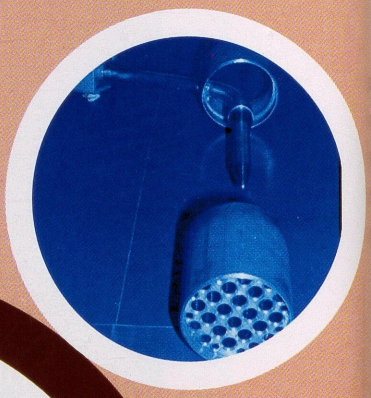
**Pak. canard:** Radio Pakistan which picked up the news from an AIR broadcast is reported to be spreading the canard that India which has already achieved nuclear capability is well set to embark on a weapons programme at short notice with the commissioning of the 50 MW research reactor at Trombay a few months back followed by the breakthrough in fast breeder technology at Kalpakkam.

Whatever the technical complexities of it, the very fact that India has crossed the nuclear barriers in this highly sophisticated sphere will go a long way in enhancing its international importance.

Chapter - 3

# Engineering R&D









# Engineering R&D

## Engineering R&D

With the decision to set up the Fast Breeder Test Reactor, an experimental programme to establish a good base in Sodium Technology was launched in Engineering Hall, REL, IGCAR. The programme was formulated in the seventies with the following objectives:

- Establishing the basic technology of liquid sodium
- Rendering support to the design, construction and commissioning of **FBTR**
- Development of sodium specific instruments and sensors

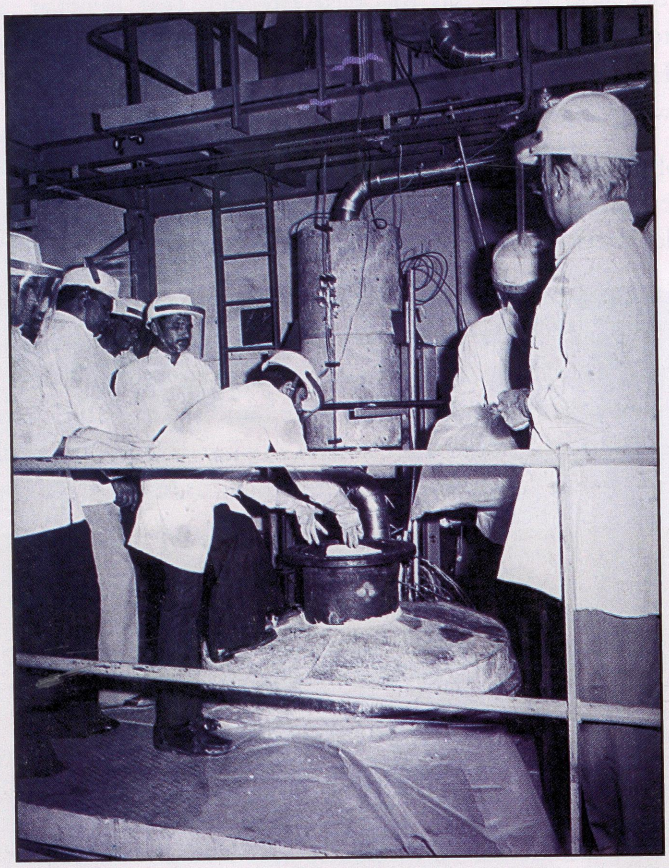
Due to the chemical affinity of sodium to air and water, handling of sodium needed special expertise. The safest way to handle sodium is to maintain it under a cover of inert gas like Argon, thereby preventing ingress of air or moisture into the system. It goes without saying that the sodium handling systems have to be leak-tight requiring welded construction with sound quality control of welds. All sodium systems are provided with a dump tank, into which sodium can be drained rapidly, in



*Sodium bricks in a drum*

the event of a leak as a safety measure. All capacities and piping for circulating sodium are to be preheated with trace heaters, as the melting point of sodium (~98°C) is above room temperature.

Conventional instrumentation for measurement of liquid level, flow and pressure cannot be used for sodium, because of its high chemical activity with air/moisture. Thanks to the excellent electrical conductivity of sodium, level probes



*Sodium brick charging into sodium purification set up*

and flow meters based on the principle of electro magnetism which are non intrusive have been developed.

The purity of sodium must be maintained to a high degree, with close control on Oxygen, Hydrogen and Carbon. Higher Oxygen content leads to corrosion and higher Carbon leads to carburisation of stainless steel. Higher Hydrogen levels especially in the secondary sodium are necessarily to be detected as it can be consequence of sodium-water reaction in the SG. Sodium Purification set up was followed by the construction and operation of many sodium rigs. These deal with

various aspects of sodium pumping, measurement of sodium flows, levels, temperatures, etc. Experience has also been gained in the safe cleaning of sodium soaked components, fighting mock sodium fires and disposing waste sodium which is a hazardous operation.

## Sodium for FBTR

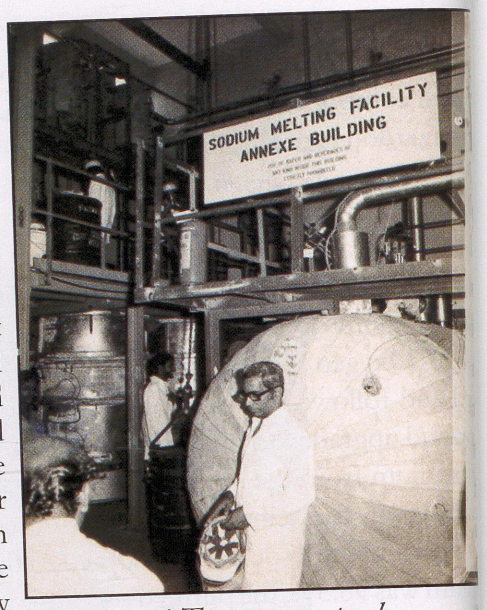
With a view to understand the liquid metal purification techniques and monitoring impurities, a small purification loop was set up in 1975. Commercial grade sodium obtained from the Indian market contained high levels of Calcium besides Oxygen as impurity. It was available in brick form and the same was melted, aged and followed by two stages of filtration and cold trapping in the purification loop and upgraded to reactor grade with oxygen level below 3 ppm. In a cold trap, the hot sodium from the system is gradually cooled in wire mesh filled tank, where the lowering of the temperature causes precipitation of excess impurities from the solution and pure sodium is returned to the system. While the purity of commercial sodium was 98.5%, the purified sodium was 99.95% pure.

***Impurities in Purified Sodium***

| Element | Concentration (ppm) |
|---|---|
| Aluminium | 5.0 |
| Boron | 0.5 |
| Cadmium | 0.3 |
| Calcium | 1.5 |
| Carbon | 15.0 |
| Chlorine | 8.0 |
| Chromium | 0.5 |
| Cobalt | 0.5 |
| Copper | 0.4 |
| Hydrogen | 0.06 |
| Indium | 3.0 |
| Iron | 3.0 |
| Lead | 1.0 |
| Magnesium | 0.2 |
| Manganese | 0.2 |
| Molybdenum | 2.0 |
| Nickel | 3.5 |
| Oxygen | 3.0 |
| Sulphur | 1.0 |
| Uranium | <50 ppb |
| Zinc | 2.0 |

The experience gained has been instrumental in finalising the purification scheme for upgrading commercial sodium to meet the requirements of 150 T of sodium for FBTR.

The system for treatment of commercial grade sodium consisted of two melter tanks of 1 T capacity each, two 6 T process tanks for ageing sodium at 150°C for calcium impurity removal, and two 30 T tanks connected to a pumped loop for circulating sodium through a cold trap. It took about 120 days to treat one batch of sodium (30 T) to achieve reactor grade quality. A total of 150 T of sodium was purified and stored in the storage tanks before transporting to FBTR. Many component test rigs were built and operated, in support for FBTR.



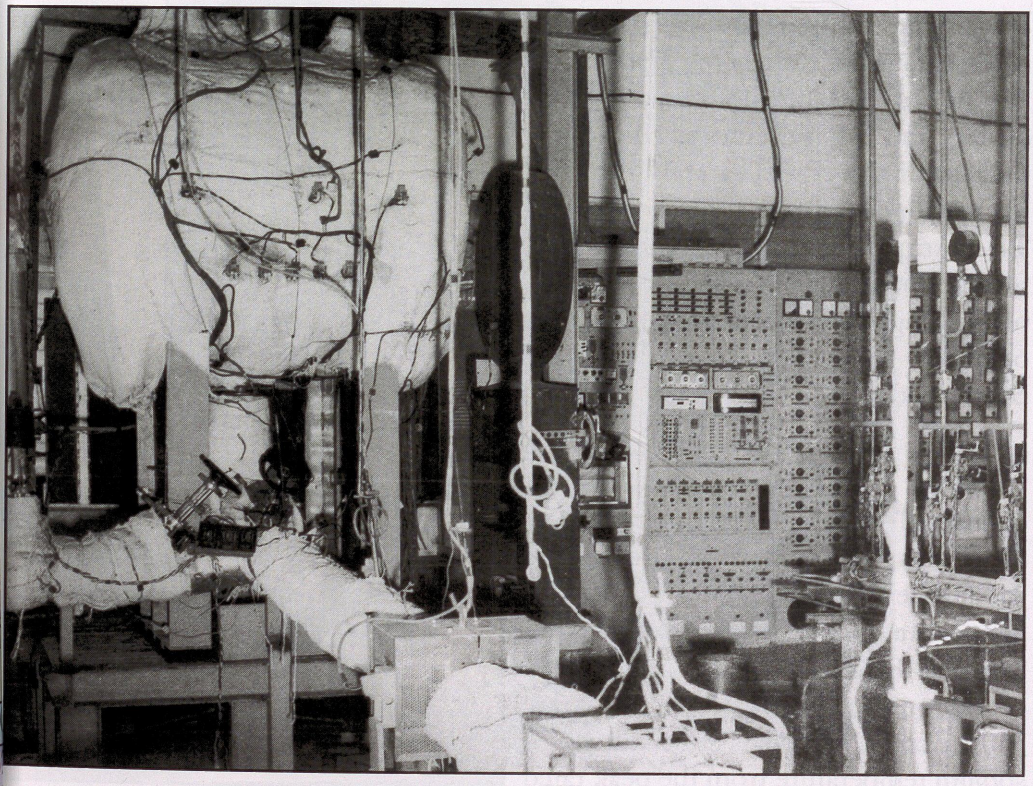

*6 Ton process tank*

***Sodium Test Rigs***

| Test Rig | Year | Temp (°C) | Flow $m^3/h$ | Hold up (litre |
|---|---|---|---|---|
| Heater Test Rig | 1974 | 550 | Static | 2 |
| Sodium Purification Loop | 1975 | 400 | 1.0 | 5 |
| Sodium Vapor Deposition Loop | 1976 | 550 | Static | 50 |
| 500 kW Loop | 1977 | 550 | 70 | 300 |
| Fuel Pin Test Rig | 1980 | 500 | 3 | 4 |
| Control Rod Drive Test Facility | 1982 | 550 | Static | 125 |
| Flowmeter Calibration Loop | 1979 | 500 | 16 | 150 |
| Test Rig for Rupture Discs | 1982 | 550 | Static | 2 |
| Thermal Shock Tests | 1985 | 550 | static | 1 |
| Test Rig for Thermocouple Time Constant | 1981 | 500 | Static | |
| Sodium Purification Facility FBTR | 1984 | 350 | 2.5 | 600 |
| Caldena Loop | 1985 | 500 | 1.0 | 3 |
| Mass Transfer Loop | 1986 | 600 | 0.21 | 1 |

## Flow meter calibration loop

Twenty five permanent magnet flow-meters are used for measuring flow ranging from 0.4 to 30 $m^3/h$ in the main and auxiliary sodium circuits of FBTR. The principle of operation of the permanent magnet flow-meters is similar to that of a DC generator and is based on the Faraday's law of electromagnetic induction. Electrical contacts (electrodes) positioned diametrically opposite each other are welded to the outer surface of the pipe with their central axis oriented normal to the direction of the lines of magnetic field and flow. A small DC voltage is developed across the electrodes by the conductive liquid metal (sodium) as it flows through and cuts the magnetic field. The magnitude of DC voltage is directly proportional to the velocity of the liquid and the polarity is determined by the direction of flow. While these flow-meters were procured from M/s GAAA France, the calibration was done in-house to avoid paying huge charges for calibration and also to get experience. Besides the facility created for the purpose was later useful for other sodium experimental work. The main sodium flow-meters of primary and secondary circuit were however calibrated by the supplier in their sodium facility.



*Flow meter calibration set-up on the weigh bridge*

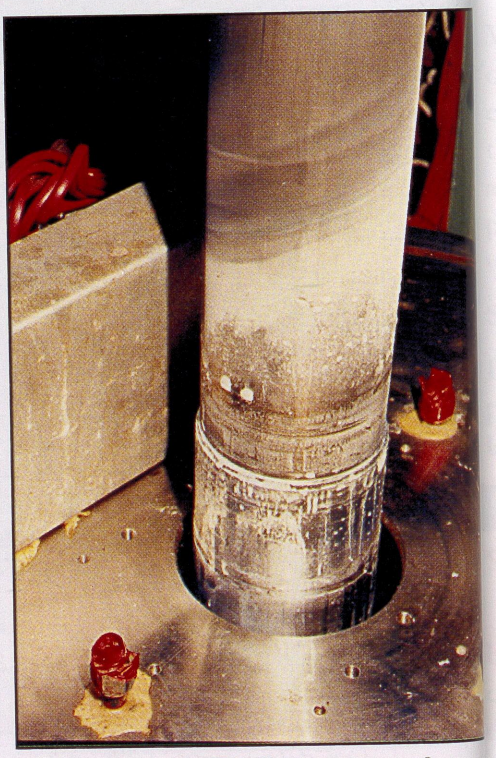
Flow meter calibration facility was erected in Engineering Hall. Absolu weigh tank calibration method was adopted for calibration. Known quantity o sodium (around 360 kg) was taken in the weighing tank and was drained to th storage tank through the flowmeter at different flow rates. In a campaign, differe flow-meters of same pipe size are installed in series and the data for calibration wa recorded simultaneously. The sensitivity was estimated based on the time take average millivolt output from the flowmeter and the actual quantity of sodiu drained. Calibration runs were taken at two different sodium temperatures of 200° and 425°C. The accuracy of the calibration was ±3%. In addition, a flowmeter (8 NB size) required for the flooding circuit of FBTR was designed, fabricate calibrated in sodium and supplied.

Alternative techniques other than the weighing method were reviewed an cross correlation of the fluctuating flow signals from two adjacent flow-meters we used to find transit time for sodium to flow through a known distance. The techniqu was found to have an accuracy of ±6%. This technique was developed for calibratin flow meters in-situ.

## CRDM test set up

A rig was built to test the indigenously manufactured CRDM in conditio representative of the reactor. The CRDM is one of the most precision-made devices and is important for controlling the reactor power besides shutting down the reactor under off normal conditions. The important tests included measurement of drop time, verification of raising & lowering speeds and operation of gripper which holds the control rod.

The test rig comprised a tall test vessel of 600 mm diameter, 8 m tall, connected to a dump tank through a drain line and purification circuit. The sodium level, cover gas pressure and temperature gradients were simulated in the tests. Tests were conducted at 180°C corresponding to FH condition and 550°C, corresponding to the maximum sodium temperature from the peak rated SA in the reactor at full power.

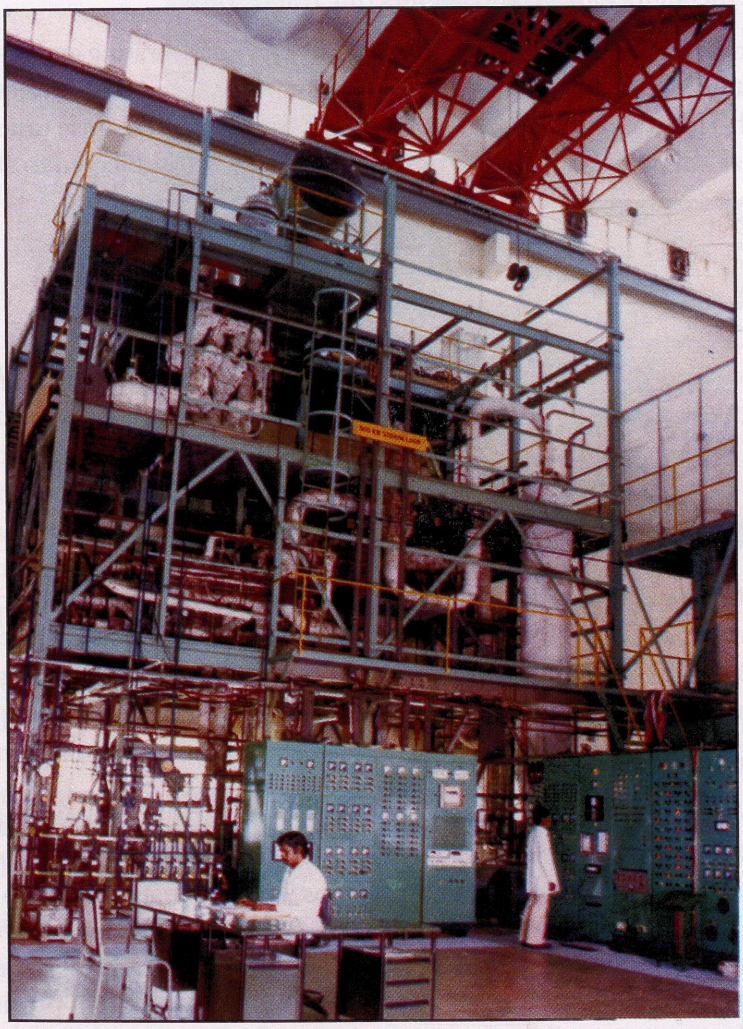


*CRDM in test vessel*

Later the main translation and gripper bellows procured for spare CRDM were also tested in sodium and qualified. Creep hold test of gripper bellows was conducted at 600°C for 1000 hours.

## 500 kW loop

A general purpose large sodium loop (500 kW) was built to test several components such as heat exchangers, immersion heaters, cold trap, etc. The loop was in the shape of 'figure-of-eight', in which a large sodium to sodium heat exchanger of 3 MW could be tested with a power input of only 0.5 MW. This was rendered possible because the same sodium circulates in the secondary and primary sides of exchanger. An indigenously designed centrifugal pump circulated sodium in this loop.



*500 kW sodium loop*

## Rupture disc development

A major breakthrough in the import substitution came about with the development of rupture discs for secondary sodium system. These discs are provided to relieve the pressure build-up resulting from a sodium water reaction in the SG. The disc works on the principle of reverse buckling and snaps open during reversal due to shearing action of sharp knife blades. The rupture discs were 300 mm diameter and 1 mm thick. Their welding to the flange was another important technology development.

The design and development of these discs was carried out by a multidisciplinary task force whose members were drawn from Design group, materials & metallurgy group and engineering tests group.

The rupture discs were tested for their pressure of bursting and response time. It was seen that at the intended pressure (8 bars), the response time lay between 4.2 - 5 millisecond. This was a very encouraging feat as the lowest response time reported in literature was 2.5 millisecond. The rupture discs were also subjected to static sodium

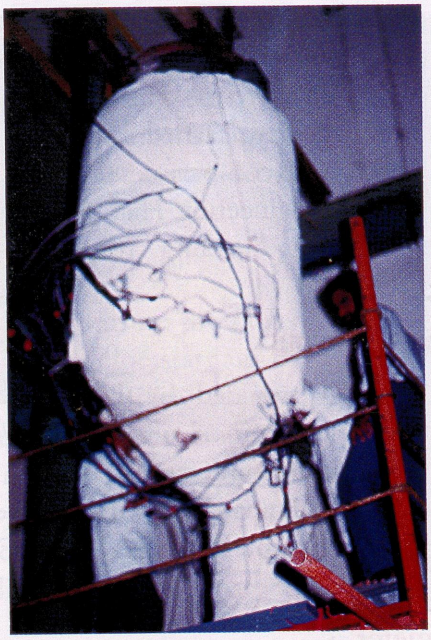


*Rupture discs – before and after*

tests for 4500 hours under normal operating conditions to assess their degradation. The discs were subjected to thermal transients that could be seen by the disc during incidents of MBFP trip and secondary sodium pump trip. This testing established the soundness of the design concept.

## FBTR secondary cold trap regeneration

Cold trap installed in the secondary circuit of FBTR got prematurely plugged. It was replaced with a new cold trap and the old cold trap was taken up for regeneration. This was carried out in three steps. First by heating the trap to 550°C to decompose sodium hydride impurities, then removing evolved hydrogen by vacuum pulling, further allowing sodium to distil out and lastly by admitting steam gradually to remove the remaining sodium & other impurities. The reaction products, mainly sodium hydroxide, was drained down and collected. Based on the analysis of the hydroxide, the amount of sodium was estimated.



*Vacuum distillation of cold trap*

### Water testing of FBTR sodium pumps

Verification of sodium pump performance is essential both from the point of view of ensuring smooth running of the complex centrifugal pumps as well as establishing their hydraulic characteristics. A water test rig was constructed towards this end, thanks to the similarity in fluid flow properties of liquid sodium and water and the hydraulic similitude. The complete characteristics were established for 3 primary sodium and 3 secondary sodium pumps. Also, endurance run of 100 hours was successfully completed.

### Subassembly hydraulics

Besides the above, a hydraulic test rig has been set up and operated to study the pressure drop vs flow and cavitation characteristic of flow limiting orifices and labyrinths for full scale SA.

## Instrumentation development

### Sodium Ionisation Detector

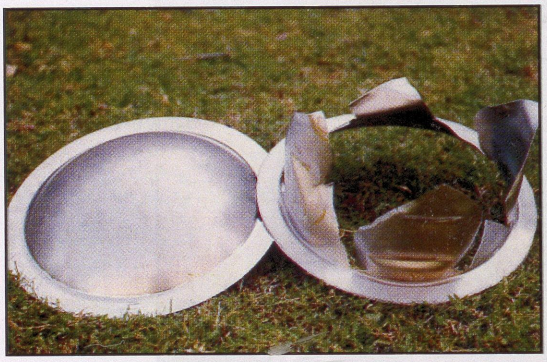
Detection of sodium aerosols in areas, where sodium leak has gone undetected by the conventional wire type and spark plug type detectors is important from safety considerations. In this direction, a sensitive Sodium Ionisation Detector has been developed and found to detect sodium leaks down to a sodium concentration of few nanograms/cc of carrier gas.

## Rupture disc development

A major breakthrough in the import substitution came about with the development of rupture discs for secondary sodium system. These discs are provided to relieve the pressure build-up resulting from a sodium water reaction in the SG. The disc works on the principle of reverse buckling and snaps open during reversal due to shearing action of sharp knife blades. The rupture discs were 300 mm diameter and 1 mm thick. Their welding to the flange was another important technology development.

The design and development of these discs was carried out by a multidisciplinary task force whose members were drawn from Design group, materials & metallurgy group and engineering tests group.

The rupture discs were tested for their pressure of bursting and response time. It was seen that at the intended pressure (8 bars), the response time lay between 4.2 - 5 millisecond. This was a very encouraging feat as the lowest response time reported in literature was 2.5 millisecond. The rupture discs were also subjected to static sodium

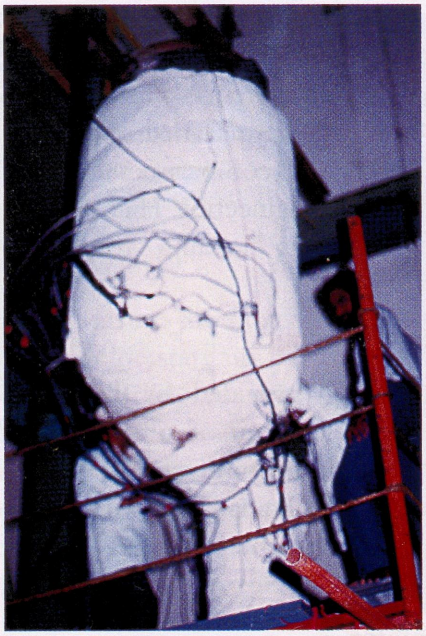


*Rupture discs – before and after*

tests for 4500 hours under normal operating conditions to assess their degradation. The discs were subjected to thermal transients that could be seen by the disc during incidents of MBFP trip and secondary sodium pump trip. This testing established the soundness of the design concept.

## FBTR secondary cold trap regeneration

Cold trap installed in the secondary circuit of FBTR got prematurely plugged. It was replaced with a new cold trap and the old cold trap was taken up for regeneration. This was carried out in three steps. First by heating the trap to 550°C to decompose sodium hydride impurities, then removing evolved hydrogen by vacuum pulling, further allowing sodium to distil out and lastly by admitting steam gradually to remove the remaining sodium & other impurities. The reaction products, mainly sodium hydroxide, was drained down and collected. Based on the analysis of the hydroxide, the amount of sodium was estimated.



*Vacuum distillation of cold trap*

### Water testing of FBTR sodium pumps

Verification of sodium pump performance is essential both from the point of view of ensuring smooth running of the complex centrifugal pumps as well as establishing their hydraulic characteristics. A water test rig was constructed towards this end, thanks to the similarity in fluid flow properties of liquid sodium and water and the hydraulic similitude. The complete characteristics were established for 3 primary sodium and 3 secondary sodium pumps. Also, endurance run of 100 hours was successfully completed.

### Subassembly hydraulics

Besides the above, a hydraulic test rig has been set up and operated to study the pressure drop vs flow and cavitation characteristic of flow limiting orifices and labyrinths for full scale SA.

## Instrumentation development

### Sodium Ionisation Detector

Detection of sodium aerosols in areas, where sodium leak has gone undetected by the conventional wire type and spark plug type detectors is important from safety considerations. In this direction, a sensitive Sodium Ionisation Detector has been developed and found to detect sodium leaks down to a sodium concentration of few nanograms/cc of carrier gas.

Sodium or sodium compounds are thermally ionised in the detector which uses Pt/Rh filament and a molybdenum collector. The ion current is a measure of sodium concentration. Tests were carried out by suitably simulating the double wall compartments of FBTR primary piping in the Engineering Hall and results were very encouraging leading to its deployment in FBTR.



*Sodium Ionisation Detector*

## Ultrasonic under sodium viewing

The good acoustic coupling provided by liquid sodium has enabled the development of ultrasonic transducer for visualising objects immersed in the opaque sodium. An ultrasonic transducer with piezoelectric device has bee developed, tested in water experiments and later used in FBTR during investigatio of FH and CCPM incidents.

## Hydrogen in sodium meter

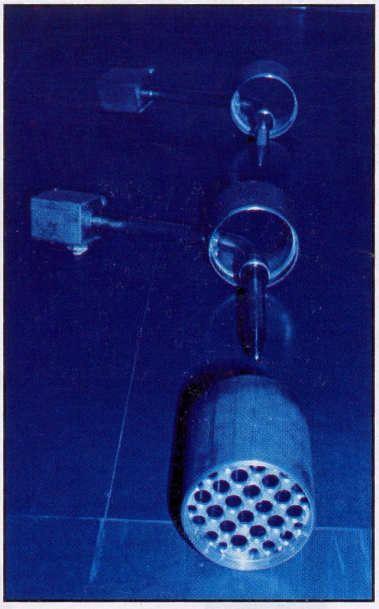
Monitoring of hydrogen in the secondary sodium is important for detectio of small water leaks in SG. The detector works on the principle of hydroge diffusion through a thin nickel membrane, the other side of which is maintained at high vacuum of about $10^{-8}$ torr. The detector was developed indigenously and teste in the CALDENA sodium loop and found to detect water/steam leaks of ~0.05 g/s

## Hydrogen in cover gas meter

At sodium temperatures below ~350°C, any hydrogen in sodium gets release to the cover gas, whenever it reaches a free sodium capacity. For detection o hydrogen in cover gas, a thermal conductivity based detector was develope indigenously at RCL, IGCAR. It works on the Wheatstone bridge principle. Th meter consists of a nickel tube assembly housed in a stainless steel vessel whic forms part of the piping in the cover gas of sodium system. For effective workin this tube needs to be maintained at 500°C and this was achieved by a heater. Pu argon as carrier gas was passed through the nickel tube and was subsequently allowe to pass through the arms of Wheatstone bridge. Due to the differences in therm conductivities of the two streams (reference argon gas & argon with sample gas) a unbalance is created in the bridge and the output of the meter varies. The output proportional to the concentration of hydrogen in the cover gas.

The meter was planned to be installed in the reaction product discharge line o FBTR expansion tank which is a stagnant argon line and hence, doubts were there o the possibility of hydrogen reaching the sensor. Therefore, FBTR arrangement w

simulated in Engineering Hall sodium loop and a natural convection was established in the circuit by maintaining proper temperatures in the cover gas lines. After establishing similar natural convection in FBTR, the meter was installed and is working satisfactorily.



*Eddy current flow meter assembly*

## Eddy current flow meter

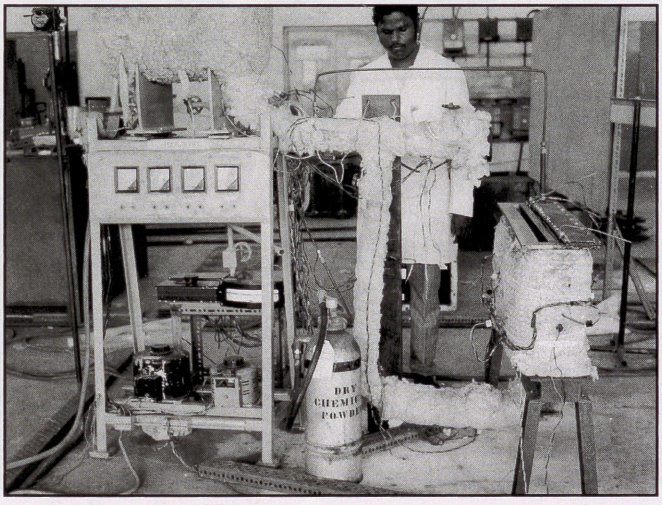
There was the need to measure the flows through the fuel SA before reactor start up to assess the plugging, if any. An Eddy current flowmeter was developed & calibrated in the 500kW loop and has been successfully utilised in FBTR. Recently, a similar flowmeter was supplied and installed in the French PHENIX reactor for measurement of low sodium flows under natural convection.

## Sodium vapour deposition studies

Temperature of sodium varies between 180°C to 550°C, depending on the reactor power level. Sodium aerosols get carried into the argon due to natural convection and get deposited in the cooler parts in the gas space. Experience has shown the difficulties in the rotation of the top shields due to sodium deposition. Studies of deposition patterns was carried out in a sodium test facility without and with anti convection baffle and labyrinths. It was established that labyrinths were effective in reducing sodium aerosols.

## LM Seal development

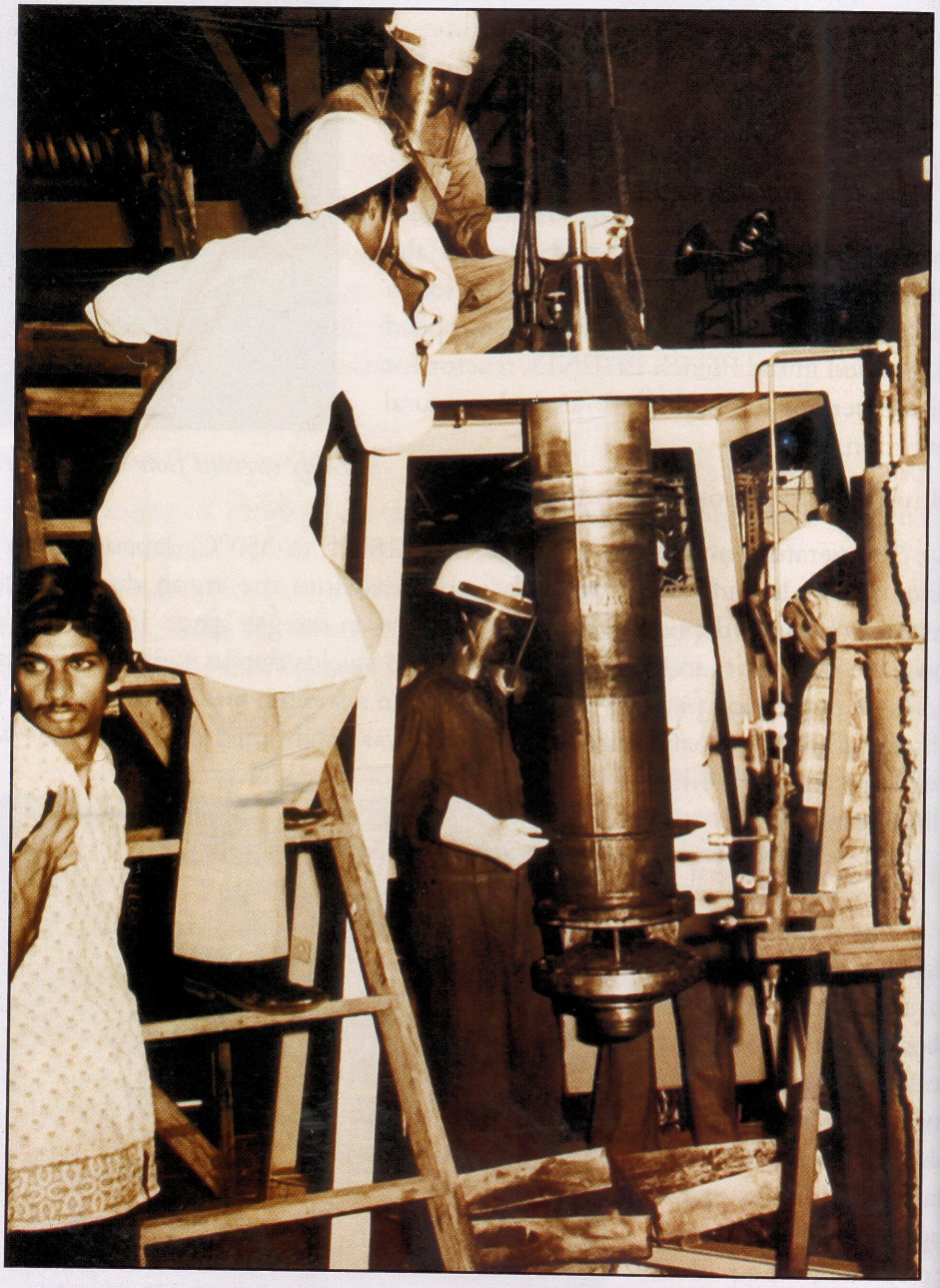
The reactor rotating plugs use LM alloy to isolate the reactor radioactive cover gas from the reactor building atmosphere and provide isolation in both static and rotating conditions of the plug, the latter being necessary for FH operations. A special tin-



*LM seal test set-up*

bismuth (42% Tin-58% Bismuth) eutectic alloy is used to seal due to its favourable solidification and adhering characteristics.

However, the use of stainless steel seal-housings needed special effort to achieve a high degree of bonding when the seal alloy is in the solid state. A specific



*Steam cleaning of 500 kW loop sodium pump*

process of tinning the stainless steel housing was developed and the alloy was poured to realise a helium leak tight seal in the frozen/solid state.

## Sodium removal from components

The components in sodium systems need to be cleaned free of sodium before attempting their repair and maintenance.

Various cleaning techniques employed are alcohol dissolution, water vapour-nitrogen process, water-vapour carbon dioxide process and vacuum distillation. In all the processes, care must be taken to keep the hydrogen concentration in air below 4% to avoid explosion. Whenever the concentration goes up, it is diluted with more inert gas. Alcohol dissolution is used for small components like valves, bellows, etc. This experience has been very useful during the maintenance works in FBTR. The advantage of using carbon dioxide in place of nitrogen used earlier is to aid in the formation of sodium carbonate through reaction with sodium hydroxide; sodium carbonate being a solid, it falls down.

## Sodium fire & fire-fighting

Sodium reacts with oxygen in air to form oxides. In contrast to conventional fires, sodium fires are characterised by low flames and dense clouds of white smoke

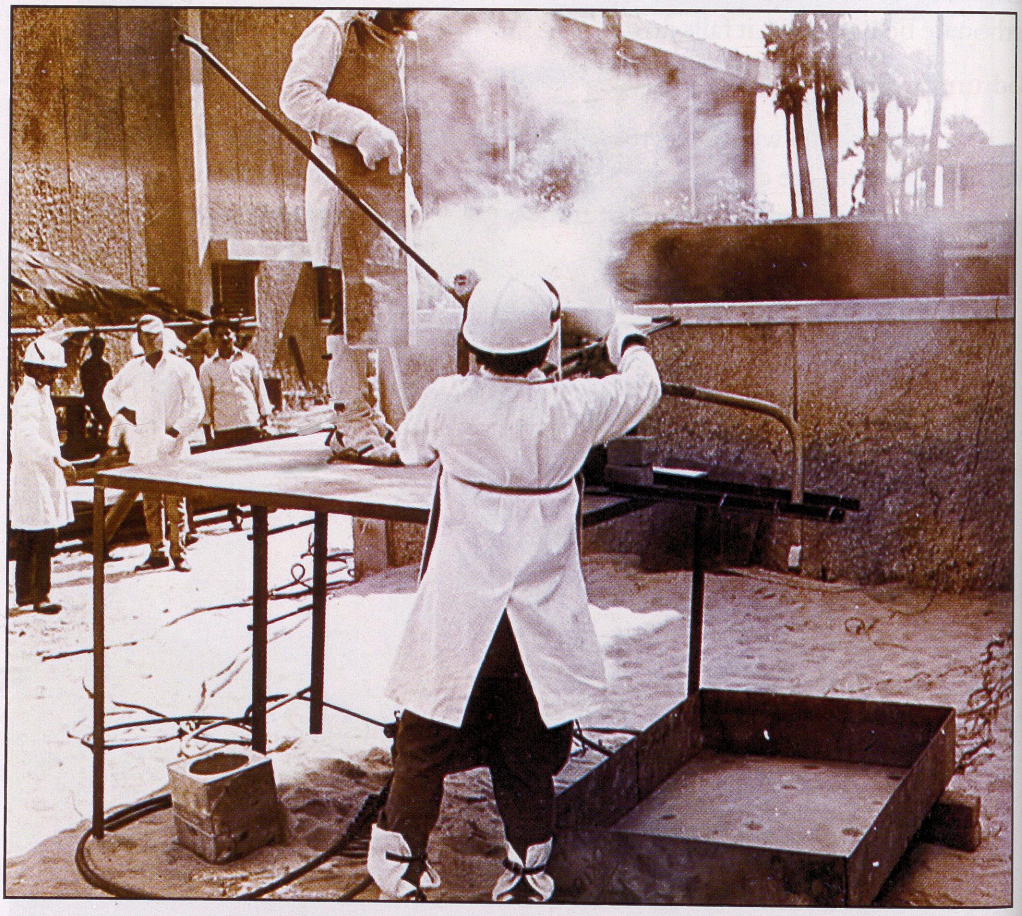


*Sodium fire studies*

(oxide). The amount of heat liberated during this combustion process is considerably less than that evolved by combustion of equivalent weight of petrol.

Sodium burns quietly in air and does not give rise to explosions as long as water or other reactive materials do not come into contact with it. Thus, the sodium fire can be approached in most of the cases. The main problem associated with sodium fire fighting is the evolution of large amount of white oxide smoke which can reduce the visibility rapidly in an enclosed space.

Experiments were carried out to assess the requirement of different chemical powders for extinguishing sodium fire. Also, the rate of cooling by the different powders was assessed. It was found that Sodium bi-Carbonate based Dry Chemical Powder was very effective and has been deployed for FBTR. Training in sodium fire fighting has been imparted to all the operation crews of FBTR.



*Extinguishing sodium fire*

The caustic nature of the combustion product as well as the potential hazard of un-reacted sodium makes it desirable to perform a complete clean up operation after the fire is put out and the materials are cool enough to handle.

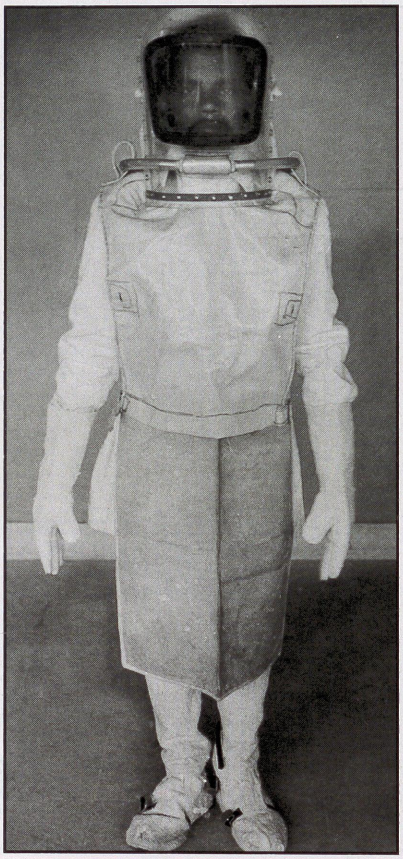
All residue, unburnt sodium and sodium contaminated insulation should be shovelled into vented metal cans (provided with cover) for disposal.



*Sodium fire blanketed by DCP*

## Personnel protection

In addition to the hazards normally associated with fire fighting, sodium presents two extraordinary hazards (Caustic fumes and projected droplets of hot metal) against which the fire fighter must protect himself. Air-line respirators or gas masks, flame-proof coveralls, a hard hat with face shield, plastic goggles, leggings and gloves of chrome leather should be worn by all personnel who approach a large sodium spill.



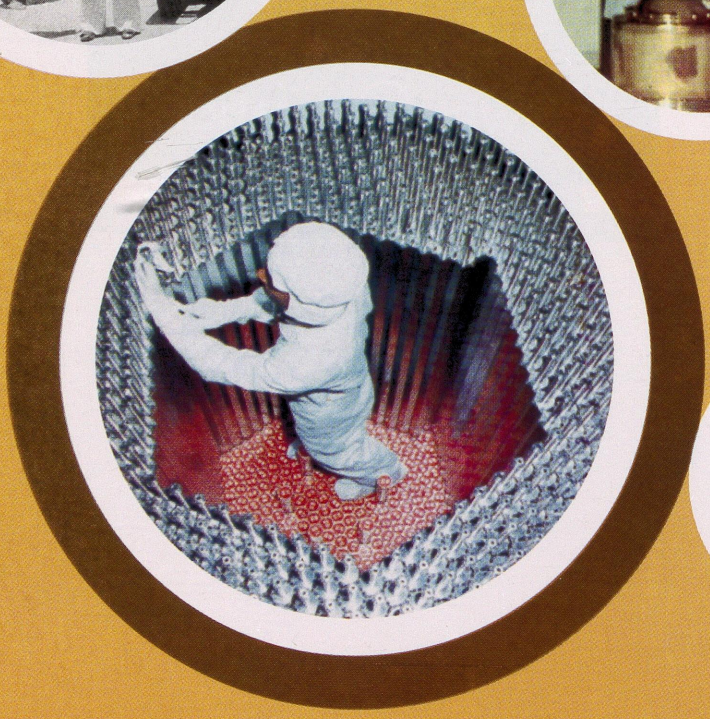
*Sodium safety dress*

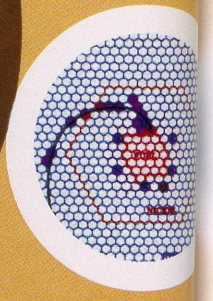
Chapter - 4

# Commissioning, Criticality and Start of Power Operation









# Commissioning, Criticality and Start of Power Operation

After successfully erecting various reactor components as detailed in the 'Project Years', commissioning of various systems were taken up in stages. Due to delay in the arrival of Steam Generators, commissioning of Steam Water System was not included in the initial phase.

**Commissioning phases:**

**The reactor was commissioned in four phases:**

- The first phase was for low power operation without the SG in place. Secondary sodium loops were operated with the SG bypassed. The entire primary sodium system and the secondary sodium system without the SG were commissioned.
- In the second phase, the steam generators were installed and connected to the secondary sodium circuit & the reactor was operated with sodium on the SG shell without valving-in water on the tube side. Heat removal was by natural convection of air through the SG casing. Reactor power was limited to about 1 MWt.
- In the third phase, the steam water system and the critical SGLDS were commissioned and water was admitted into the SG in January 1993. Heat removal was through the Dump Condenser.
- The TG was commissioned in 1997.

Auxiliary systems like electrical power supply, service water, compressed air, service argon, service nitrogen, A/C & ventilation systems were erected and commissioned by 1980. All the major heat transport systems were in place by 1982.

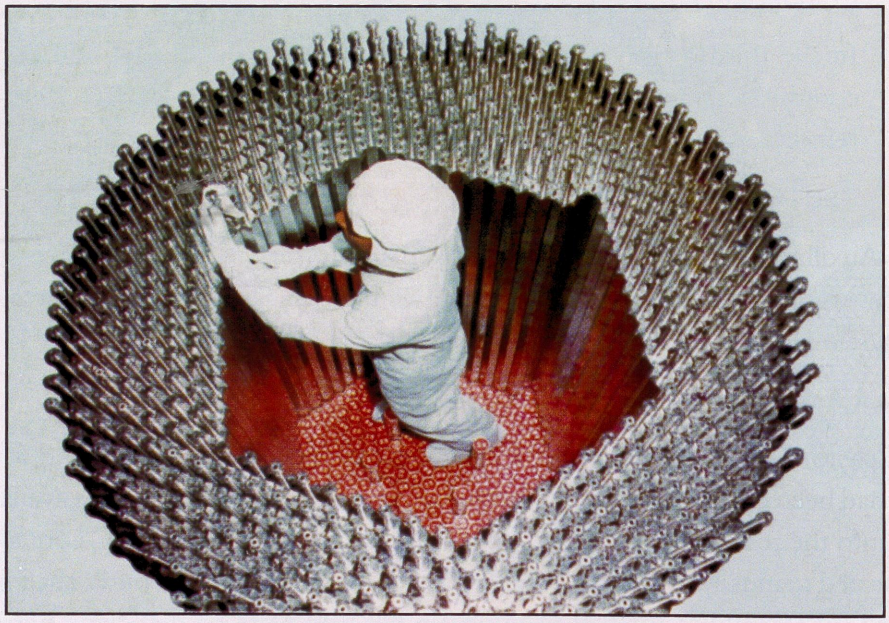
## Reactor Assembly

The Reactor Vessel along with grid plate, neutron & thermal shields and rotating plugs had been erected by 1983. Control plug opening was the only access available to enter into the reactor vessel. As a preparatory step for commissioning, core loading was started manually to load steel & nickel reflectors, control rod outer sheaths and dummy subassemblies for fuel positions in the grid plate by deploying personnel

inside RV with the help of ladder positioned through control plug opening. Lighting and compressed air supply was provided inside the reactor vessel. Every subassembly was loaded physically and its orientation was checked visually. After loading 19 SA in the reactor vessel, during level checking, it was observed that three caps kept for position indication in the grid plate were missing. So, all the SA were discharged for tracing the missing caps which could be found below the neutron shields later. The neutron shields were lifted and caps were retrieved and put in position. Thereafter, the administrative controls were strengthened. The core loading was restarted and utmost care was taken to ensure that no loose items are taken inside RV. Core loading was carried out only in the second and night shifts. Day time was utilised to stock the SA for charging, inventory control of caps & housekeeping at attic. While core loading, gaps between SA, verticality of control rod sleeves, radial distance of control rod outer sheaths from core centre and the force required for lifting the SA were measured and found to be in order.

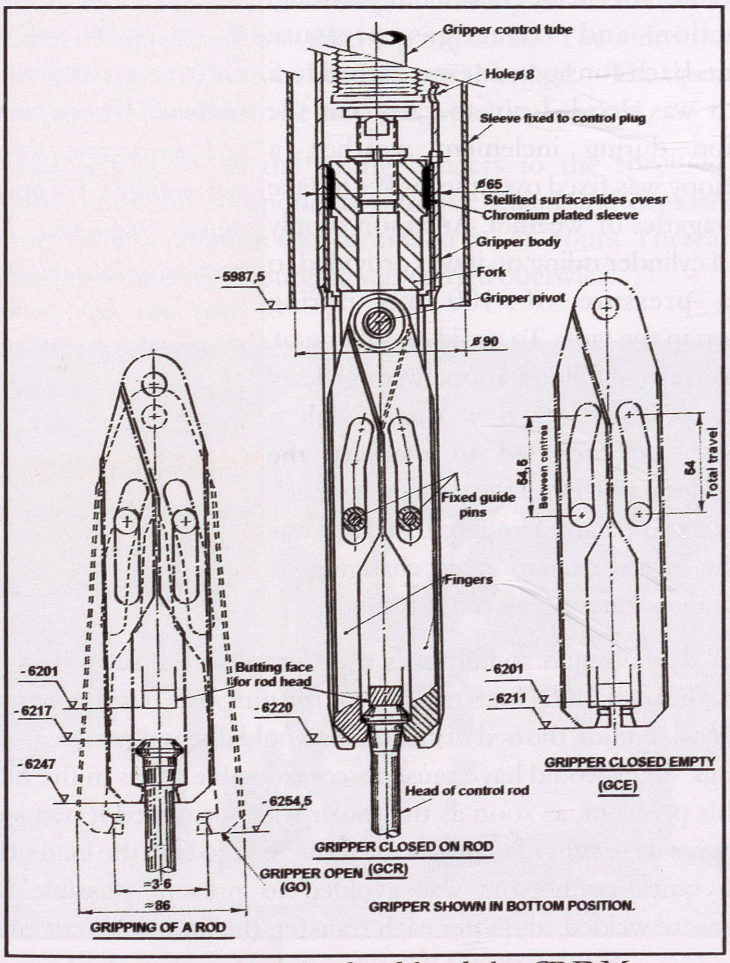
## Control Rod Drive Mechanisms

FBTR has six control rods for regulation of power and safe shutting down of the reactor. During fuel handling operations, the control rods are released from the CRDM by opening the gripper fingers and are deposited in the core. CRDM is then kept raised by 70 mm and disengaged to enable the rotation of the rotating plugs. During initial trials, it was felt that if misalignment exists between the axis of the



*Manual Loading of Core*

CRDM & axis of control rod, there is a chance of gripping of the control rod with one of the fingers even though the gripper is actually open. Hence, it was decided to check the misalignment physically. This warranted stationing of personnel inside the reactor vessel while operating the gripper fingers. The control plug along with all CRDM was lifted with RCB crane. A chain pulley block was kept as standby. With two engineers inside the reactor vessel, the control plug was reinstalled; thus confining the engineers inside the reactor till the misalignment checks were completed. Redundant dust free breathing air supplies and low voltage lighting were provided inside the reactor vessel. A communication facility was also established. After the misalignment check, the control plug was removed and the engineers came out, much to the relief of the anxious team outside. The control plug was again reinstalled after successful misalignment check. This was followed by functional checks in argon at 180°C and in sodium at 250°C & 350°C. The results were satisfactory.




*Gripping of control rod heads by CRDM*

## Heat Transport Systems

The first major step towards commissioning of the heat transport system was transporting of purified nuclear grade liquid sodium from the rig in Engineering Hall, REL to FBTR. Considering the short distance and as the freezing & melting time of sodium inventory in the transportation tank was more than 10 days, it was decided to transport the sodium in liquid state from Engineering Hall to FBTR. The tank was mounted on a 15 T trailer, sodium was filled in the tank (maintained at 125°C) and the consignment was moved to FBTR. Tank was provided with instrumentation for temperature monitoring, level detection and cover gas pressure measurement. Each consignment was limited to 5 T. Though it was decided not to carry out the transportation during inclement weather, a tarpaulin canopy was fixed over the tanker to take care of the vagaries of weather. An argon supply header with a cylinder riding on it was provided to maintain a pressure of 100 mb during transportation in the tank. To avoid overfilling of transportation tank at filling station, two high level probes were provided. A flow meter with a recorder was also provided to measure the amount of sodium filled into the tank during each transportation campaign. The tractor-trailer was driven on the lowest gear to avoid changing of gear during transportation to avoid sloshing.

*The misalignment checking of the Control Rod Drive Mechanisms with respect to the heads of the control rods was a job involving some amount of personal risk and discomfort. The commissioning engineer and the inspection engineer entered the reactor through the Control Plug opening. The Control Plug with all the CRDM was reinstalled, thus confining them inside the reactor. The mechanisms were operated one by one and the actual gripping and releasing of the control rods was checked. Targets were then fixed on the control rod heads. Using a pointer gripped by each mechanism, the misalignment was measured. The precautions taken include redundant breathing air supplies and low voltage lighting. The crane was kept parked above the Control Plug with a chain pulley block as a redundant handling facility, in case the RCB crane hoist developed any problem. The whole operation took about four hours till they were 'released'.*

At FBTR, the liquid sodium was transferred to the secondary and primary storage tanks through filling lines erected for the purpose. It was seen that for a load of 5 T weight of sodium, the bed of the trailer would dip by 4 cm due to compression of leaf springs. This would have caused a considerable stress in the filling line. To surmount this problem, as soon as the trailer was positioned it was jacked up and wooden sleepers were provided below the trailer bed to take the load off the wheels. The use of quick connectors was avoided to prevent possible leakages. All connections were welded, and after each transfer, the lines were cut and blanked by welding.

## Secondary Sodium System

Initially, the secondary sodium storage tanks were filled. The storage tanks were vacuum degassed to 3 x $10^{-2}$ Torr at 175°C for 70 hours. Subsequent to degassing, the tanks were filled and flushed with argon and oxygen level of 18 vpm was achieved. Mercury was filled in relief pots of both loops to the required level and set to relieve at 78 mb. Tanks were filled by pressurizing the transportation tank to 400 mb. Nine such transfers were carried out to fill 24 $m^3$ of sodium into each one of the two secondary storage tanks. More emphasis was laid on the discontinuous level probes as the continuous level probes were not functioning properly. Also, proper metering of sodium filled into the transportation tank was done. High level probes of the secondary storage tanks were checked prior to each filling campaign. Filling of storage tank was commenced on April 3, 1984 and completed on May 3, 1984. Subsequently, sodium was frozen and argon cover gas was maintained.

Preparatory work for commissioning and filling of secondary sodium main circuit and primary storage tank was taken up in parallel. The individual nozzles of SG modules in both top and bottom headers were blanked. The open ends of the headers, where rupture discs had to be installed, were also blanked.

After cold setting of all the spring hangers to the specified values, the secondary system lines and components were preheated and degassed by pulling vacuum. The degassing operation was continued for 48 hours. Thereafter, vacuum was broken with argon and argon purging was carried out twice.



*Sodium transportation from REL to FBTR*

## Heat Transport Systems

The first major step towards commissioning of the heat transport system was transporting of purified nuclear grade liquid sodium from the rig in Engineering Hall, REL to FBTR. Considering the short distance and as the freezing & melting time of sodium inventory in the transportation tank was more than 10 days, it was decided to transport the sodium in liquid state from Engineering Hall to FBTR. The tank was mounted on a 15 T trailer, sodium was filled in the tank (maintained at 125°C) and the consignment was moved to FBTR. Tank was provided with instrumentation for temperature monitoring, level detection and cover gas pressure measurement. Each consignment was limited to 5 T. Though it was decided not to carry out the transportation during inclement weather, a tarpaulin canopy was fixed over the tanker to take care of the vagaries of weather. An argon supply header with a cylinder riding on it was provided to maintain a pressure of 100 mb during transportation in the tank. To avoid overfilling of transportation tank at filling station, two high level probes were provided. A flow meter with a recorder was also provided to measure the amount of sodium filled into the tank during each transportation campaign. The tractor-trailer was driven on the lowest gear to avoid changing of gear during transportation to avoid sloshing.

*The misalignment checking of the Control Rod Drive Mechanisms with respect to the heads of the control rods was a job involving some amount of personal risk and discomfort. The commissioning engineer and the inspection engineer entered the reactor through the Control Plug opening. The Control Plug with all the CRDM was reinstalled, thus confining them inside the reactor. The mechanisms were operated one by one and the actual gripping and releasing of the control rods was checked. Targets were then fixed on the control rod heads. Using a pointer gripped by each mechanism, the misalignment was measured. The precautions taken include redundant breathing air supplies and low voltage lighting. The crane was kept parked above the Control Plug with a chain pulley block as a redundant handling facility, in case the RCB crane hoist developed any problem. The whole operation took about four hours till they were 'released'.*

At FBTR, the liquid sodium was transferred to the secondary and primary storage tanks through filling lines erected for the purpose. It was seen that for a load of 5 T weight of sodium, the bed of the trailer would dip by 4 cm due to compression of leaf springs. This would have caused a considerable stress in the filling line. To surmount this problem, as soon as the trailer was positioned it was jacked up and wooden sleepers were provided below the trailer bed to take the load off the wheels. The use of quick connectors was avoided to prevent possible leakages. All connections were welded, and after each transfer, the lines were cut and blanked by welding.

## Secondary Sodium System

Initially, the secondary sodium storage tanks were filled. The storage tanks were vacuum degassed to 3 x $10^{-2}$ Torr at 175°C for 70 hours. Subsequent to degassing, the tanks were filled and flushed with argon and oxygen level of 18 vpm was achieved. Mercury was filled in relief pots of both loops to the required level and set to relieve at 78 mb. Tanks were filled by pressurizing the transportation tank to 400 mb. Nine such transfers were carried out to fill 24 $m^3$ of sodium into each one of the two secondary storage tanks. More emphasis was laid on the discontinuous level probes as the continuous level probes were not functioning properly. Also, proper metering of sodium filled into the transportation tank was done. High level probes of the secondary storage tanks were checked prior to each filling campaign. Filling of storage tank was commenced on April 3, 1984 and completed on May 3, 1984. Subsequently, sodium was frozen and argon cover gas was maintained.

Preparatory work for commissioning and filling of secondary sodium main circuit and primary storage tank was taken up in parallel. The individual nozzles of SG modules in both top and bottom headers were blanked. The open ends of the headers, where rupture discs had to be installed, were also blanked.

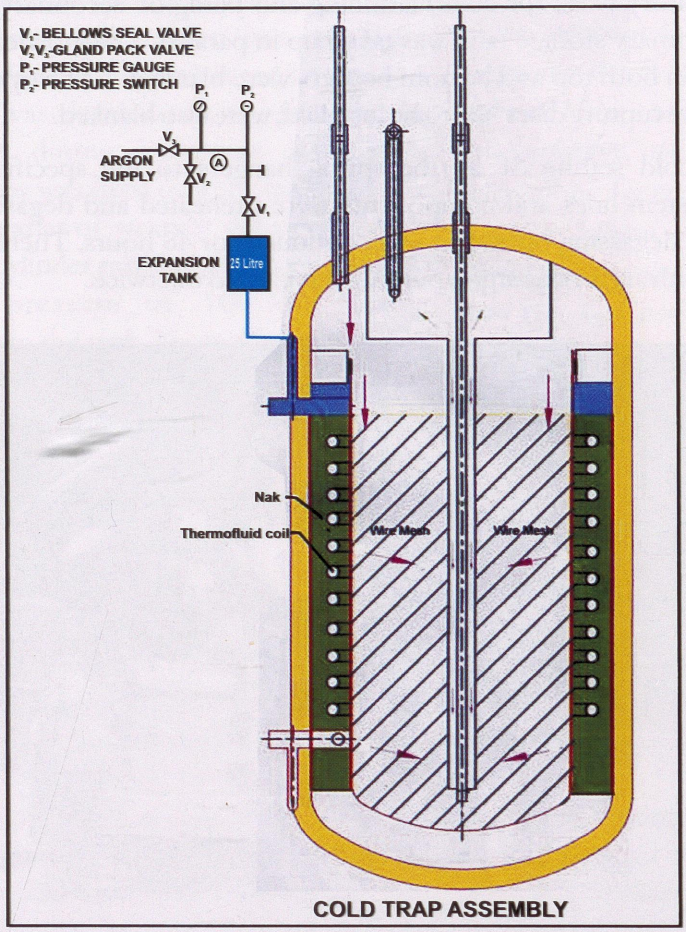
After cold setting of all the spring hangers to the specified values, the secondary system lines and components were preheated and degassed by pulling vacuum. The degassing operation was continued for 48 hours. Thereafter, vacuum was broken with argon and argon purging was carried out twice.



*Sodium transportation from REL to FBTR*

## NaK leak incident in Cold Trap

During preheating of the east loop cold trap, a fire incident occurred due to NaK leak near spark plug level probe mounted on the NaK jacket. Preheating was stopped immediately and fire was extinguished using Dry Chemical Powder. About 2.7 litres of NaK had leaked out. It was realized that the cover gas space provided for thermal expansion of NaK was inadequate, resulting in over-pressurisation of the NaK jacket and failure of the spark plug joint. Since the inner vessel had been subjected to external pressure, several examinations were conducted to ensure the integrity of the cold trap. An argon pot of 25 litres extra capacity was connected to NaK jacket cover gas space with a provision for pressurising and venting the gas space. Low and high level probes were removed and opening was blanked. The cover gas pressure switch setting was set at 0.75 $kg/cm^2$ for initiating alarm. A leak collection tray was also provided at the bottom of the cold trap.




*Modified cold trap with expansion tank and blanked spark plug*

Interestingly, it was observed during this incident that the insulation over the cold trap contained the leak to a large extent and prevented direct exposure of NaK to atmospheric air. Hence, it is advisable to insulate NaK lines and vessels even if they operate at room temperature from the point-of-view of reducing the intensity of fire.

## Filling of purification circuit, main circuit and purification

Prior to preheating, several cycles of pressurisation and depressurisation with argon were carried out to bring down the $O_2$ content to <10 vpm. The purification circuit was filled using the EM pump after priming it by pressurising the storage tank to 600 mb. Due to wetting problem, EM flow meter did not work initially. A sustained flow of 2.5 $m^3$/h was obtained at an operating voltage of 300 V for the pump. The progress of filling of the circuit was monitored mainly with the help of thermocouples installed on the lines and capacities. Plugging indicator was valved in and plugging temperature was found to be 132°C at a system temperature of 170°C. The cold traps were put into operation and the plugging temperature came down to < 110°C in about five hours. Then the storage tank sodium temperature was raised in steps of 25°C upto 300°C with cold trap in operation & plugging temp was always found to be <110°C. The sodium temperature was lowered back to 250°C.

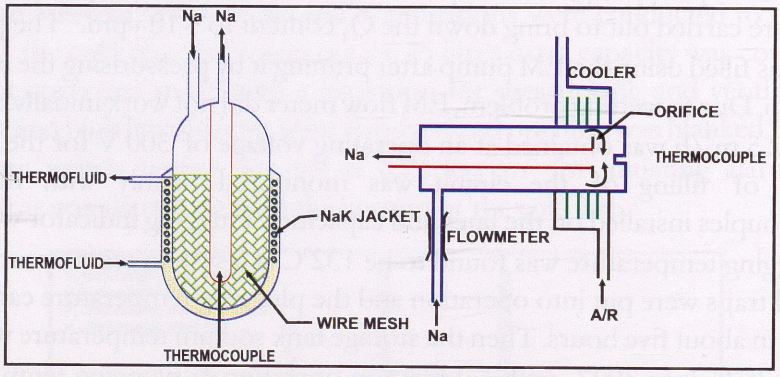
All the lines and components of secondary system were preheated and maintained at 200 ± 25°C. Before preheating of IHX, entire primary circuit was evacuated and filled with nitrogen. Preheating of IHX tube bundle was done by heaters placed on the shell and dished end of the top sodium collector. This method was not successful because of the strong natural convection currents getting established in the primary system during IHX preheating. Hence, the Preheating and Emergency Cooling circuit of the Primary system was commissioned on a war footing. Complete primary circuit along with IHX was preheated using hot air flowing through the double envelope by PHEC circuit.

The secondary storage tank temperature was maintained at 150°C and the pipeline temperatures were maintained above this value. This facilitated to follow the sodium filling by observing the dip in temperature sensed by the thermocouples mounted on the pipelines. The progress of filling was also monitored by the drop in storage tank level and rise in the levels of secondary capacities. About 12 $m^3$ of sodium was filled in the main circuit. Mandatory tests such as fast dump test (required in the event of any sodium leak), leak detector simulation test, etc. were carried out.

The system characteristics of the secondary sodium circuit during Phase I operation without SG were different from the normal operation. This resulted in operation of pump away from the Best Efficiency Point. There was also some

apprehension about incipient cavitation. The optimum speed of 409 rpm was selected to get minimum deviation from BEP.

Except the NaK leak incident, there were no other major problems encountered while commissioning the secondary sodium system. The NaK fire incident gave an opportunity for the operating staff for fire-fighting and gain good experience. Sodium purification circuit worked exceedingly well and Plugging temperature of <110°C could be obtained.




*Cold trap and plugging indicator*

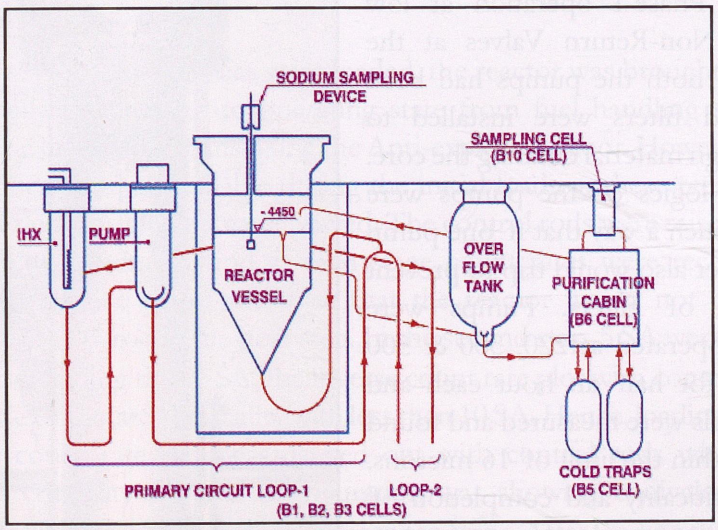
### *Sodium Purification*

*Oxygen and hydrogen react with sodium and form oxides, hydrides or hydroxides. The solubility of these compounds in sodium decreases with decreasing temperature. Sodium is purified by cold trapping. In a cold trap, impure sodium from the system is passed through a wire mesh packed inside the body of the cold trap. The cold trap is cooled from outside. As sodium gets cooled, the impurities get precipitated and get trapped in the wire mesh. The purified sodium passes through the riser tube and joins the system. An economizer is usually a part of the purification system, i.e. the incoming hot sodium is cooled by the outgoing cold sodium in order to decrease the cooling load on the cold trap and reduce its size.*

*Sodium purity is expressed in terms of 'Plugging temperature'. Liquid sodium is passed through a Plugging Indicator and is progressively cooled. The temperature at which the compounds start precipitating and plugging the orifices of the Plugging Indicator, resulting in a flow reduction of 20%, is called Plugging Temperature. Reactor grade sodium is to be maintained at a plugging temperature of <120 °C, which corresponds to oxide impurity of < 1 ppm and hydride impurity of < 60 ppb.*

## Primary sodium system

The primary sodium storage tank was filled after bringing down the oxygen level in the cover gas by three cycles of vacuum degassing at 300°C, breaking the vacuum and filling with argon. Each cycle of operation lasted for three days. Sodium was charged into storage tank in seven charging operations with storage tank and its filling line preheated and maintained at 200 ± 25°C. The continuous level probe functioned extremely well during filling operations. Each filling operation took 45 to 60 minutes and was quite smooth. The expected and achieved levels after each charging were exactly matching. A total of 35 $m^3$ of sodium was charged in the storage tank.




*Schematic of the primary sodium circuit*

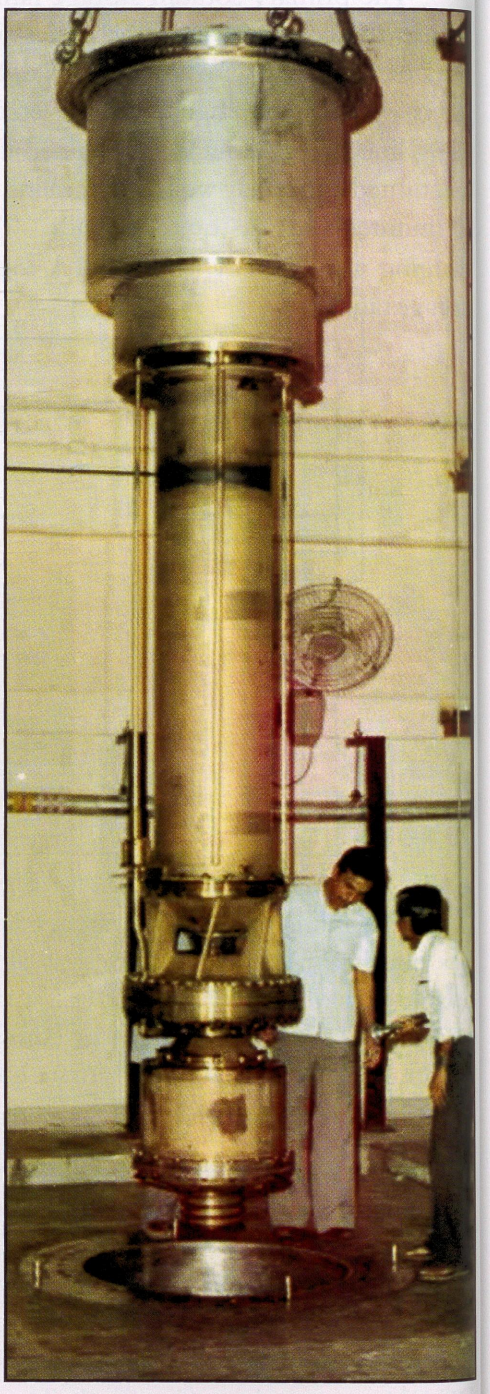
The filling of primary system from the storage tank was preceded by extensive leak-checks on the top closures of all capacities using diverse methods like soap bubble, balloon, argon detectors and helium detectors. All the modifications recommended by safety authority consequent to NaK leak incident from secondary cold trap were carried out in primary cold trap also. Purification lines including both cold traps were preheated and maintained between 150 & 200°C.

The primary capacities and pipelines were heated to ~200°C using the preheating circuit which circulates nitrogen around the double envelope. For this purpose, in the PHEC system, air used earlier was changed over to nitrogen for circulation. After priming the EM pump by pressurising the storage tank to 300 mb, all the capacities were filled. The sodium in RV after reaching the overflow pipe level (-4450 mm) started overflowing to the overflow tank. The purification circuit was filled using the DC EM pump in Overflow Tank. The system temperature was raised to 250°C and

the plugging temp was measured to be 124°C. Sodium temperature was increased to 350°C, sodium was purified and plugging temperature was brought down to less than 120°C. The instrumentation of the primary sodium system performed extremely well during both the commissioning and operation phases.

## Primary Sodium Pumps

During commissioning of primary system and Phase-I operation at low power, the Non-Return Valves at the discharge of both the pumps had been removed and filters were installed to prevent foreign material entering the core. The control logics of the pumps were modified in such a way that if one pump trips, the other also would trip to prevent backwashing of filters. Pumps were started and operated at 220, 300 & 500 rpm speeds for half an hour each and vibration levels were measured and found to be well within the limit of 16 microns. After first criticality and completion of experiments at zero power, it was necessary to remove these filters and reinstall the NRV. The pumps were taken out using SF-I, decontaminated of sodium and dismantled. On visual examination, black/brown spots were found on some components, e.g. shaft, strainer and intermediate piece. These spots did not disappear even after the surface treatment. The torque required for loosening the nuts of the fasteners was more than the torque applied at the time of their tightening during assembly of the pump. The pumps were reassembled with NRV and reinstalled.



*Primary Pump removal*

## Approach to first Criticality

The Antimony-Beryllium neutron source was loaded in the Nickel reflector region of the core to get adequate flux for the detectors in addition to inherent neutron source. After ascertaining sufficient count rates in neutron detector channels, fuel loading towards the first Criticality was started on October 2, 1985. The main rules followed were:

- Loading the SA from centre to periphery
- Loading towards the control rods
- Loading towards the source
- Loading symmetrically

Once seven subassemblies were loaded, the reactor was brought to operating state. Normally, changeover to operating state from fuel handling state requires freezing of the LM Seals and installing the Anti-explosion Floor. However, due to the large sub-critical margin available during the initial loading, these requirements of freezing and floor installation were waived. The control rods were raised in steps, till they were fully withdrawn, and at each stage count rates were recorded. It was ensured from inverse count rate plot that the reactor would not go critical by withdrawing control rod in the next step. In the second step, 5 SA were to be added. However, after adding the 9$^{th}$ SA, the inverse count rate plot with control rods down was showing extrapolated criticality with less than 10 SA. Hence, loading was stopped and inverse count rate plot was carried out with control rods withdrawn. The extrapolated criticality for this configuration was showing equivalent to 22 SA. Thereafter, loading was restarted and done in ten steps. After loading the 22$^{nd}$ fuel SA, LM seals were frozen and anti-explosion floor was kept in place. The control rods were raised in predetermined steps and reactor went critical at 2210 hours on October 18, 1985. Control rods A, B, C, D & F were at 344 mm & criticality was achieved when Control rod E was raised to 356 mm.

## The 'partial' SCRAM incident

On October 19, 1985 when the reactor was made critical for the second time as a part of low power reactor physics experiments, an unintended partial SCRAM incident took place. Control rods A, E and F had dropped from the operating position of 346 mm and brought the reactor to shut down state without SCRAM annunciation in control room. This was promptly noticed by control room personnel by watching reduction in power level and position of control rods from potentiometers. Hence, manual SCRAM was ordered immediately by the operator. On investigation, the partial drop was found to be due to non-persistent noise signal

***Criticality***

*In the fission reaction if one can control the chain reaction such that out of 2.5 neutrons that are released in the fission, only one neutron is used in the fission, then the reaction rates and hence the power remains constant. This is called controlled chain reaction. In the nuclear reactors, fuel, coolant, clad, the moderator and other structural materials are arranged in such a way that controlled chain reaction take place. In this arrangement, the reactor power is constant. This state is called critical state. When reactor reaches this state, we call reactor attained 'criticality' or reactor is 'critical'*

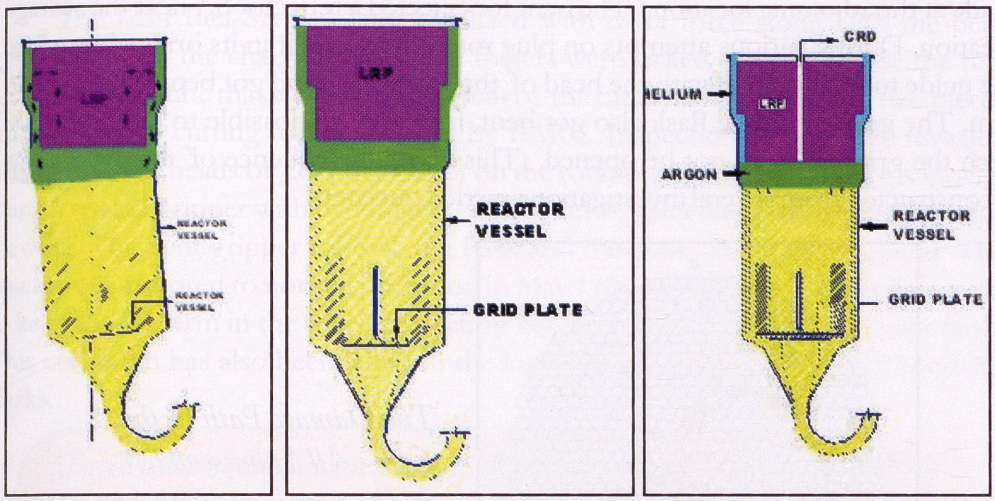
of pulse duration less than the SCRAM circuit latching time and difference in decay characteristics of electromagnets. Several improvements were made to avoid its recurrence, the major being the adjustment of latching time of SCRAM circuit to 40 ms taking into consideration the least response time of the EM coils. Also, additional LOR was incorporated in case of reduction in the current in EM coil of any of the CRDM.

## Noise Pick-up by Neutronic Channels

The number of spurious SCRAM due to noise pickups was quite frequent in the initial days of reactor operation. After a lot of observations and analyses, it was concluded that the operation of RCB crane was one of the main sources of noise pick-up by neutronic channels. It was also noticed that there were noise pick-ups by the neutronic channels during lightning. These observations gave additional clues for the noise pick-up. The noise pick-up was minimised by physically segregating signal cables & power cables, providing metallic conduit to neutronic instrumentation cables to improve shielding, improving the cable joints at RCB cable penetrations and improving grounding system of power & signal cables. After all these modifications, the performance of the neutronic instrumentation system was satisfactory.

## Raising the temperature and Reactor Vessel deflection

The first criticality was followed by several physics tests related to measurement of CR worth, SA worth, feedback reactivity coefficient related to power, coolant void, coolant temperature, coolant flow & cover gas pressure, kinetic experiments, flux tilting and flux measurement experiments. For the measurement of temperature coefficient of reactivity, the primary system temperature was progressively being increased to 375°C using the reheaters in secondary loops. The grid plate was found to be getting displaced laterally, as seen from the readings of the DMD. When the sodium temperature was raised isothermally to 270°C, the grid plate shift reached 8 mm. Further raising the sodium temperature was stopped for investigation.



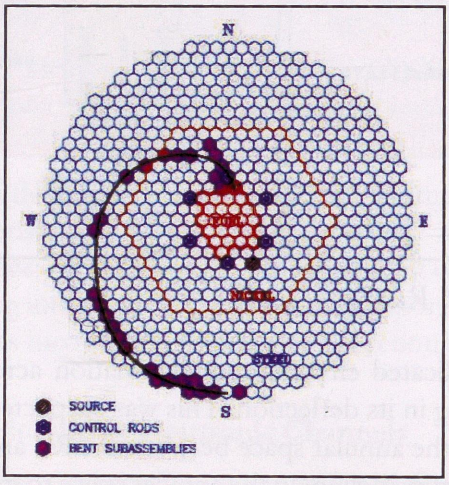

*Deflection of Reactor Vessel*

The RV temperature pattern indicated circumferential variation across the diameter in the cover gas space, resulting in its deflection. This was suspected to be due to convection currents of argon in the annular space between the RV and LRP. The convection was suppressed by admitting helium in the annular space to maintain a density gradient in the cover gas space. This works on the simple principle that heavy argon, getting heated up by the hot sodium, tries to rise up and as it encounters the less dense helium at the top and cannot raise any further. The convection gets only localised. In order to maintain the density gradient continuously, helium was admitted at the top and argon through the argon pipe above the sodium level. The gas mixture was continuously evacuated using the CRD system in order to maintain the cover gas pressure as well as the density gradient. This was found to be effective in bringing down the grid plate deflection and the temperature gradient in RV. The primary temperatures could then be safely raised to 375°C and reactor physics measurements completed.

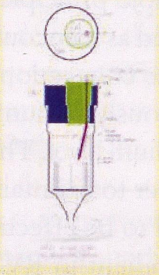
## The Fuel handling incident

In May 1987, a fuel SA was required to be transferred to the periphery from the core as part of reactor physics experiments to find out its reactivity worth. The SA was not fully withdrawn to the transfer position in the guide tube. The interlock preventing the rotation of plugs when the SA was not in transfer position had remained inadvertently bypassed. Due to the rotation of plugs with the SA foot protruding below the level of heads in the core, the heads of SA on the rotational path got slightly bent as also the foot of the SA. The bent foot made it impossible to lower the SA into the storage location. In a bid to lower the SA, several attempts were

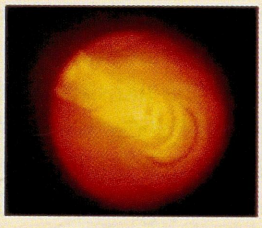
made at the adjoining locations. The bent foot ejected a reflector SA near the storage location. During various attempts on plug rotation to load it in its original location, the guide tube interfered with the head of the ejected SA and got bent by about 320 mm. The gripper of the flask also got bent, making it impossible to lower the SA; even the gripper could not be opened. (This complex sequence of the events was reconstructed from several investigations carried out later).




*The Damage Path in the Core- All Subassemblies on the path were Nickel or steel reflectors*



Bent Guide Tube in Reactor Vessel



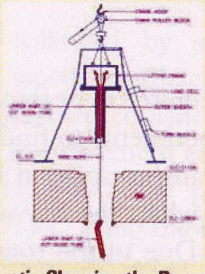
Reflection of Guide Tube in Sodium



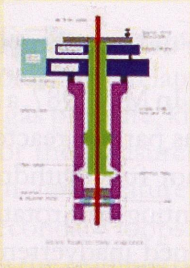
Guide Tube Cutting Machine Being Installed



Removing the Bottom Piece of Guide tube after Cutting



Schematic Showing the Removal of Guide tube in two Pieces after Cutting



Guide tube Cutting Sequence

*The Guide Tube Incident (1987-89)*

The bent fuel SA was later extracted with extra force along with the bent gripper tube of the charging flask. The fingers were jacked open to release the fuel SA. The guide tube that was not coming out of the canal had to be cut into two pieces using a special cutting tool developed by BARC. Inspection of the core revealed damages to the heads of 28 reflector SA on the rotational path. These were removed using a special gripper with two fingers, with sodium drained and using periscope for viewing. The bent gripper tube of the flask and the guide tube were replaced. The reactor was restored to normal operation in May 1989. A mechanical swivelling lock, to keep the SA firm in the transfer position has since been installed in transfer flask. This condition has also been wired to the logics of plug rotation and fuel handling flasks.

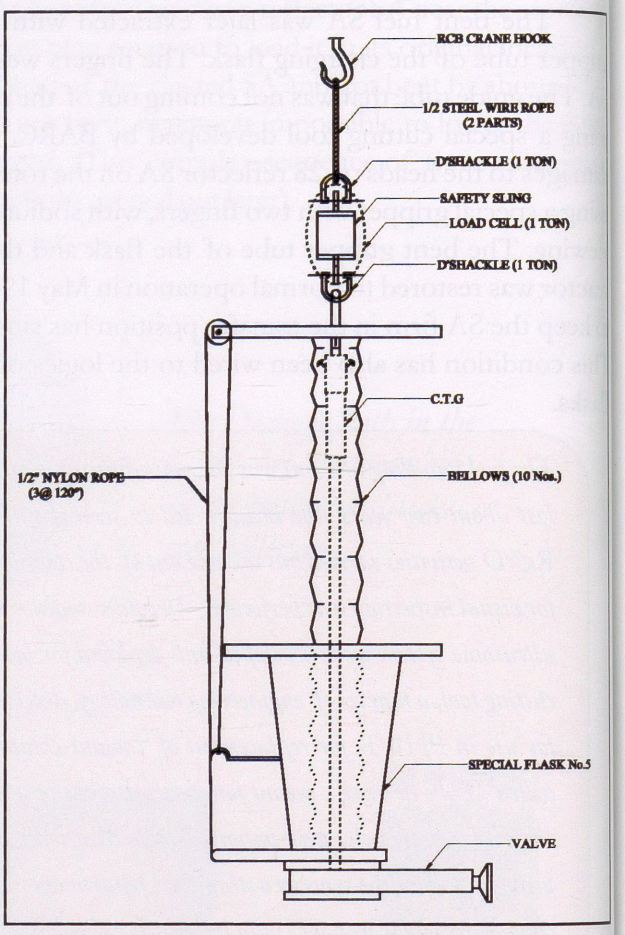
*The fuel handling incident was the most difficult challenge faced by FBTR. Though we lost about two years, the incident led to several spin-off benefits due to the flurried R&D activities carried out to come out of the complex situation. The use of Periscope for visual inspection was perfected. An under-sodium ultrasonic scanner and a novel air ultrasonic system were developed and deployed for measuring the guide tube bend. The cutting tool, a marvel of engineering technology, developed by BARC was later modified for use in PHWR for replacement of coolant channels. In addition, a special device called `Tube de visee`, meant for measuring the positions of SA heads with respect to the FH canal axis, was imported from Rapsodie and commissioned. The incident brought to light the synergy within the Department. The successful come back from this complex problem won for India international acclaim in the fast reactor community.*

## Dropping of Capsule Transfer Gripper from RCB crane

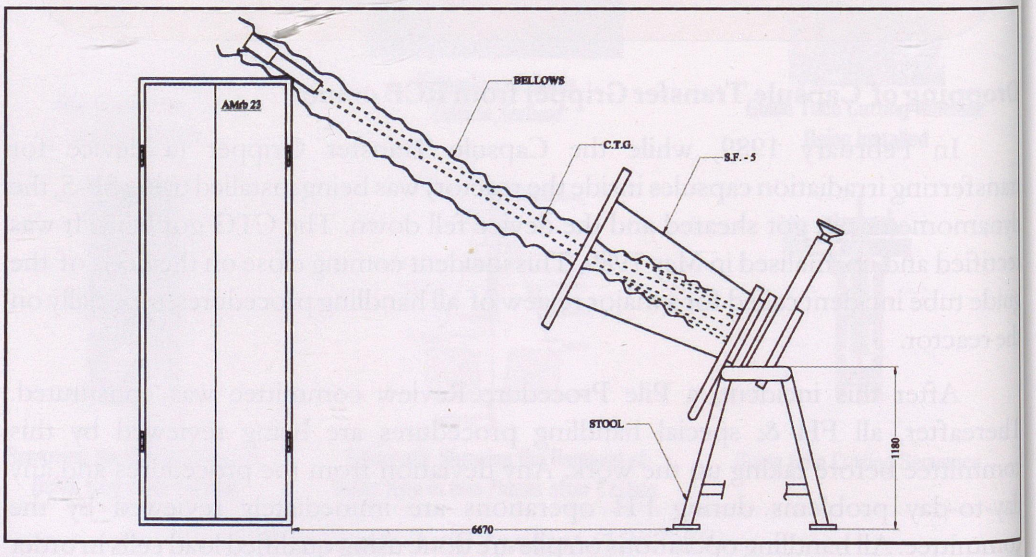
In February 1989, while the Capsule Transfer Gripper (a device for transferring irradiation capsules inside the reactor) was being installed using SF-5, the dynamometer pin got sheared and the device fell down. The CTG got bent. It was rectified and normalised in May 1989. This incident coming close on the heels of the guide tube incident called for a major review of all handling procedures, especially on the reactor.

After this incident, a Pile Procedure Review committee was constituted. Thereafter, all FH & special handling procedures are being reviewed by this committee before taking up the work. Any deviation from the procedures and any day-to-day problems during FH operations are immediately reviewed by the committee. All handling operations on pile are done using qualified load cells in order

to monitor free lowering by self-weight during loading and any obstruction to the components during extraction. All procedures are prepared in the form of check-lists and signed by the agency concerned. Any inevitable bypassing of interlocks is critically reviewed by Station Management and duly entered in the Station 'Jumper Register'. These measures have gone a long way in making FH operations smooth and incident-free since 1989.



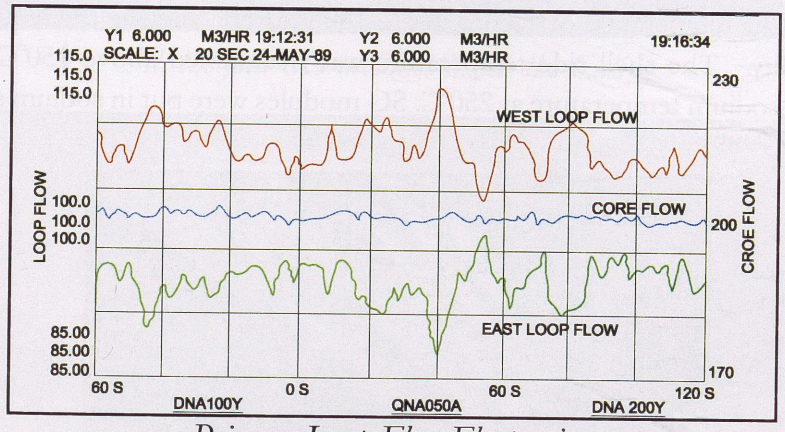

*SF-5 with CTG*



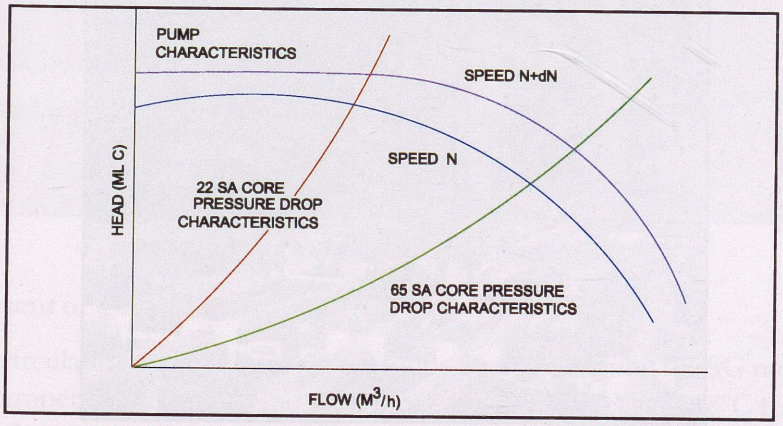

CTG After Fall

## Flow fluctuations in Primary sodium system

During reactor operations up to a power level of 500 kWt, the reactor tripped many times due to LOR initiated by low sodium flow in individual primary loop, set at 80% of the normal flow. This was investigated and found to be due to inherent periodic flow fluctuation in individual loops. Fluctuation in one loop is a mirror image of the other loop whereas the flow through the reactor is practically constant. It was concluded that high resistance of the small core and operation of the hydraulically coupled pumps in the flat region of their curves were making individual loop flows very sensitive to speed fluctuation. Any slight malfunction of control system results in wider flow fluctuation, resulting in LOR. The set point for LOR on low loop flow was reduced to 60% and a SCRAM input (low flow through core ($Q_{mini}$)) was introduced. The speed controls of the Ward Leonard drives were improved to maintain the speed of the pump within ±1 rpm of the set speed; a major achievement by the control engineers.




*Primary Loop Flow Fluctuation*

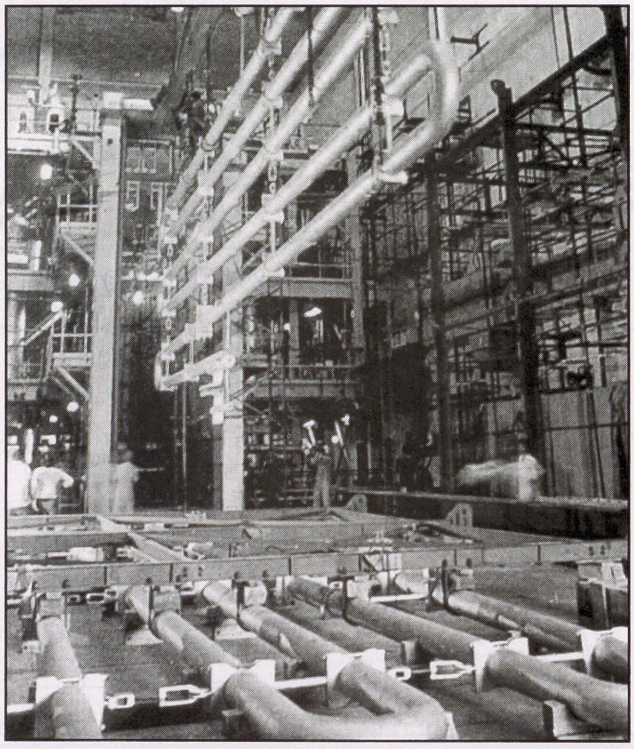



*Pump characteristics showing operating region for small and full core*

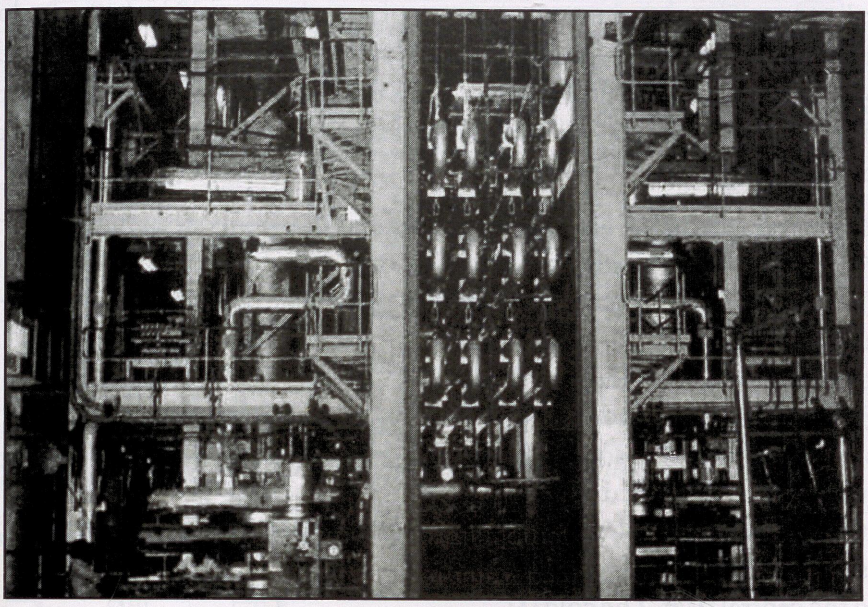
## Connection of SG modules to sodium headers

After completion of Phase-I operation of the reactor, balance construction activities like connection of SG modules and commissioning of flooding system & steam water system were taken up. While connecting SG modules of east loop frozen sodium was seen in the inlet and outlet sodium headers. Almost half the diameter of each header was found filled with sodium. This was due to passing of the isolation valves. Sodium was scooped out, the headers were thoroughly cleaned and SG modules were connected. However, this caused increase in plugging temperature to the order of 175° C, when east SG modules were filled. Based on this experience, while connecting the west modules, sodium in the west header was completely drained after melting the sodium and collected in SS drums. For this purpose, a hole was made in the blank provided in place of rupture disc.

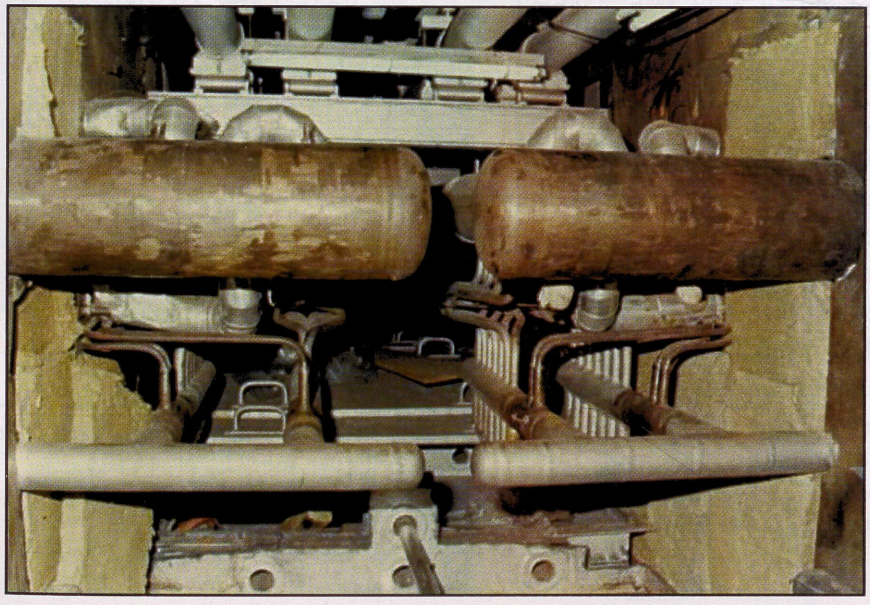
Sodium was filled in SG after preheating the SG modules to 150°C by circulating hot water through the tubes. The 'Pumping power' of MBFP was utilised for raising the temperature of feed water to 180°C after achieving the required feed water quality. The shell side temperature was maintained above 150°C and the secondary sodium temperature at 250°C. SG modules were put in sodium service in November 1989.



*Erection of SG modules inside the SG cabin*



*All the four SG modules inside the Casing*



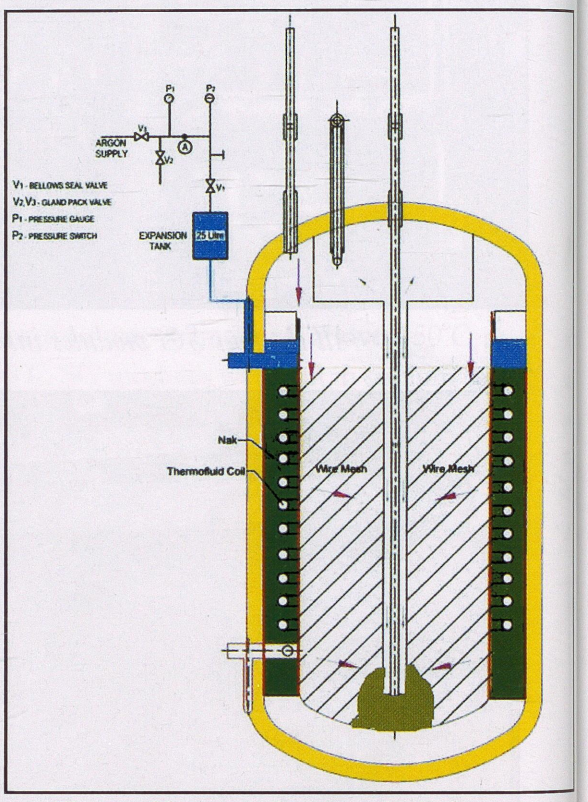
*Sodium Headers of Steam Generator*

## Replacement of East Cold Trap

On circulating sodium through the loops after connecting the SG modules, the plugging temperatures of east and west loops went up to 180 & 168°C respectively, obviously due to pick- up of oxides and hydrides from the residual sodium in the

sodium headers. The plugging temperatures of west loop could be brought down t 105°C in 40 hours, whereas it took about 60 days for the east loop to get purified fro a plugging temperature of 180°C to 105°C. It is estimated that east and west cold trap would have trapped about 6.1 kg and 0.4 kg of sodium impurities respectively durin the above works.

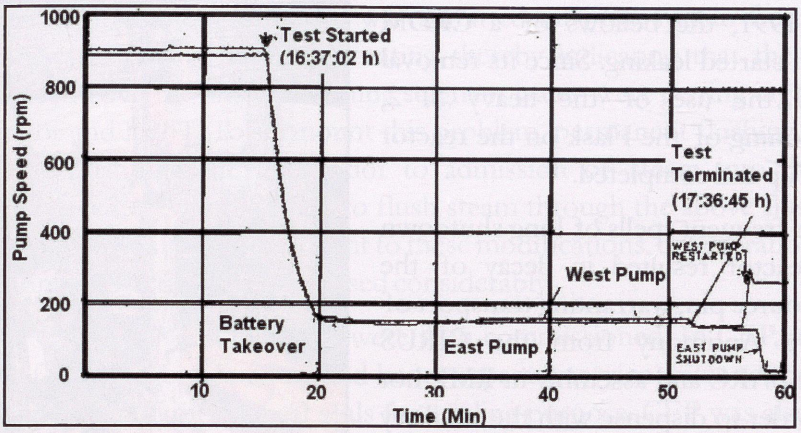
After connecting the east modules, frequent flow reductions were observed i cold trap indicating its partial choking. On one occasion, there was full chokin Though partial flow could be restored by heating, it was felt advisable to replace the cold trap. Though the cold trap had not exhausted its design load of impurities, it was suspected that impurities could have plugged the passage between the dished end and the central riser. The cold trap was taken out and replaced by a modified cold trap. The modified cold trap has increased diameter of central rising pipe, provision of lateral holes at the cold point location of the central pipe and improved thermo wells for better thermal contact. The sodium in the east loop was purified by the new modified cold trap. The removed cold trap was decontaminated, modified and is kept as a spare.




*East cold trap indicating choked portion*

***Primary Coast Down and Battery Take-Over Test***

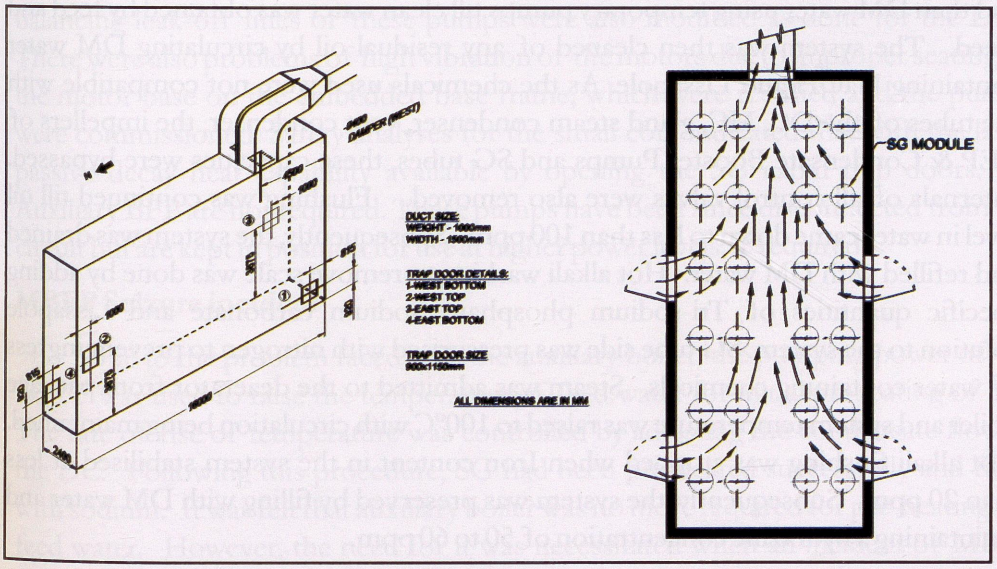
*A test was conducted in December 1989 to study the coast down characteristics (speed and flow) of the pump and verify the ability of the batteries to run the pumps as per the design intent. The pumps coasted down from 900 to 200 rpm in about 4 minutes. The batteries took over at a pump speed of about 175 rpm and operated the pumps for 37 minutes at 150 rpm at the time of termination of the test. The core flow during battery operation was about 50 $m^3/h$; adequate to remove decay heat during station blackout.*




*Primary pump coast down and battery take over test*

## Operation up to 1 MWt with SG trap doors open

Opening of the trap doors of SG casing provides the ultimate heat sink for decay heat removal. Tests were conducted to find out the heat removal capacity by natural convection of air through the SG casing. Total heat removal at 375°C of isothermal sodium temperature was 1 MWt, which is significantly higher than required for decay heat removal. For getting safety clearance for this operation, SID system was installed for monitoring sodium leak inside the SG casing and its actual response was verified by burning a few grams of sodium inside the casing.

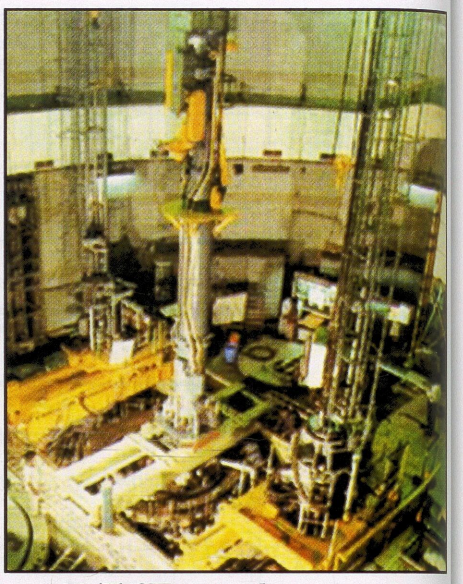



*Passive Heat Removal through SG casing* *Trap doors of SG casing*

In 1991, the bellows of a CRDM lower part started leaking. Since its removal warranted the use of the heavy SF-2, commissioning of the Flask on the reactor was taken up and completed.

The frequent spells of long shutdown of the reactor resulted in decay of the neutron source pin, warranting transport of radioactive Antimony from the CIRUS reactor in BARC and assembly in RML hot cells. In order to dispense with the auxiliary neutron source, two pre-start up channels using high sensitivity boron lined proportional counters were incorporated.

Reactor was operated in 1991-92 at 1 MWt, with the longest run of 33 days in Jan-Feb 1992. During this period, intermediate power physics experiments were completed.



*30 T SF-2 on the reactor for removal of CRDM lower part*

## Steam Water Circuit

Initially, the condensate and feed water systems were flushed with raw water and then DM water using temporary pumps till clean water was obtained by feed and bleed. The system was then cleaned of any residual oil by circulating DM water containing NaOH and Lissapole. As the chemicals used were not compatible with the tubes of the DC, MC, gland steam condenser, vent condenser, the impellers of CEP & Condensate Booster Pumps and SG tubes, these capacities were bypassed. Internals of all control valves were also removed. Flushing was continued till oil level in water came down to less than 100 ppm. Subsequently, the system was drained and refilled with DM water. Hot alkali washing to remove scale was done by adding specific quantities of Tri-sodium phosphate, sodium carbonate and Lissapole solution to the system. SG tube side was pressurised with nitrogen to prevent ingress of water containing chemicals. Steam was admitted to the deaerator from package boiler and system temperature was raised to 100°C with circulation being maintained. Hot alkali flushing was stopped when Iron content in the system stabilised at less than 20 ppm. Subsequently, the system was preserved by filling with DM water and maintaining a hydrazine concentration of 50 to 60 ppm.

Steam lines were flushed with steam from auxiliary boiler. Cyclic flushing was carried out to loosen the mill scales by subjecting the pipe lines to thermal cycling. It

was observed that whenever steam was admitted into the system from auxiliary boiler, the water quality was deteriorating thereby indicating that the corrosion products, formed in the lines containing stagnant water, were getting ingressed into the deaerator and LPFT. To surmount this problem, permanent flushing lines were provided to flush out the lines prior to admission of steam into the system. Operating procedure was modified to flush steam through the above flushing lines till clear steam comes out. Subsequent to these modifications, deterioration of water quality during preheating was minimised considerably.

The CEP, CBP and the DLP were then commissioned. In CEP, mechanical seals were found to get heated up and hence separate service water connection was given to the jacket of mechanical seals for cooling purpose. CEP was also found to cavitate when the DC vacuum was higher than -0.75 kg/cm$^2$ due to inadequate cross sectional area of the suction strainers. They were replaced with 'Y' type strainers of 300% flow area and the diameter of the suction line also was increased to 300 from 150 NB. After these modifications, the cavitation problem was not encountered. In DLP, after about 300 hours of operation, the balancing discs of both the pumps developed severe wear resulting in seizure of the pumps. The balancing leak-off line had to be rerouted to avoid this problem. Graduation markings were provided at the free end of the pump shaft to monitor the wear of the balancing disc.

As per the original design, two Auxiliary BFP capable of developing a head of 40 kg/cm$^2$ had been provided for decay heat removal in the event of offsite power failure. These pumps fed by class-III DG power supply were commissioned. The balancing leak-off lines of these pumps were also modified as done for the DLP. There were also problems of high vibration of the motors due to improper seating of the motor base on the embedded base frame, which were rectified and the pumps were commissioned. Safety analyses for the small core indicated that with the large passive decay heat capability available by opening the SG cabin trap doors, the Auxiliary BFP are not required. These pumps have been since disconnected from the circuit but are kept in position for use at higher power levels, if required.

### MBFP Seizure incident

Due to the problem faced with the auxiliary boiler, the pumping power of the MBFP was used to raise the temperature of feed water for initial preheating of SG. The rate of rise of temperature was controlled by adjusting the condensate flow to the DC. Following this procedure, SG had been pre-heated successfully and filled with sodium. It was felt that auxiliary boiler was no more required for pre-heating the feed water. However, the need for it was necessitated when an incident of MBFP seizure took place in 1992. While pre-heating the feed water using the pumping power of MBFP-A, there was a problem in the condensate circuit. Though, the

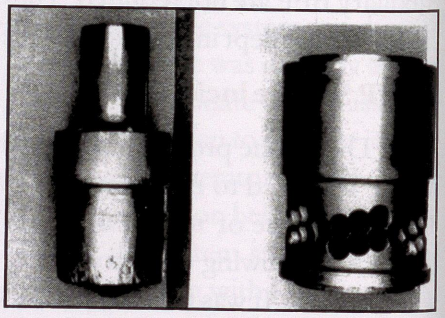
condensate circuit was taken out of service, MBFP-A continued to run in recirculation mode for feed water heating. The restarting of the condensate system resulted in admission of cold condensate into the deaerator causing sudden collapse of its pressure. This has resulted in reduction in NPSH available for MBFP operation; losing the 'balancing pressure' and its seizure.

*In multi-stage pumps with large discharge heads, the pressure difference between the suction and discharge sides of the impellers causes a net unbalanced force which will tend to push the rotor to one side. This unbalanced force is compensated by balancing devices, usually balancing discs, pistons or drums through which a small part of the liquid from the discharge side is allowed to 'leak off'. The leak-off liquid is normally led to the suction line of the pump. If the leak-off flow is inadequate, it will result in the pump losing the balance, damaging the thrust bearing and/ or impellers.*

Upon dismantling, severe damage to pump internals and thrust bearing was observed. The re-assembly took considerable time and effort as dismantling of stuck impeller had to be done with utmost care. A task force studied the problem in depth and suggested many modifications to preclude the recurrence of such incidents. Modifications to improve available NPSH at pump suction were carried out. The balancing leak-off line, which was earlier joining suction of the pump, was rerouted to the deaerator and continuous cold injection was provided at the suction to improve available NPSH. Thereafter, the performance of the pump improved during transient conditions. Additional recirculation line was also provided to avoid operation of the pumps at low flows. Operating procedures were modified and feed water heating by package boiler steam was strictly adhered to. This incident resulted in a delay of eight months in putting the SG in service.

### Steam Bypass and Dump Control Valves

The performance of the steam / water system was ham-strung by the problems encountered in the functioning of the steam bypass and dump control valves. Typical problems encountered were erratic and sluggish operation of these valves, inability to close fully with the process pressure of 125 $kg/cm^2$ and cavitation-erosion of control valve plug. The valves were operating under severe conditions.



*Eroded plug and cage of bypass valve*

During power rising phase, these valves were facing two phase flow condition and large differential temperature at certain power levels of the reactor. The inlet conditions were changing very rapidly leading to differential thermal expansion in the valve resulting in seizure of the valve plug. This was overcome by adopting a modified power rising procedure in which the power of the reactor was stabilised at various intermediate power levels to thermally soak the internals of the valve. The Integral Bypass valves of the isolation valves of these valves, to this day, continue to serve as manual control provisions. The motive oil circuit also needed several modifications. It is planned to replace all the control valves with pneumatic valves of better design.

### DM Water

Initially, the DM plant capacity was 1 $m^3$/h. The inadequate capacity of the DM water plant in conjunction with less storage capacity has resulted in bottlenecks during start-up as well as inability to control silica by feed and bleed during heating up phase. The DM water was then transported from MAPS in a tanker. The DM plant was later augmented to 5 $m^3$/h with extra storage capacity.

The commissioning and operation of the steam/water system has given us a rich experience in surmounting various problems. Certain short term solutions have worked out with limitations. Later on, major modifications were carried out in this system to increase the availability of the system and also to survive the transients.



*Transportation of DM water*

### Complimentary and augmentation of shielding

All sodium, argon and nitrogen lines penetrate into B-cells in RCB. Complementary shielding was provided at locations wherever there was loss of bulk shielding due to penetrations to minimise radiation doses in the accessible areas and to restrict the radioactivity build-up in the secondary sodium. In all, about 30 T of complementary shielding has been provided at 71 locations. During the erection process, adequate care had to be taken to avoid any damage to pipes and thermocouples.

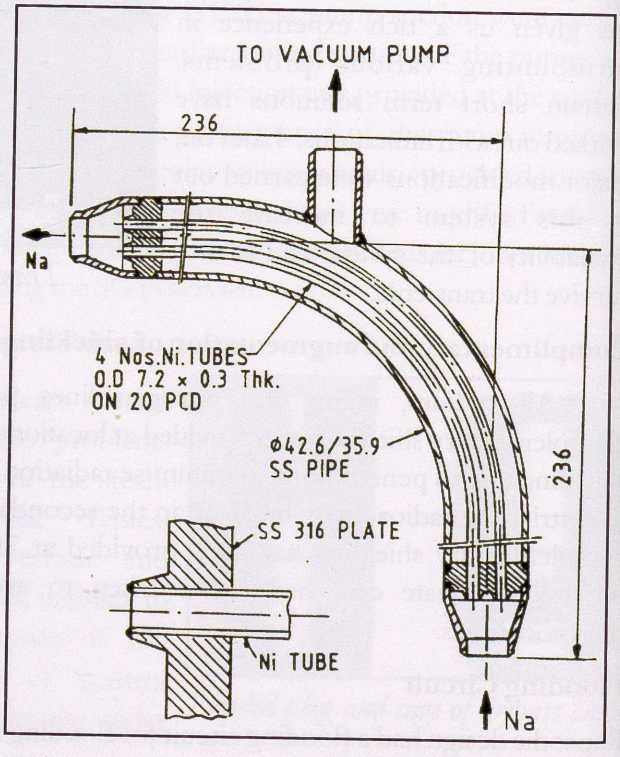
### Flooding Circuit

Rapsodie design had a flooding circuit for flooding the steel vessel with sodium in the event of leak from RV and its double envelope. This was not included in the original

design of FBTR. However, this was added later at the insistence of the safety authority. The original layout had no provision for the flooding tanks and hence, they were accommodated in the ground floor of the MB. Both the flooding tanks were initially installed in Engineering Hall-I, where they were used for sodium purification. These tanks were having 0.5 T of residual sodium. They were transported from Engineering Hall-I to MB of FBTR & erected. The tanks were filled with filtered sodium using the transportation tanker. Each tank was charged with 34.5 $m^3$ of sodium in 14 transfer operations to fill both tanks. Both tanks were filled with 1.5 $m^3$ extra capacity of sodium above high level probe for periodic testing as recommended by safety authority. After filling & commissioning of manual and auto flooding tanks, sodium was injected into RV and 5% level rise (310 liters) in OFT was noticed to confirm the healthiness of the flooding lines.

## Steam Generator Leak Detection system

The SG tubes have high pressure water/steam inside and sodium outside. Any leak of water/steam into sodium has to be detected at the incipient stage itself and action taken. If not detected early, it may escalate into a bigger leak and damage the SG. The detection of any water/steam leak into sodium is carried out by online measurement of the hydrogen concentration in sodium at ppb levels. Sodium sampled from SG is heated to 450°C in a reheater and is then passed through the thin tubes of the Nickel diffuser. The shell side of the diffuser is kept under high vacuum using sorption pump and SIP. With sorption pump, vacuum is pulled up to $10^{-3}$ torr & with SIP vacuum is increased to $10^{-8}$ torr. The Mass Spectrometer provided in the vacuum circuit is tuned to measure the hydrogen concentration. The system was commissioned and the required vacuum could be attained in about 10 minutes.

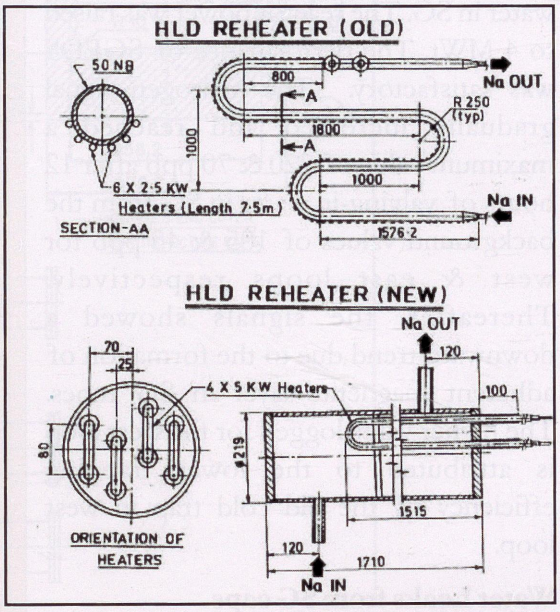



*The Nickel Diffuser of SGLDS*

On July 11, 1991, a small sodium leak was observed in east loop SGLDS sampling line valve due to failure of bellows. The valve was replaced.

## Failure of SGLDS Reheaters

The reheaters provided to maintain the temperature of sodium at the inlet of the nickel diffuser at 450±1°C consisted of 50 mm pipe over which six $MgO_2$ insulated, SS sheathed surface heaters had been mounted. During the course of commissioning, there were several (three on west & two on east loop) failures of heater elements. To investigate the cause, thermocouples were installed on the sheath of the heaters and temperatures at different points were noted. In the proximity of pipe hangers, the heater could not be laid adequately close to the pipe line resulting in local hot spots. The sheath temperatures at these locations were more than 600°C which is the maximum limit. The Reheaters were then redesigned using immersion type of heater elements, fabricated and installed.




*Reheaters of SGLDS*

After filling SG with sodium, the SGLDS was put in service and 1 $m^3$/hr flow per module was established. $H_2$ injection was carried out at 180°C, 250°C, 300°C & 400°C, and the signals of Mass Spectrometers were recorded. The detector starts responding after 3 minutes of injection which is the transit time from injection point to detector.

The response of the system is very poor (20%) at a sodium temperature of 180°C indicating that most of the injected gas remains undissolved in sodium and gets released into the cover gas. At 250°C and above, the response is very good. With this observation, it was decided to admit water into SG only at sodium temperature > 250°C. As a long term solution to admit water at 180°C itself, Hydrogen in Argon Detector developed at RCL was installed in FBTR.

Later, an Electrochemical Hydrogen meter developed at RCL was installed in SGLDS for monitoring Hydrogen in sodium. At a higher sodium temperature of 400°C, the response of both diffusion type meter and Electro Chemical Hydrogen Meter are excellent.

## Valving in water in SG

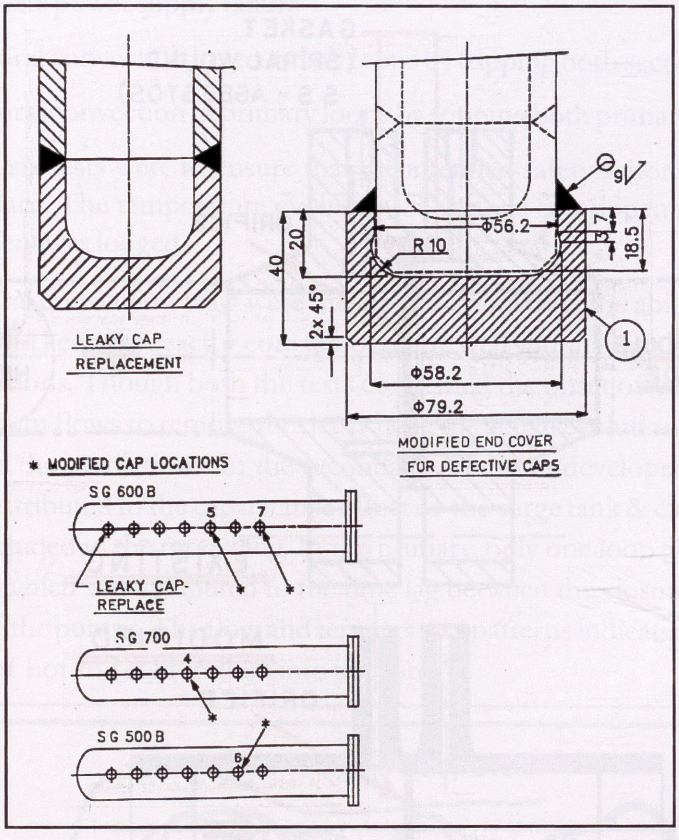
After carrying out all the necessary modifications in steam water system, the system was put in to regular operation in January 1993 after valving-in the feed water in SG. The reactor power was raised to 4 MWt. The performance of SGLDS was satisfactory. The hydrogen signal gradually increased and reached a maximum value of 220 & 70 ppb after 12 hours of valving-in water in SG from the background values of 105 & 45 ppb for west & east loops respectively. Thereafter, the signals showed a downward trend due to the formation of adherent magnetite layer in SG tubes. The higher value logged for the west loop is attributed to the lower trapping efficiency of the old cold trap in west loop.

### *Feed water chemistry*

*The once through type steam generator (as compared to recirculation type) requires high pure feed water to minimise scaling and corrosion. The feed water used, therefore, in absence of blow down provision in SG, should have very low Total Dissolved Solids (<50 ppb) and Dissolved Oxygen (<10 ppb). The pH of the system is maintained between 8.8 to 9.2 and cationic conductivity below 0.5 μS/cm at SG inlet. These specifications are met by providing an online Condensate Polishing Unit for full condensate flow and by dosing Ammonia at CPU outlet and hydrazine at feed pump suction for pH and $O_2$ respectively as a part of all volatile treatment. A number of online meters are employed to monitor the feed water chemistry parameters like sodium, silica, dissolved oxygen, pH, cationic conductivity and chloride. The operation of SG without any tube leak for more than 25,000 hours so far testifies the feed water purity and steam/ water system performance.*

## Water Leaks from SG caps

After 70 hours of operation at 4 MWt, secondary sodium auxiliary system leak and secondary sodium SID ion current high alarms appeared in control room. Sodium leak detector in trays of SG was also found to be in actuated condition. On investigation after opening SG trap doors with proper precautions, it was noticed that there was a water leak from SG to its casing. Immediately, the reactor was shut down by manual LOR. Water leak took place due to a linear pinhole defect in the end cap of one of the orifice assemblies at SG inlet. All similar caps (35 numbers) were ultrasonically inspected and four more were found having indication of linear defects. The leaking cap was replaced and additional covers were welded on defective caps and SG modules were re-qualified. After this, the SG was valved-in and power was raised to 7.5 MWt on March 31, 1993.

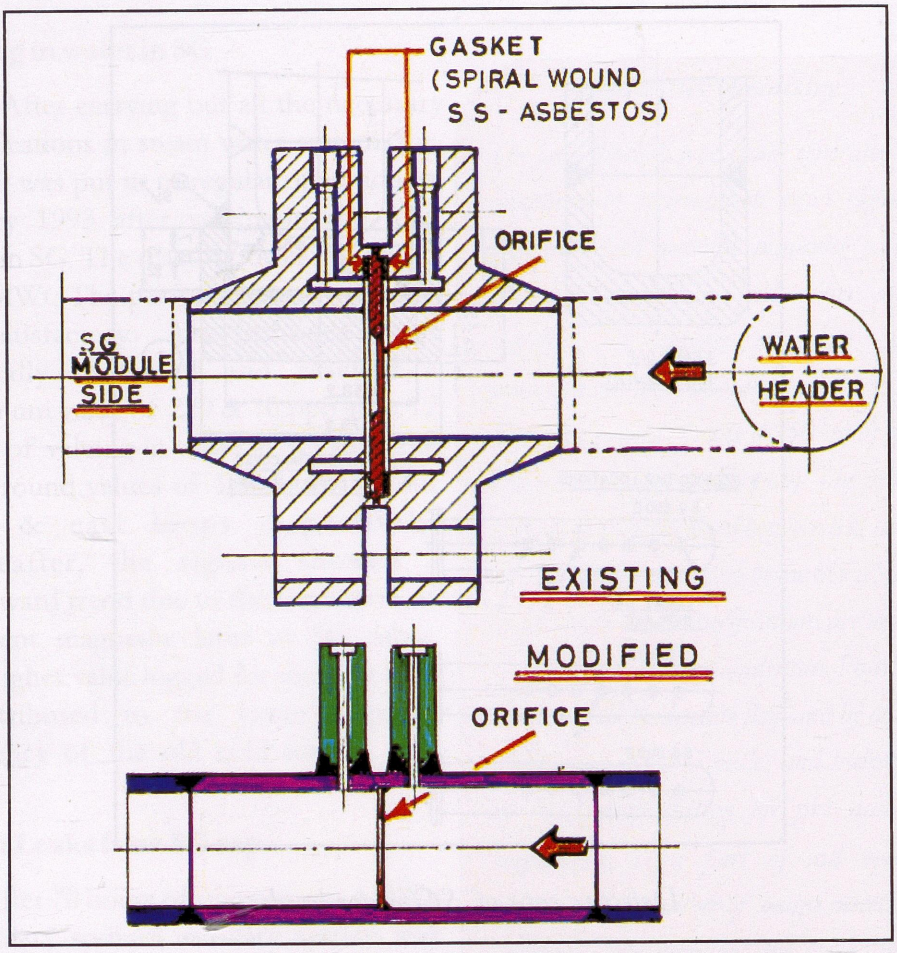



*Defective caps of SG water header*

## Water Leaks through the Orifice meters

Reactor was operated at 8 MWt in various spells in 1993. In August when the reactor was operating at 8 MWt, feed water was found leaking again in SG casing. The reactor was shut down by ordering manual LOR. The leak was found to be from orifice flanges. Investigations revealed that the leak tight orifice flanges under ambient conditions tend to develop leak under operating conditions as a result of differential thermal expansion between the water sub-headers and SG modules. All the orifice flanges were replaced with welded spool with integral orifices and reactor was restarted.

After achieving sustained operation at 8 MWt during September-October for 15 days with all systems working satisfactorily, core was expanded by adding 24$^{th}$ fuel SA in November 1993 and irradiation campaign number 2 was started. Power was raised to 10.2 MWt for the first time in December, 1993. Thus, it took eight years from first criticality to raise the power to 10.5 MWt.




*Old and modified orifice flow meters*

## Safety Related Commissioning Tests

Towards validating the safety analyses for incident conditions, a series of tests was conducted in 1994 for getting safety clearance for sustained operation. The tests were:

i. Manual LOR

ii. Manual SCRAM

iii. Main Boiler Feed Pump Trip

iv. One secondary sodium pump trip

v. One primary sodium pump trip

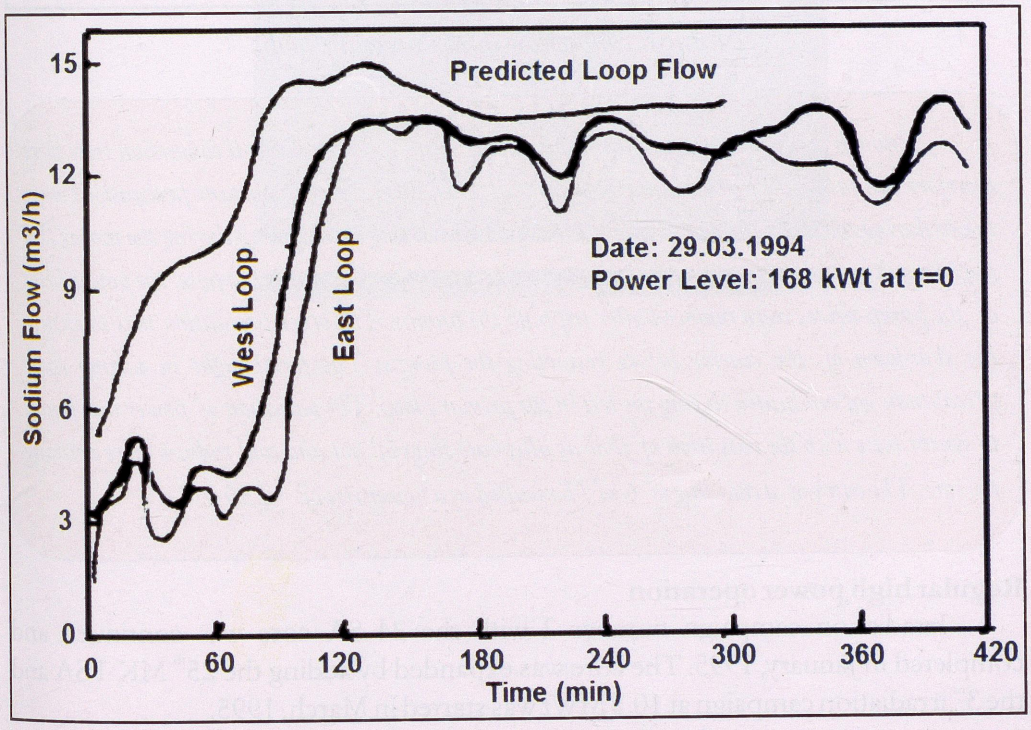
vi. Off-site power supply failure

vii. Natural convection in secondary loops by tripping both secondary pumps

viii. Natural convection in primary loops by tripping both primary pumps

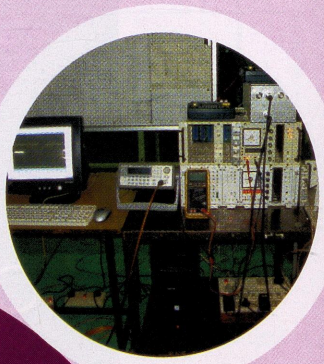
The first six tests were to ensure that the intended safety actions (tripping of reactor) take place. The temperature evolution of various components during each simulated incident was logged.

The natural convection tests were meant to demonstrate the ability to remove the decay heat and keep the reactor core temperature well within limits during station black-out conditions. Though both the tests confirmed the onset of enough natural convection sodium flows to remove the decay heat, the actual evolution of flows was at variance with the predictions. In the secondary, the flows developed after a time lag, which was attributed to the capacitance effect of the surge tank & expansion tank that was not included in the modelling. In the primary, only one loop participated in the convection which was attributed to the time lag between the closing of the non-return valves of the pumps. The flow and temperature patterns indicated the entry of alternate slugs of hot and cold sodium into the core.




*Natural convection in secondary loops*









# Operational History since 1995

## Commissioning of TG & Synchronisation

FBTR turbine, a 16.4 MWe, single cylinder with 16 stages, non-reheat type, is designed for a temperature of 480°C and pressure of 125 kg/cm$^2$. All the fire protection measures stipulated by safety authority were complied with before undertaking rolling trials. Turbovisory instrumentation was calibrated. Alignment of rotor was checked and the maximum misalignment (axial & radial) was 0.4 mm. During first rolling of turbine in March 1996, when the Integral By- Pass valve of turbine stop valve was opened, the turbine started rolling by itself due to leak past the Combined Isolation cum Emergency Stop valve. After soaking the turbine at 750 rpm for 1 hour, TG speed was raised to 1680 rpm by opening IBP valve further. High vibrations were felt at this speed and hence, TG was brought to standstill by closing IBP valve. Later on, it was confirmed that these high vibrations were due to crossing of the critical speed of 1680 rpm. After rectifying the problem with CIES valve, turbine was rolled 14 times up to the synchronous speed of 3000 rpm for cleaning the system and also for conducting commissioning tests.

On May 20, 1996 when turbine speed was being raised, heavy sparking was observed from the exciter brushes at 2650 rpm. The brushes and brush holders were found damaged and replaced. After checking the healthiness of exciter, turbine was rolled to 3000 rpm on May 21, 1996 and over speed rings were tested by oil injection. One of the trip valves was not getting reset and turbine got tripped. During the prolonged operation at synchronous speed, the turbine exhaust hood temperature reached 102°C as against the limit of 150°C at full load as specified by the supplier. In order to limit the exhaust hood temperature to less than 100°C at low loads, water spray was introduced at the turbine exhaust from CEP discharge. During initial phases of rolling, on a couple of occasions, reactor as well as turbine tripped due to tripping of CEP/DLP. This was attributed to level fluctuation in their suction capacities and CEP cavitation. Modifications were carried out to avoid this by increasing the size of CEP suction lines from MC & DC and restricting the vacuum in DC to -0.6 Kg/cm$^2$.

There was a minor oil fire between HP casing and Front pedestal. Turbine was tripped manually and oil fire was extinguished. The oil leak was due to higher clearance (0.5 mm) between gland ring segments and TG shaft as against the requirement of 0.198 mm. New rings were procured as per correct specification and

installed. There was no oil leak thereafter. One smoke detector was installed near the front pedestal.

Dry checks were completed on alternator, exciter and controls in parallel. The turbine was continuously run at 3000 rpm for drying out the alternator. Nuclear steam suitable for continuous running and loading was available in July when the reactor power was raised to 11.5 MWt. TG was rolled and synchronised to the power grid at 2045 hours on July 11, 1997.

Since then, TG is always connected to the grid during all campaigns.

## The Evolution of Core

FBTR has successfully completed 15 irradiation campaigns and the 16$^{th}$ is nearing completion. The term 'Irradiation Campaign' in FBTR refers to the period of reactor operation with a specific core configuration.

The evolution of FBTR core has been based on the burn-up capability of the MK-I fuel, established through stage wise PIE. Initially, MK-I fuel was rated for a conservative burn-up of 25 GWd/t & LHR of 250 W/cm as it is a unique fuel of its kind. Based on out-of-pile tests, the allowable LHR was raised to 320 W/cm. It was planned to progressively phase out the MK-I core at the target burn-up of 25 GWd/t and replace it by MK-II fuel to eventually reach a 76 fuel SA core capable of 38.2 MWt.

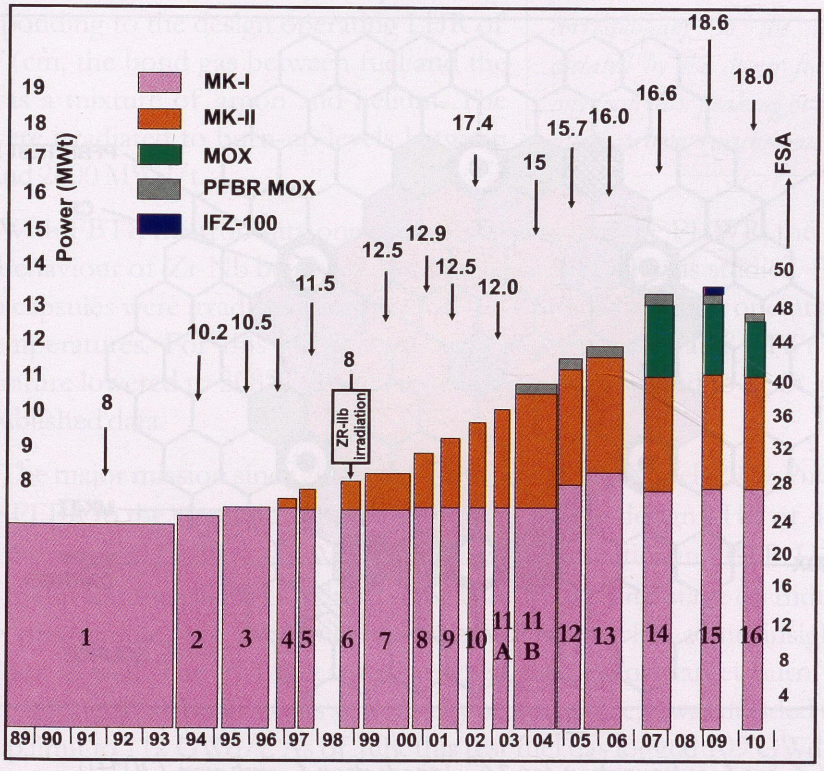
The central SA reached a burn-up of 25 GWd/t in May 1996 and was discharged for PIE. The core was enlarged by loading a fresh MK-I and MK-II SA and operation was continued pending the results of PIE. The PIE of SA at 25 GWd/t revealed the fuel burn-up capability up to 64.4 GWd/t. PIE was carried out at 50 GWd/t in 2000, based on which the burn-up was enhanced to 100 GWd/t. PIE was again carried out at 101.5 GWd/t in 2003, based on which the burn-up limit could be enhanced to 155 GWd/t. Up to 2003 (11$^{th}$ campaign) reactor operation was continued by progressively adding MK-II SA to compensate for the reactivity loss due to burn-up and discharge of the MK-I SA for PIE. With a desire to keep the fissile inventory at a minimum level till the closing of the carbide fuel cycle is demonstrated on a regular basis and in view of difficulties in making the large number of MK-II SA required to be loaded in the outer rings for reactivity compensation, the induction of MK-II fuel was suspended in 2003. Totally 13 MK-II SA had been loaded till 2003. Subsequently, the high burn-up MK-I SA are being replaced only by fresh MK-I SA. The PFBR MOX test fuel irradiation was initiated in the 11-B irradiation campaign in July 2003.

In order to validate the fuel fabrication route of the PFBR MOX fuel, eight high Pu MOX SA (44% $PuO_2$) were loaded in the core at the end of 2006. The core

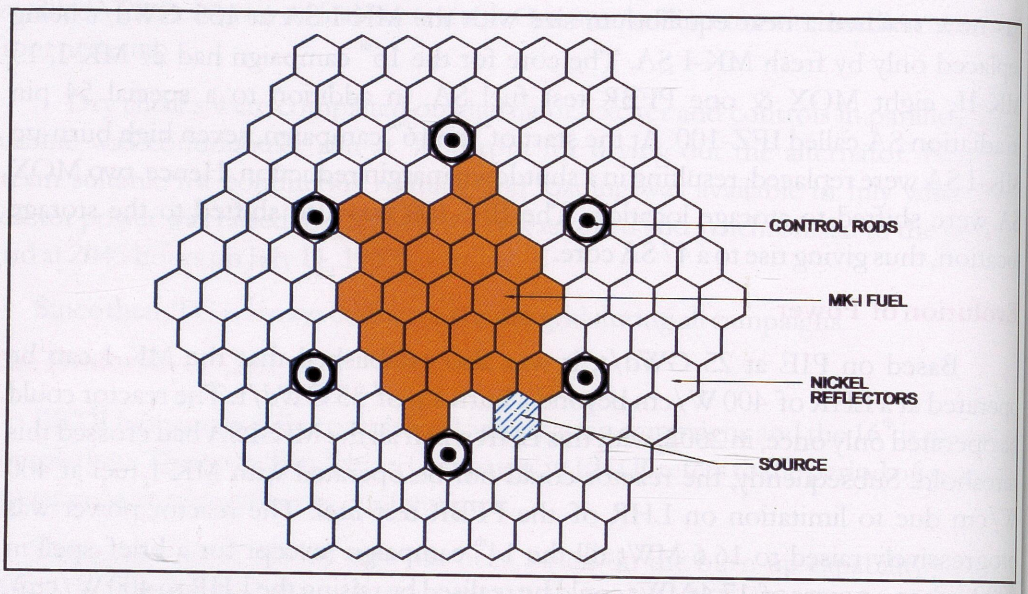
has now reached a near equilibrium size with the MK-I SA at 155 GWd/t being replaced only by fresh MK-I SA. The core for the 15th campaign had 27 MK-I, 13 MK-II, eight MOX & one PFBR test fuel SA, in addition to a special 54 pin irradiation SA called IFZ-100. At the start of the 16th campaign, seven high burn-up MK-I SA were replaced, resulting in a shutdown margin reduction. Hence, two MOX SA were shifted to storage location. The IFZ-100 was also shifted to the storage location, thus giving rise to a 47 SA core.

## Evolution of Power

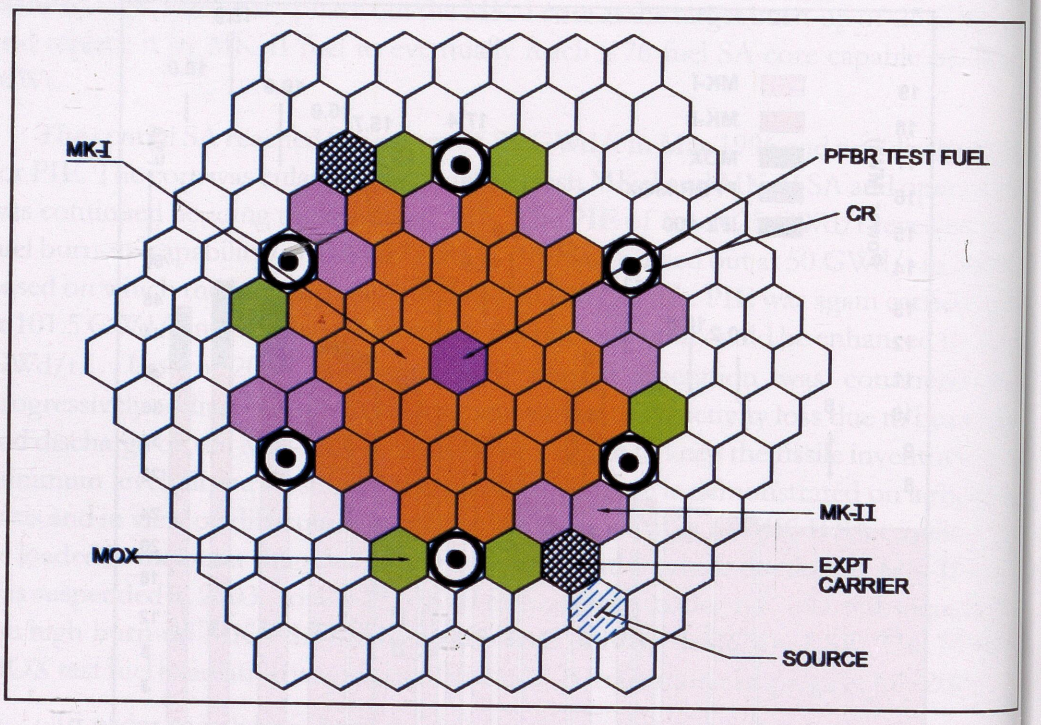
Based on PIE at 25 GWd/t, it was also established that the MK-I can be operated at a LHR of 400 W/cm beyond a burn-up of 35 GWd/t. The reactor could be operated only once, in 2002, with this LHR when all the MK-I SA had crossed this threshold. Subsequently, the reactor could not be operated with MK-I fuel at 400 W/cm due to limitation on LHR of the PFBR test fuel. The reactor power was progressively raised to 16.6 MWt till the 14th campaign (except for a brief spell in 2002, when a power of 17.4 MWt could be realised by raising the LHR to 400 W/cm). During the 15th campaign, the reactor power could be raised to 18.6 MWt and the TG power was 3.2 MWe.




*Evolution of Reactor Core and Power*




*Core for the first irradiation campaign (1985)*




*Core Configuration for 16th Irradiation Campaign (2010)*

## Evolution of Sodium temperatures

Till the end of the 13th campaign, the reactor outlet temperature remained around 430°C except during the 10th campaign when the temperature could be taken to 444°C due to operation at 17.4 MWt. Realising the limitations on further core expansion, it was decided to achieve operating temperatures close to the design temperatures by blanking three tubes out of seven in each SG module. This major modification occupied the whole of 2008. For the first time, in the 15th campaign, we were able to reach a sodium temperature of 482°C.

## Use as an irradiation facility

In addition to being a self-driven irradiation facility for the Pu rich monocarbide fuels, seven pins of MK-I & MK-II compositions were irradiated during 1994-96 to study their initial restructuring behaviour. They were loaded at the periphery of the core (5th ring) and were operating at a LHR of 168 W/cm. To get the fuel temperatures corresponding to the design operating LHR of 320 W/cm, the bond gas between fuel and the clad was a mixture of argon and helium. The pins were irradiated to burn-up levels between 1600 and 2600 MWd/t.

### *What is a Driver Fuel?*

*The term 'driver fuel' refers to the bulk of the fuel which contributes to the neutron flux for irradiation of other structural materials or fuels in an irradiation facility. Any other fuel pin or fuel subassembly loaded for test irradiation will operate at the Linear Heat Rating corresponding to the flux level dictated by the driver fuel without any local flux peaking or depression due to its own enrichment.*

With FBTR neutron flux one order higher than in the PHWR, the irradiation creep behaviour of Zr-Nb being used in the Indian PHWR was studied. Pressurised Zr-Nb capsules were irradiated during 1998-2000 simulating their operating stresses and temperatures. For this purpose, reactor was operated at 8 MWt with inlet temperature lowered to 308°C. The creep behaviour was found to be in accordance with published data.

The major mission since 2003 is to irradiate the MOX fuel (29 % $PuO_2$) chosen for the PFBR to the target burn-up of 100 GWd/t at the design LHR of 450 W/cm. To get the required LHR of 450 W/cm with the flux limitation in FBTR, $U^{233}$ has been added to the test fuel. With $U^{233}$ slated to be used in the third stage of India's nuclear energy programme, the irradiation is expected to provide some insight into its behaviour as well. The PFBR test fuel reached the original target burn-up of 100 GWd/t in June 2010. Based on its satisfactory performance, it was decided to raise the burn-up limit to 112 GWd/t. As of July, this test fuel has logged 109 GWd/t without any failure

*By taking more than 1000 pins of the MK-I driver carbide fuel to burn-up values as high as 155 GWd/t without any failure and with the successful demonstration of its fuel cycle closure, India can rightfully claim to have mastered the technology of one of the most advanced fast reactor fuels. The stainless steel clad and wrapper of the carbide fuel have been taken to one of the highest levels of neutron irradiation reported in literature for stainless steels (~85 dpa), albeit at lower irradiation temperatures. With the plans to progressively raise the operating temperature to higher levels, it would be interesting to watch whether the same burn-up could be sustained, because the current burn-up limitations come not so much from the fuel as from the clad and wrapper. By switching over to the swelling resistant D9 alloys (used in PFBR fuel) for the MK-I fuel of FBTR, it will be possible to raise the carbide fuel burn-up to higher than 155 GWd/t even with higher operating temperatures.*

*The MOX fuel simulating the PFBR composition has also reached 109 GWd/t, validating the design and fabrication of the MOX fuel and portending success to the MOX fuel programme. It may be said that as a fuel irradiation facility, FBTR has fulfilled two thirds of its mission to our fast reactor programme. In the years to come, metallic fuels would be tested on a large scale.*

Irradiation of tensile samples, swelling-specimens and pressurised capsules of D-9 alloy in the fourth ring was initiated in the 14$^{th}$ campaign. These will be irradiated for three Effective Full Power Years to take the dpa to >70. SS 316 specimens were irradiated to ~3 dpa in this campaign and are under evaluation in connection with the residual life assessment of the Grid Plate. During the 15$^{th}$ irradiation campaign, one single pin simulating the PFBR fuel composition was irradiated at 400 W/cm for 14 days to study the initial restructuring of the fuel. Also, the feasibility of trial production of the isotope $Sr^{89}$ from Yttria was studied. A capsule containing Yttria pellets was irradiated in the central location for 70 days and discharged.

*$Sr^{89}$ is an isotope which has therapeutic applications for relieving bone pain in some types of cancer, and is currently marketed under the brand name "Metastron" in USA and Canada (and costs about USD 3000 per dose). $Sr^{89}$ can be produced in fast reactors by (n,p) reaction of $Y^{89}$. As the element produced is different from the starting element, chemical separation of $Sr^{89}$ in a pure state with very high specific activity is possible. Specific activities of about 19 kCi of $Sr^{89}$/g of Sr have been achieved in fast reactors vis-à-vis about 10 Ci/g of Sr in thermal reactors. The radio-chemistry laboratory of IGCAR has successfully separated $Sr^{89}$ from the Yttria pellets irradiated in FBTR; thus, opening up a new avenue for the societal applications of FBTR.*

## Physics and Engineering Tests

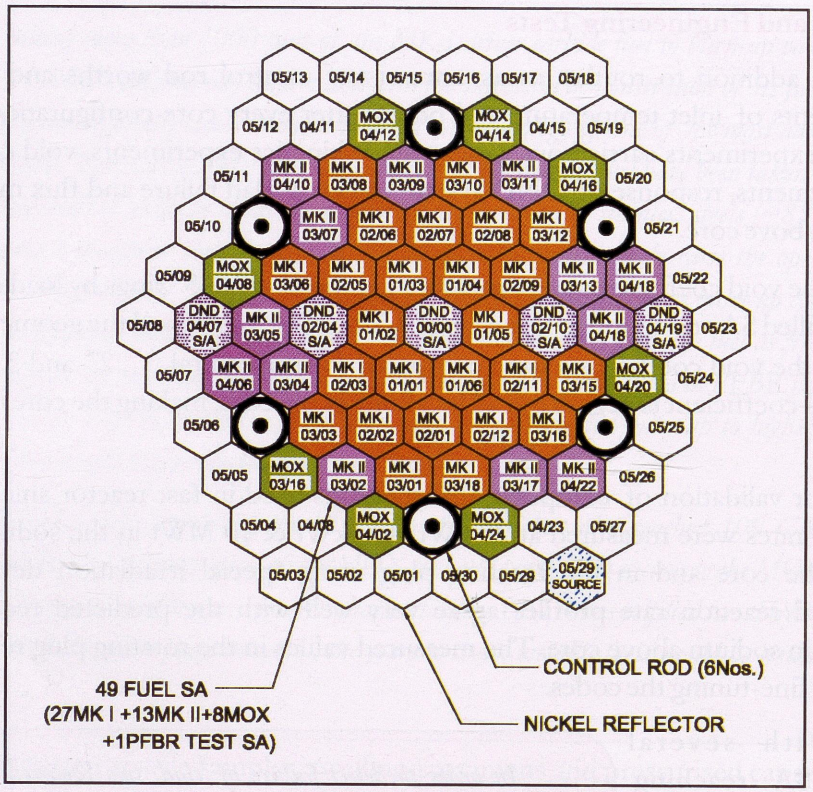
In addition to routine measurements of control rod worths and reactivity coefficients of inlet temperature and power after every core configuration change, physics experiments carried out were: reactor kinetics experiments, void coefficient measurements, response of DND system to detect clad failure and flux mapping in sodium above core.

The void coefficients were measured in the 3rd, 4th & 5th rings by loading special helium filled SA and found to be negative. Using these values and the geometry model of fuel, the void coefficients were computed for the central, 1st, 2nd and 3rd ring SA. The void coefficient is negative at all locations in the core, making the core inherently safe.

For validation of computational methods used in fast reactor shield design, reaction rates were measured at 25 kWt, 400 kWt & 10 MWt in the sodium region above the core and in the rotating plug using special irradiation devices. The measured reaction rate profiles agree very well with the predicted reaction rate profiles in sodium above core. The measured values in the rotating plug region were used for fine-tuning the codes.

With several MK-I SA reaching high burn-up levels, there was a pressing desire to demonstrate the response of DND system to detect a clad failure and its ability to identify the failure location. Experiments were conducted with a special SA with 19 perforated pins of Natural Uranium-Ni alloy. The SA was loaded at several locations and the contrast ratios of the signals were measured. The results confirmed the ability of the system to detect a clad failure. It will be possible to identify a failed fuel along the east-west line of the core centre. But at diagonal locations, the contrast ratios may not enable easy identification.

*In order to detect failure of clad, two Delayed Neutron Detectors are provided in primary sodium circuit, one in east loop and the other in west loop. Sodium is sampled before Intermediate Heat Exchanger inlet and presence of delayed neutrons are detected by boron counters of 8 Cps/nv sensitivity. In addition, there are two gas activity monitors provided; one in the Clad Rupture Detection circuit of cover gas and the other in the combined discharge of CRD & primary cover gas. These detectors detect the presence of gross-gamma activity in cover gas. The sampled gas is also sent to an experimental circuit where the age of the failed SA is estimated from the yield of gaseous fission products such as $Kr^{87}$ and $Xe^{135}$.*




*Location of Experimental Subassembly*

In July 2010, the gamma fields above the core were measured. The relative contributions from $Na^{24}$, $Na^{22}$ and reactor power (up to 400 kWt) have been assessed. These measurements will be used for fine-tuning the shielding and activation codes. Currently, experiments are in progress to study the reactivity effects of high burn-up SA due to bowing. The results will be used by the reactor physicists and core structural designers to validate and improve their codes.

PHEC is provided for decay heat removal during non-availability of primary loops due to leak in main pipe and its outer envelope outside A1 cell and also non-availability of both secondary loops due to common mode failure, such as sodium leak in SG casing. The heat removal capacity of the system under isothermal conditions was tested in October, 1991 and found to be adequate.

The heat removal capacity of BSC system and the evolution of concrete temperatures under simulated station black-out conditions were verified experimentally in July, 1992 & September 1997. Results of the test showed that the biological shield concrete and bulk structural concrete temperatures were well within

the limits. The tests also revealed a need to improve the cooling of concrete around the outlet pipes (appendix cooling). Based on this finding, before starting the 15th irradiation campaign at higher outlet temperatures, chilled water heat exchangers were interposed in the cooling air supply ducts.

## Testing of Devices

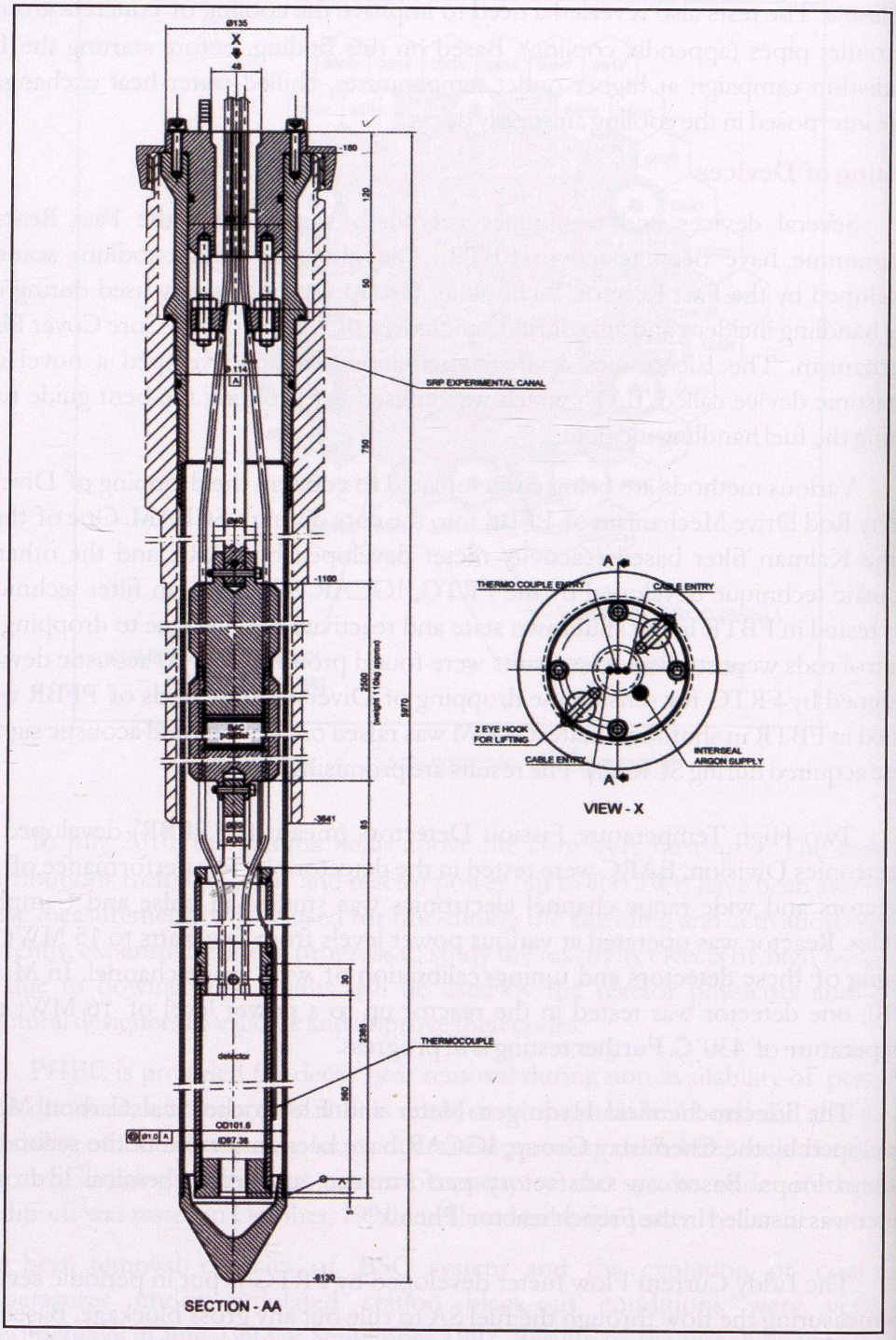
Several devices and techniques eventually required for the Fast Reactor Programme have been tested in FBTR. The ultrasonic under-sodium scanner developed by the Fast Reactor Technology Group was extensively used during the fuel handling incident and also during an incident of seizure of the Core Cover Plate Mechanism. The Electronics & Instrumentation Group developed a novel air-ultrasonic device called 'BAT', which was utilised for profiling the bent guide tube during the fuel handling incident.

Various methods are being contemplated to confirm the dropping of Diverse Safety Rod Drive Mechanism of PFBR into the core during a SCRAM. One of these is the Kalman filter based reactivity meter developed by BARC and the other is acoustic technique developed by the FRTG, IGCAR. The Kalman filter technique was tested in FBTR in the shutdown state and reactivity changes due to dropping of control rods were studied. The results were found promising. Two acoustic devices designed by FRTG for sensing the dropping of Diverse Safety Rods of PFBR were tested in FBTR in shutdown state. CRDM was raised one-by-one and acoustic signals were acquired during SCRAM. The results are promising.

Two High Temperature Fission Detectors (meant for PFBR) developed by Electronics Division, BARC were tested in the detector pit. The performance of the detectors and wide range channel electronics was studied in pulse and Campbell modes. Reactor was operated at various power levels from few watts to 15 MWt for testing of these detectors and tuning/calibration of wide range channel. In March 2010, one detector was tested in the reactor up to a power level of 16 MWt and temperature of 430°C. Further testing is in progress.

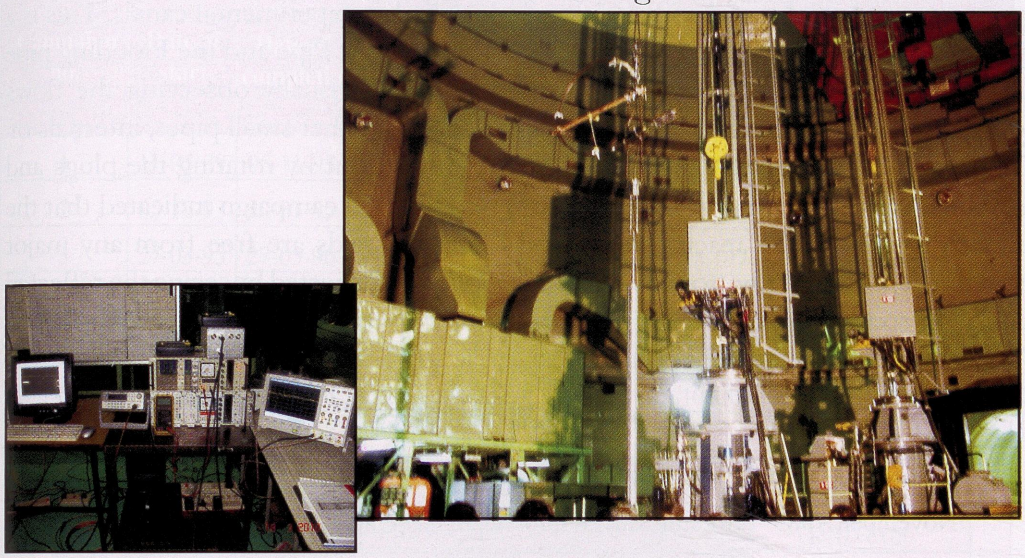
The Electrochemical Hydrogen Meter and Electrochemical Carbon Meter developed by the Chemistry Group, IGCAR have been in service in the secondary sodium loops. Based on satisfactory performance, an electrochemical hydrogen meter was installed in the French reactor 'Phenix'.

The Eddy Current Flow meter developed by FRTG is put in periodic service for measuring the flow through the fuel SA to rule out any gross blockage. Based on its satisfactory performance, one sensor was supplied by FRTG to French for end-of-life experiments in Phenix reactor.

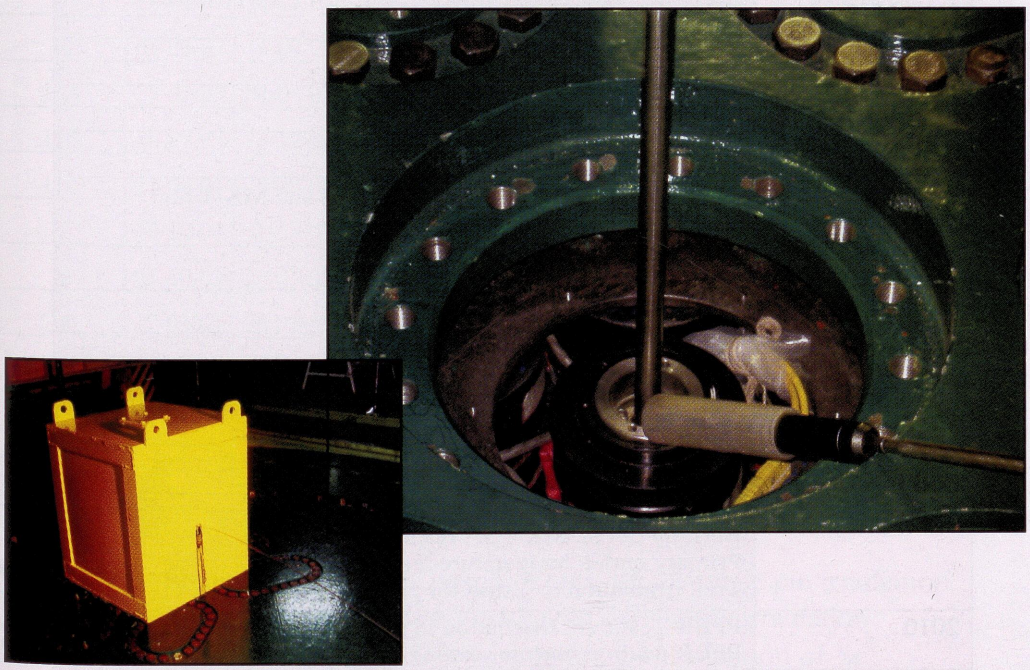



*PFBR High Temperature Fission Counter Testing Assembly*

*Installation of PFBR High Temperature Fission Counter in Reactor and Testing*



*Installation of gamma detector into the central canal plug and neutron shield plug on the reactor*



## Visual inspection of Reactor Vessel internals

The visual inspection of RV internals is carried out once in two years using periscope and a lighting device (projector) installed in experimental canals. This is a mandatory surveillance to ensure that the Siphon Break Pipe and the Flooding pipe inside the RV are functional. This is ascertained by visually observing the flows established through these pipes. Visual inspection of other small pipes, internals of RV, level probes, sodium surface etc. is also carried out by rotating the plugs and results were satisfactory. Visual inspection after the 15$^{th}$ campaign indicated that the inner surfaces of the reactor vessel and thermal shields are free from any major sodium deposits. The bottom of the rotating plugs is inspected by seeing the reflected image in sodium after stopping both the primary pumps to create a placid surface. The surfaces are found to be clean.

## Major Milestones

| | |
|---|---|
| October 18, 1985 | First criticality |
| November, 1989 | Sodium valved into Steam Generator |
| January, 1993 | Water valved into SG |
| December, 1993 | Power raised to 10.5MWt |
| 1994-1995 | Safety related Engineering Experiments |
| May, 1996 | MK-I SA burn-up reached 25 GWd/t |
| July, 1997 | TG Synchronised to grid |
| 1998-1999 | Zr-Nb irradiation for PHWR |
| April, 1999 | MK-I SA burn-up reached 50 GWd/t |
| March, 2002 | Power raised to 17.4 MWt |
| September, 2002 | MK-I SA burn-up reached 100 GWd/t and MK-1 fuel operated at design LHR 400W/cm |
| July, 2003 | Start of PFBR Test Fuel Irradiation |
| October, 2005 | MK-I SA burn-up reached 150 GWd/t |
| February, 2006 | MK-I SA burn-up reached 155 GWd/t |
| 2008 | Replacement of SG rupture disc and blanking of 3 tubes of SG |
| January, 2009 | DND Experiments at 10 MWt |
| February, 2009 | Power raised to 18.6 MWt / 3.3 MW (e)<br>Burn-up of one MK-I SA reached 165 GWd/t<br>Primary outlet temperature reached 482° C<br>Trial irradiation of Yttria for producing $Sr^{89}$ |
| 2010 | PFBR Test Fuel Irradiation in progress (109 GWd/t in July)<br>PFBR neutron detector testing |

**Campaign History**

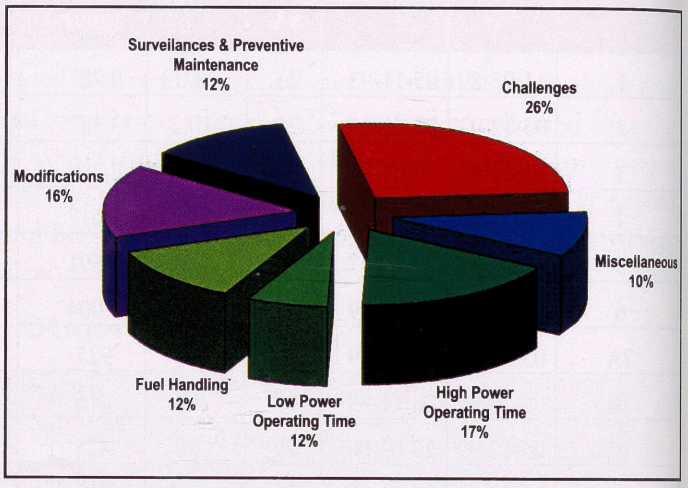
| Campaign no. | Date of start | Date of completion | No. of fuel SA | Max Power (MWt) | Total Operating time (hours) | Total thermal energy generated (MWh) | Purpose |
|---|---|---|---|---|---|---|---|
| 1 | 11-05-89 | 05-11-93 | 23 | 8.0 | 9,701 | 5,776 | Regular run |
| 2 | 05-12-93 | 20-01-95 | 24 | 10.2 | 3,387 | 12,330 | |
| 3 | 27-03-95 | 27-05-96 | 25 | 10.5 | 2,333 | 15,197 | |
| 4 | 03-11-96 | 30-05-97 | 26 | 10.5 | 1,617 | 12,312 | |
| 5 | 14-04-97 | 29-10-97 | 27 | 11.5 | 1,591 | 13,628 | |
| 6 | 25-06-98 | 11-01-99 | 28 | 8.0 | 1,904 | 10,130 | Zr-Nb irradiation |
| 7A | 05-02-99 | 17-07-99 | 29 | 8.0 | 925 | 5,778 | |
| 7B | 08-10-99 | 08-11-99 | 29 | 10.0 | 416 | 1,938 | Regular run |
| 7C | 06-12-99 | 23-02-00 | 29 | 1.8 | 425 | 96 | DND expt |
| 7D | 25-03-00 | 23-04-00 | 29 | 12.5 | 493 | 5,049 | Regular run |
| 7E | 19-05-00 | 10-06-00 | 29 | 0.0 | 1 | 0 | Void coefficient measurement |
| 8 | 01-09-00 | 04-02-01 | 31 | 12.9 | 2,579 | 28,674 | |
| 9 | 21-04-01 | 28-12-01 | 33 | 13.4 | 2,268 | 26,810 | Regular run |
| 10 | 01-03-02 | 11-09-02 | 35 | 17.4 | 1,875 | 24,630 | |
| 11A | 15-01-03 | 04-04-03 | 36 | 12.0 | 348 | 388 | Flux above core |
| 11B | 16-07-03 | 23-12-03 | 39 | 15.0 | 3,135 | 44,077 | |
| 11C | 25-08-04 | 28-08-04 | 39 | 15.0 | 82 | 691 | PFBR Fuel |
| 12 | 16-10-04 | 31-05-05 | 42 | 15.7 | 3,520 | 52,239 | Irradiation |
| 13 | 01-10-05 | 25-07-06 | 43 | 16.0 | 1,443 | 15,715 | |
| 14 | 17-02-07 | 26-01-08 | 49 | 16.6 | 2,541 | 38,524 | * |
| 15 | 06-12-08 | 27-05-09 | 50 | 18.6 | 2,000 | 30,305 | ** |
| 16 | 11-02-10 | in progress | 47 | 18.0 | 2,292 | 34,897 | PFBR Fuel Irradiation |
| | | **Total** | | | 44,876 | 3,79,184 | |

* PFBR test fuel irradiation, DND experiments, PFBR single pin irradiation, Grid plate flux measurement, Grid Plate material sample irradiation

** PFBR test fuel irradiation, DND experiment, trial production of $Sr^{89}$

## Performance Statistics

During the period from the first valving-in of water into SG to March, 2010, the high power operating time was 17%, low power operating time was 7% and fuel handling occupied 12% of the time. The reactor fuel handling being off-line, the cumulative availability was 36%. Technical challenges and the comebacks contributed to 26% of the downtime, plant modifications contributed to 16%, surveillances & maintenance activities to 12% and unscheduled outages due to reactor trip to 10%.

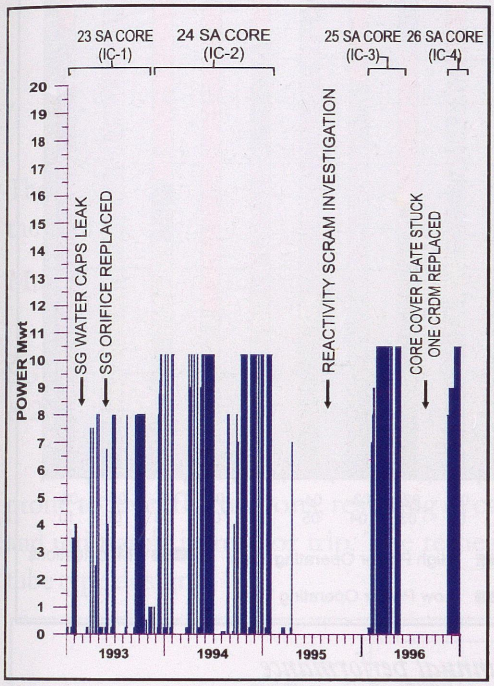


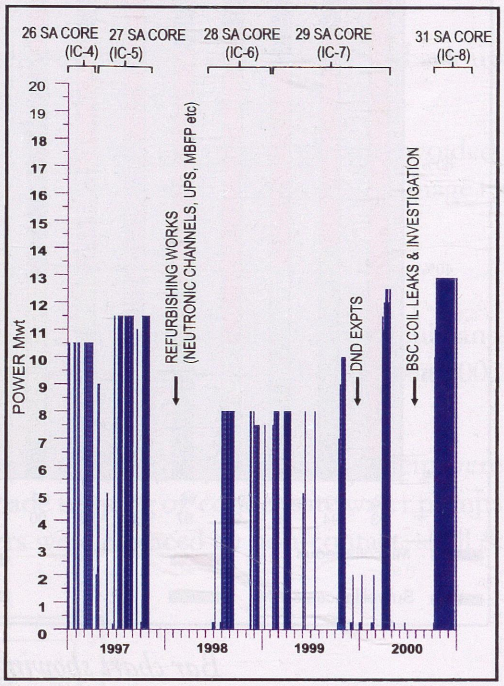

*Pie chart of operation statistics*

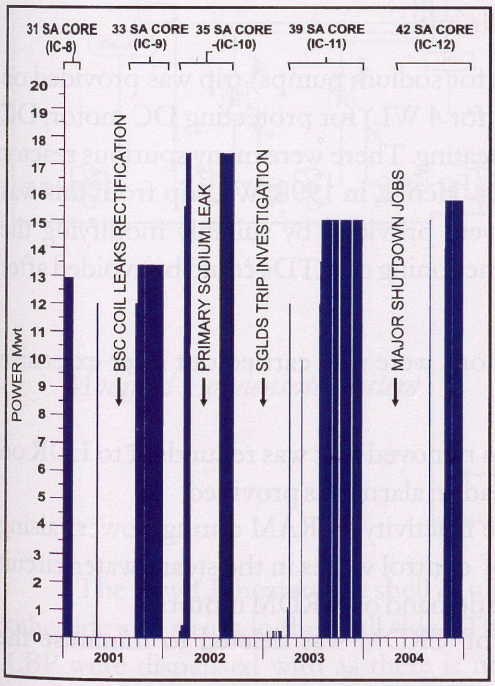
**Summary of Performance Statistics till July 10, 2010**

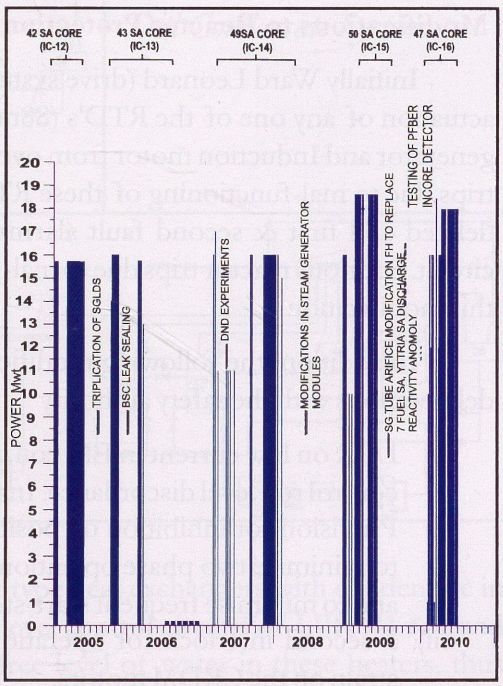
| Parameter | Since first Criticality |
|---|---|
| Power (MWt) | 18.6 |
| LHR (W/cm) | 400 |
| Bulk sodium temperature (°C) | 482 |
| Operating time (hours) | 44,876 |
| Thermal energy produced (GWh) | 379 |
| TG synchronisation time (hours) | 7,981 |
| Electrical energy generated (MU) | 16.48 |
| Longest operating campaign (days) | 72 |
| Cumulative operation of four Na pumps (hours) | 6,57,000 |
| SG operation (hours) | 27,816 |
| No. of LOR's / Controlled shutdowns | 281 |
| No. of SCRAM | 163 |

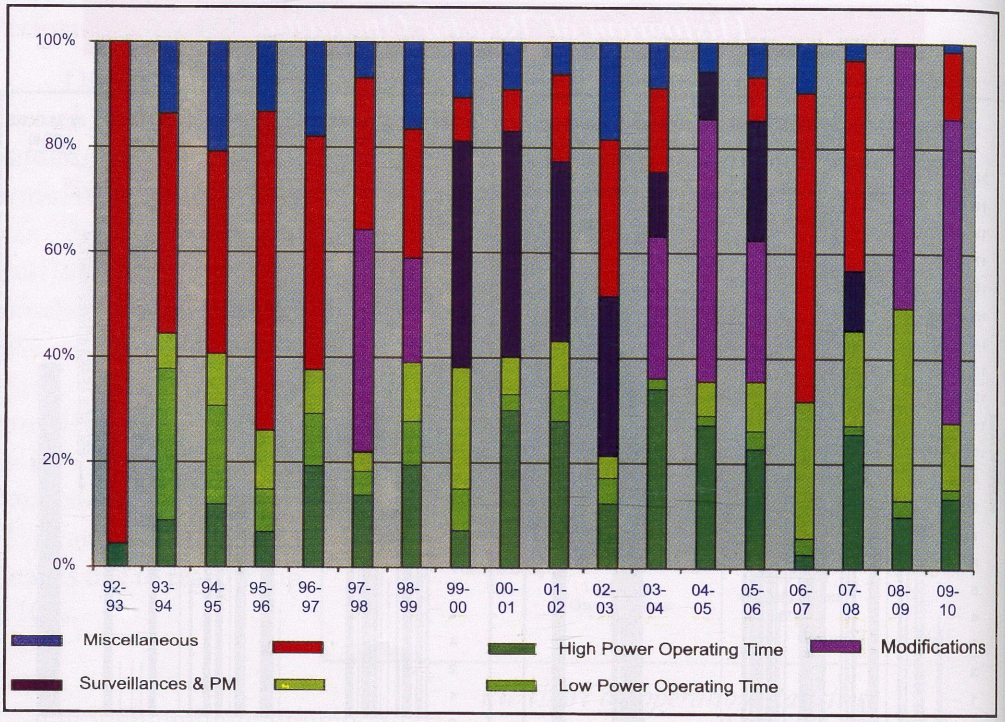
## Histogram of Reactor Operation






*Bar chart showing annual performance*

## Modifications to Reactor Protection Circuit

Initially Ward Leonard (drive system for sodium pumps) trip was provided on actuation of any one of the RTD's (88nos for 4 WL) for protecting DC motor, DC generator and Induction motor from overheating. There were many spurious reactor trips due to mal-functioning of these RTDs. Hence, in 1998, WL trip from this was deleted and first & second fault alarms were provided by suitably modifying the circuit. Spurious reactor trips due to mal-functioning of RTDs could be avoided after this modification.

In addition, the following modifications were also carried out after extensive deliberations with the safety authority

i) LOR on low current in EM coil was removed as it was redundant to LOR on control rod level discordance. Instead an alarm was provided.

ii) Provision for inhibition of positive reactivity SCRAM during power raising to minimise two phase operation of control valves in the steam water circuit and to minimise frequent start-stop demand on CRDM motors.

iii) 3 second interlock for operation of CRDM was deleted to minimise the strain on the CRDM motors.

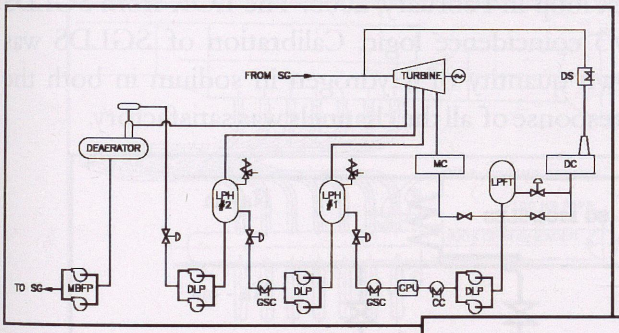
iv) All class-II LOR parameters were converted to class-I LOR parameters
v) LOR threshold on control rod level discordance was raised from 20 to 40 mm
vi) Log P SCRAM threshold was increased from 110 to 125% with a back-up SCRAM at 10% if the linear channels do not take over

After these modifications, many spurious reactor trips could be avoided. These also led to enhanced life of the CRDM and minimised the erosion damage in the control valves in steam water system.

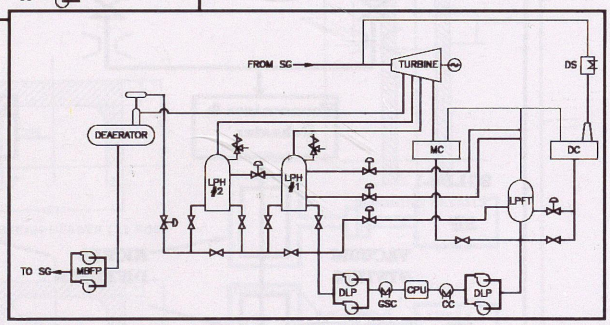
## Modifications to improve plant availability

Most of the unscheduled outages were from the steam-water circuit and SGLDS. Major modifications were carried out on both these systems between 2002 & 2004, which resulted in significant reduction in unwarranted reactor trips.

The contact type re-heaters, which were used in the steam water system were prone to level fluctuations, resulting in cascade tripping of condensate water pumps and ultimately in reactor trip. The re-heaters were replaced by non-contact, shell & tube type design.




*Old Condensate System*

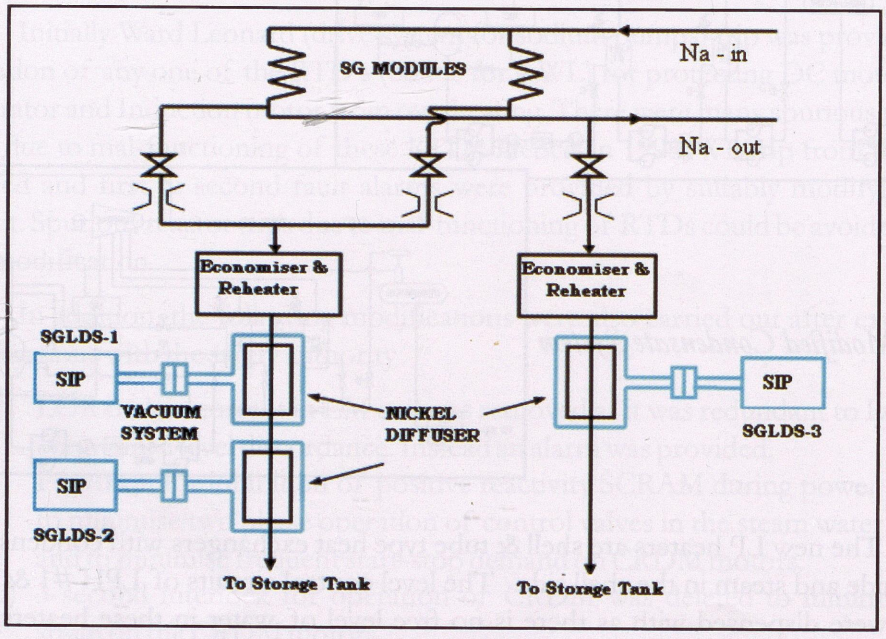



*Modified Condensate System*

The new LP heaters are shell & tube type heat exchangers with condensate in tube side and steam in the shell side. The level control circuits of LPH #1 & 2 and CBP were dispensed with as there is no free level of water in these heaters, thus simplifying the system. By the use of surface type heaters and necessary

modifications in the logic circuit of the pumps, the system was made more rugged and immune to process disturbances. Retrofitting of the new heaters in place of old ones had many technical challenges, which were overcome successfully.

Any leak of water/steam from the tube side to the sodium in the shell side of the SG is detected at the incipient stage itself by a sensitive SGLDS. The leak is sensed by an increase in the current of the Sputter Ion Pump, which maintains ultra high vacuum of $<10^{-8}$ torr in the shell side of a nickel diffuser. In 1997, the Mass Spectrometer signals were replaced by the SIP current signals. The SIP current is wired to initiate LOR based on absolute value of the current as well as its rate of rise. Each secondary loop was provided with one SGLDS. There was no provision to confirm the genuineness of a leak signal and the availability of the system was very much dependent on the single channel. To improve the availability of the plant and to avoid spurious actuation of leak signal, the system was duplicated in the first phase by adding one more nickel diffuser-vacuum system in each loop in series with the existing SGLDS. A third system with its own re-heater & economiser, nickel diffuser and SIP was then added, taking sodium sample from the common sodium outlet header. The retrofitting and commissioning was completed for the east loop in September, 2005 and for the west loop in February, 2006. The LOR from SGLDS was modified to incorporate 2/3 coincidence logic. Calibration of SGLDS was carried out by injection of known quantity of Hydrogen in sodium in both the secondary sodium systems and response of all the channels was satisfactory.

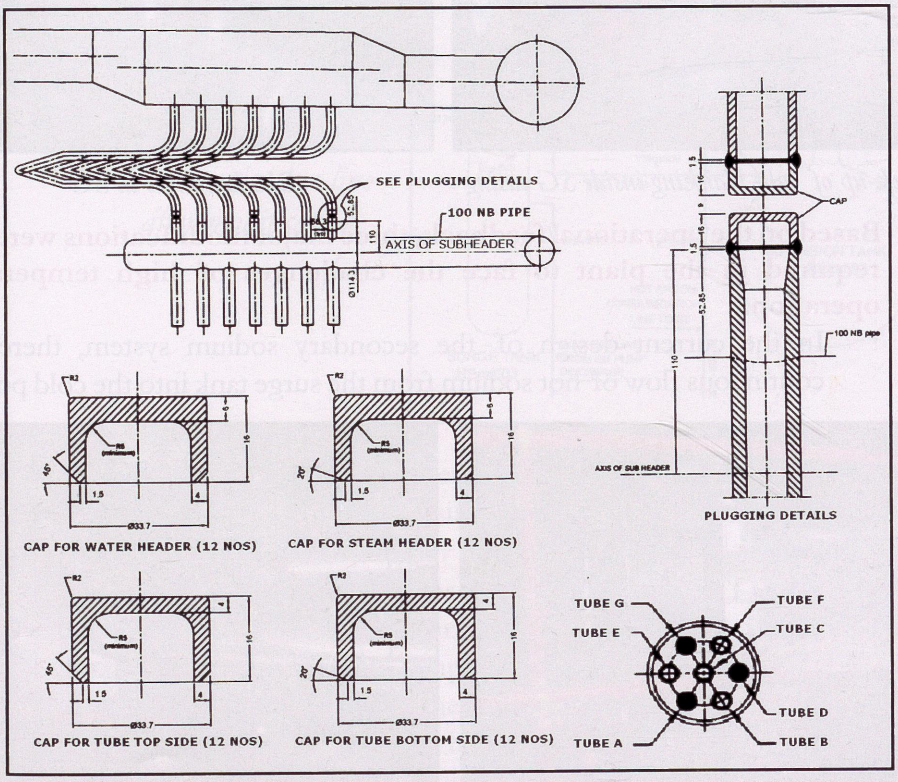



*Triplicated SGLDS*

The initiative for triplicating the Steam Generator Leak Detection System came from a suggestion from Dr.Anil Kakodkar, former Chairman of AEC, when he was a member of the Safety Review Committee for Operating Plants (SARCOP). The modifications to the steam water system and SGLDS were so effective that they resulted in the longest continuous run of the reactor for 72 days during the 14th irradiation campaign.

## Modifications for raising the sodium temperatures

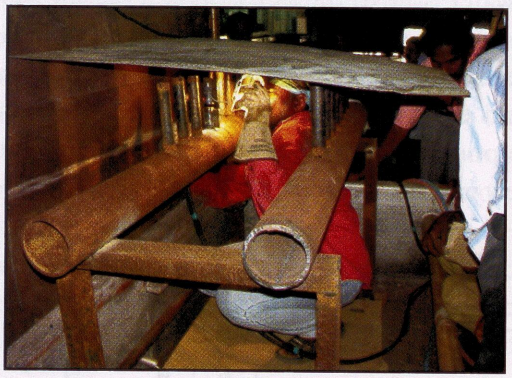
In order to conduct meaningful irradiation of fast reactor materials, it is required to raise the system temperature closer to the design value. To achieve this with the current constraints on core size and reactor power, three options were studied viz., operation with reduced feed water flow to the SG, isolation of one SG module from each secondary loop and blanking of three water tubes in each SG module. The last option was considered best from the decay heat removal capability, ease of carrying out the modification and ease of restoration when the core is enlarged to 40 MWt. With this, it will be possible to realise reactor inlet & outlet temperatures of 380°C & 490°C respectively with the present small core.



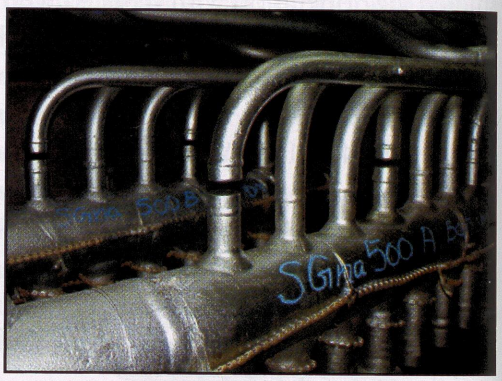

*Details of blanking in SG module*

## Operational History since 1995

A full fledged mock-up set up simulating the site constraints was erected inside the SG casing. For confirming correspondence of the cut tube at top and bottom, water was poured from top end and exit from the bottom tube end was checked. Discrepancies were noted between the top and bottom orientation in four tubes in the west loop modules with respect to the drawing. The correspondence was then established by pouring chilled water into the open end of the cut tubes at the top and feeling the chillness of the tubes at the bottom. This was further confirmed by the gamma radiography and then the tubes were blanked on both ends. Post-Weld Heat Treatment, Radiography Test and Helium Leak Test on all the weld joints were carried out. Finally, hydro testing of SG at 135 bar was done by operating the MBFP. Thermocouples were provided in two of the SG module shells to monitor the shell temperature distribution.



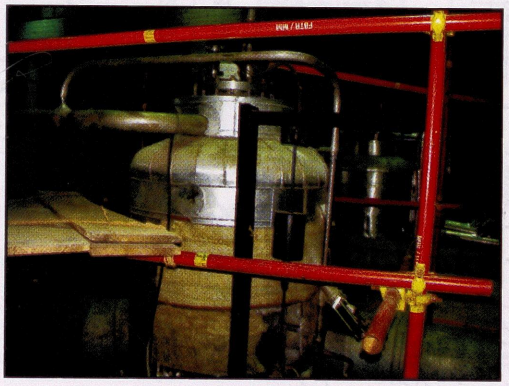
*Mock-up of tube blanking inside SG casing*



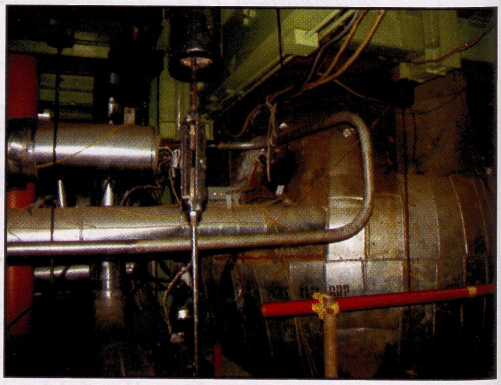
*Blanked tubes in SG*

**Based on the operational feedback, three major modifications were also required in the plant to face the challenges of high temperature operation:**

* In the current design of the secondary sodium system, there is a continuous flow of hot sodium from the surge tank into the cold pool of

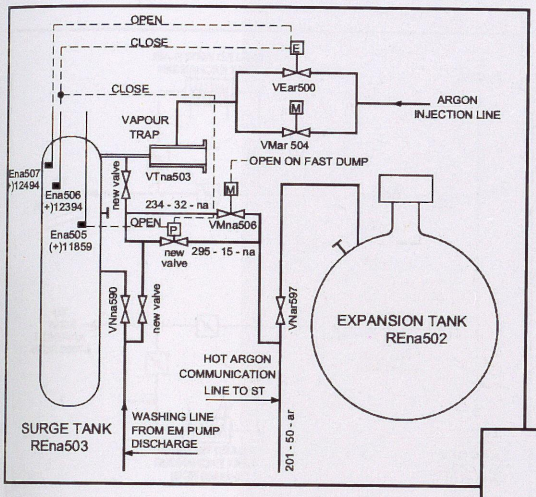


*Surge Tank*

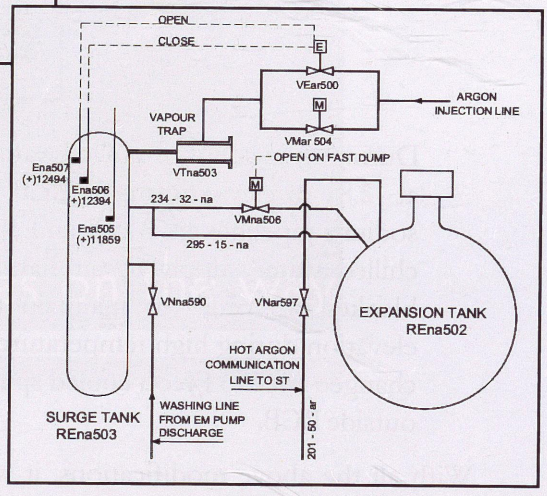


*Expansion Tank*

the expansion tank through a communication line. In Phenix reactor, cracks were seen in the expansion tank due to high thermal striping stresses at the location where the hot sodium mixes with the cold pool. This circuit in FBTR secondary loops was hence modified by converting the sodium line into a hot argon line.

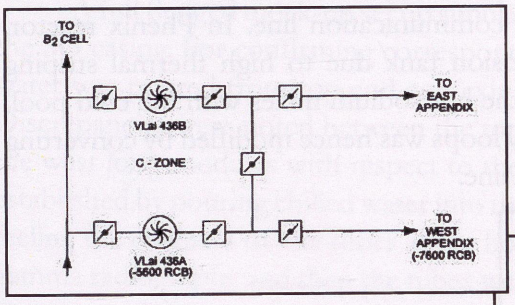



*Communication line before modification*



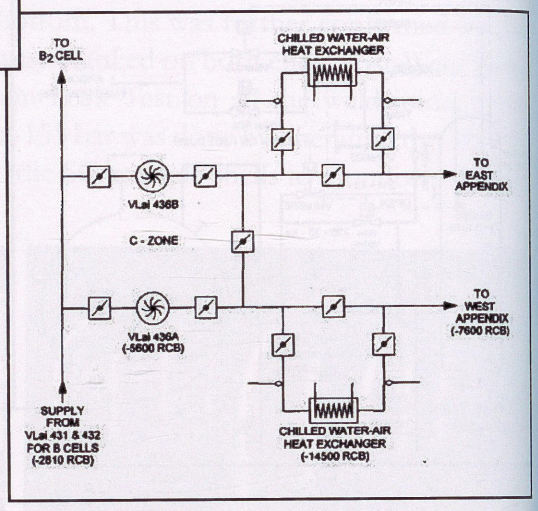

*Communication line after modification*

* In order to maintain the concrete temperatures surrounding the reactor outlet pipes within the permissible limits, chilled water heat exchangers were provided at the cooling air supply side. During the start of the 15th campaign, this modification was found be inadequate. On investigation, large leaks were noticed in the air ducts to the appendices in B-cells. These were arrested and concrete temperatures were found to be within limits thereafter. Subsequently, further improvements were carried out by replacing both the blowers with higher capacity ones. Each blower is now capable of cooling both the east and west appendices. One will be in service and the other standby.




*Air cooling circuit of concrete before modification*

*Air cooling circuit of concrete after modification*




* During an incident of chilled water leak from an air handling unit located at -2.81 m elevation inside RCB during initial stages, water fell on the sodium pipelines in B-cell and flooded the basement of RCB. Hence, chilled water supply to air handling units located above B-cells was blanked. In order to maintain the ambient temperature at -2.81 m elevation during high temperature operation, the air handling units were changed over to Freon cooled split-units with their outdoor units located outside RCB.

With all the above modifications, it was possible to raise the reactor outlet sodium temperature to 482°C during the 15$^{th}$ irradiation campaign without any problem.

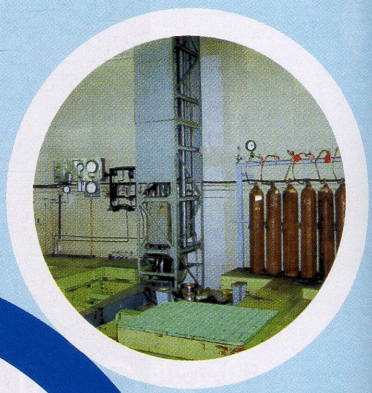
## Confidence from the 16$^{th}$ irradiation campaign

Before the start of the 16$^{th}$ campaign all major technical issues affecting plant availability and capability for high temperature operation have been addressed. The smooth operation of reactor in this campaign has demonstrated that all the modifications are effective. The reactor is now fully poised for operation at the design temperatures with higher availability and is capable of taking up irradiation of fuels & structural materials at higher temperatures

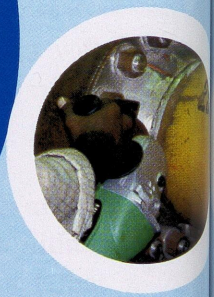
Chapter - 6

# Challenges on the way









# Challenges on the way

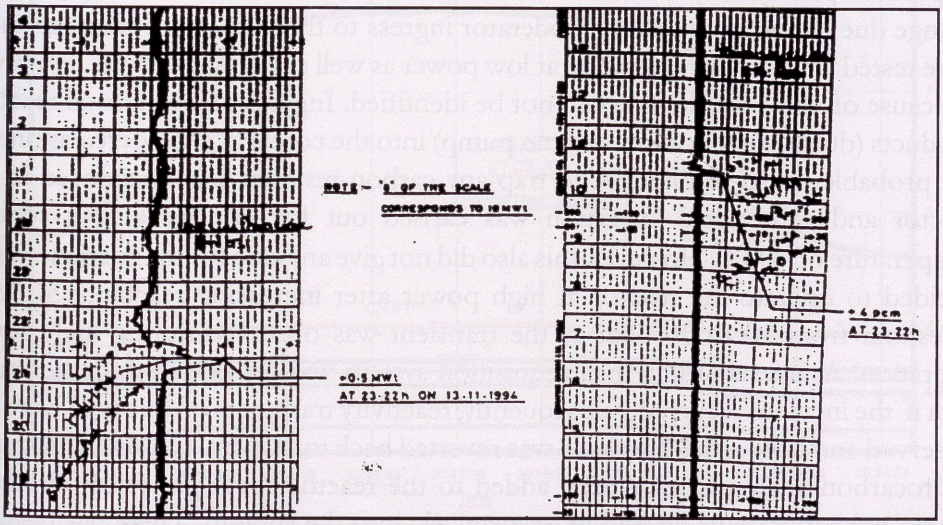
*The sharp edge of a razor is difficult to walk on. So is the path to perfection, the learned say. But perfection is attainable and shall be attained.*

***Kathopanishad***

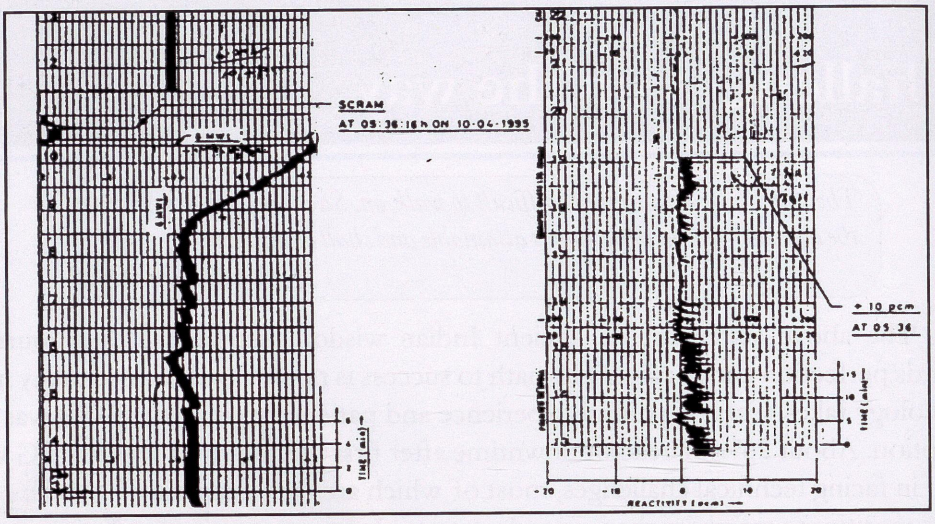
The above statement of ancient Indian wisdom summarises any journey towards perfection and success. The path to success is rarely a bed of roses. Any new technology takes years of painful experience and patience to master. FBTR was no exception. About 26% of reactor downtime after first valving-in of water in SG was spent in facing technical challenges, most of which are characteristic of a small core that is sensitive to reactivity changes and prone to sharp temperature gradients.

## Reactivity incidents

In November, 1994 while reactor was operating at 10.1 MWt, reactor power started increasing without any movement of control rods. Control rods were lowered to restore the power back to 10.1 MWt. However, the power continued to rise and reached 10.45 MWt within a minute. Control rods were again lowered by 7.6 mm (22 pcm) to bring back the power. The reactivity recorder registered a sudden increase of 4 pcm during the incident. No permanent gain in the reactivity was observed before and after the incident. This was confirmed by shutting down the reactor and re-measuring the shutdown margin.




*The power and reactivity recorded during the transient (13-11-1994)*




*The power and reactivity recorded during the transient (10-04-1995)*

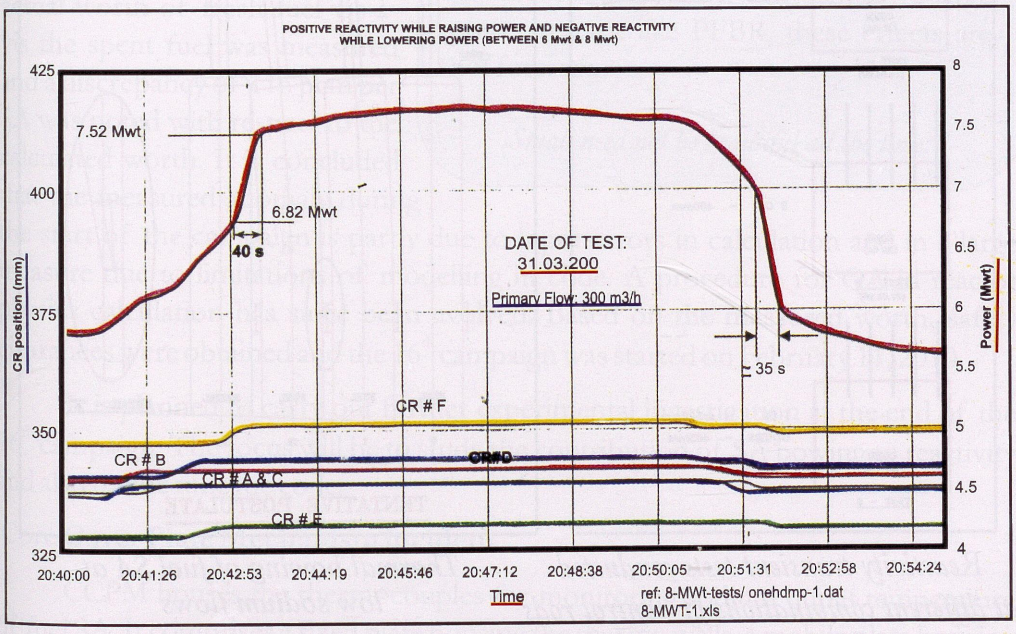
In April, 1995 when reactor power was at 7.1 MWt during the process of power raising to 10 MWt, there was a sharp increase in the power by 450 kWt in 7 seconds and reactor underwent SCRAM on high positive reactivity (Threshold of ±10pcm). The reactivity recorder also indicated a sudden increase of 10 pcm. A permanent gain of about 13.8 pcm was observed after this incident.

These incidents were analysed in detail. Totally 19 postulates were studied to find out the possible cause for the observed transients. They were related to change in process parameters, movement of control rod, movement of fuel, reactivity change due to sodium voiding, moderator ingress to the core, etc. Several of these were tested with reactor operating at low power as well as on high power. However, the cause of these incidents could not be identified. Ingress of sodium-oil reaction products (due to oil leak from sodium pump) into the core was considered as one of the probable cause. A filter SA to trap any carbon residues was introduced in the reactor and sub-critical operation was carried out for one week at a sodium temperature of 400°C. However, this also did not give any clue for the incident. It was decided to operate the reactor at high power after increasing reactivity SCRAM threshold from 10 to 30 pcm as the transient was observed during high power operation. A fast recording data acquisition system was developed to gather more data if the incident recurred. Subsequently, reactivity transients of this type were not observed and hence, the threshold was reverted back to 10 pcm. On-line nitrogen & hydrocarbon analysers have been added to the reactor cover gas reject circuit to monitor the ingress of air and oil respectively into the sodium. There has been no increase of nitrogen & hydrocarbon concentration beyond the background.

In 1998, when reactor was being operated at 8.0 MWt for the irradiation of Zirconium – Niobium alloy, a reactivity transient of repeatable nature was encountered. While raising the power, it was found to go up slowly from 7.1 MWt to 7.9 MWt without any movement of control rods and remained stable at 7.9 MWt. While lowering the power, it was found to come down by itself from 7.1 MWt to 6.3 MWt and stabilise.

**Following were the salient observations.**

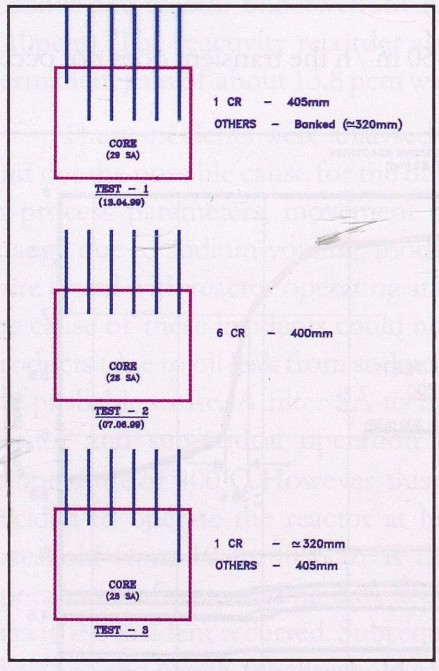
1. The transient was self limiting and the power remained stable after the transient.
2. The reactivity gained (~ 30 to 40 pcm) while raising power was lost while lowering the power. There was no permanent gain in reactivity. The power increase was around 700 to 800 kWt.
3. Tests at different Power/Flow ratios revealed that the onset of transient occurs at a specific value of mean core gradient temperature (m) of around 90°C. The transient can be observed at any power level, if the above m is reached.
4. The value of m at the onset of the reactivity transient increases with increase in primary sodium flow and the magnitude of the transient also comes down.
5. At a flow of equal to or more than 450 $m^3/h$ the transient does not occur irrespective of m value.



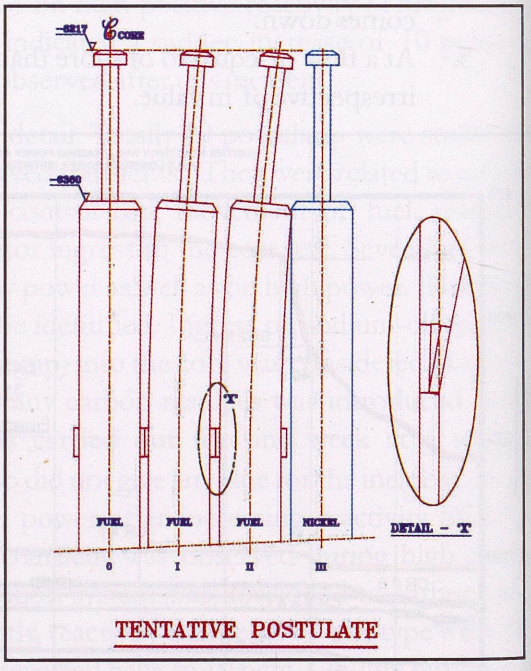

*Power variation and control rod position during the transient*

## Challenges on the way

The cause of the incident was investigated in detail. 35 tests were conducted. Many of the tests were with different combinations of control rod heights, to see whether thermal deformations of the control rods were the cause. From the experiments conducted, it was concluded that the transient is not caused by the control rods or the carriers containing Zr-Nb specimens. The phenomenon was suspected to be due to slight, thermally induced geometric changes in the fuel subassemblies (probably thermal bowing at low subassembly flows). It has come to light because the reactor was operated at low power and low flow rates for Zr-Nb irradiation. The reactor is normally operated at higher power (12 to 15 MWt) and correspondingly at higher flow rates (>450$m^3$/h) and hence, there is no transient seen. Tests were conducted in the presence of specialists from safety authority and it was proved that up to the trip threshold (110% of operating power) there is no second transient. With this, clearance was obtained from safety authority for continuing reactor operation for Zr-Nb irradiation and for the subsequent campaigns with primary flow higher than 450$m^3$/h. However, whenever core configuration and operating power is changed, it is mandatory to establish the stable regime of power operation. With the progressive expansion of the core, reactivity transients have become a thing of the past.




*Reactivity transient tests conducted at different combinations of control rods*




*Thermal bowing of fuel SA at low sodium flows*

## Reactivity anomaly at the start of the 16th campaign

After completion of FH campaign for replacing 7 high burn-up MK-I SA by fresh MK-I SA, reactor start-up for the 16th campaign was initiated on December 25, 2009. There was an anomaly of 811 pcm in the shutdown margin as against 150 pcm permitted as per Technical Specification after any core configuration change. The start up was aborted and to respect the shutdown margin limit, two MOX SA were shifted from the 4th ring to the storage locations. A Task Force verified the various inputs in reactor physics calculation and observed a few input errors. These were not found adequate to explain the anomaly. This was the first time that seven high burn-up SA were replaced without any intermediate measurements of fuel subassembly worth. The actual worth of fresh fuel vis-à-vis the spent fuel was measured and a discrepancy of 116 pcm per SA was noted with respect to the calculated worth. It is concluded that the measured anomaly during the start of the campaign is partly due to input errors in calculation and in a large measure due to limitations of modelling in code. A procedure for QA in reactor physics calculation has since been evolved. Based on the measured worth, safety clearances were obtained and the 16th campaign was started on February 11, 2010.

The various reactivity incidents seen in FBTR are typically linked with the small size of the core, with attendant sensitivity of core reactivity to small changes in various core physics and geometric parameters. The steep flux gradient will cause bowing of SA in the outer rings. The steep temperature gradients can be accompanied by thermal bowing. The effect of fission products is difficult to model, because, in addition to the poison effect due to absorption cross-sections, the scattering effects may also contribute to core reactivity. Finally, the small size of the core results in a compounded effect of all these factors. Nearly one year of plant availability was lost due to reactivity incidents. In a large power reactor like PFBR, these effects are not foreseen.

*Small need not be beautiful all the time.*

It is planned to carry out further experimental investigation at the end of the 16th campaign. The focus will be to study the contribution of SA bowing to reactivity and also to fine tune the code.

## Core Cover Plate Mechanism incident

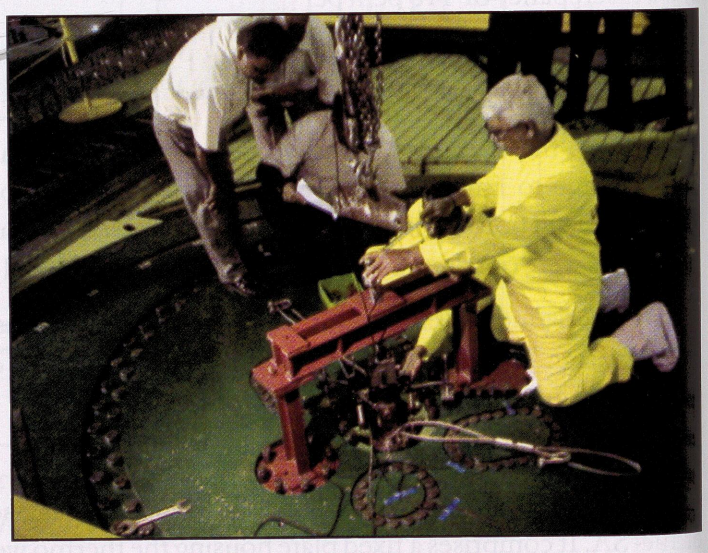
CCPM houses the thermocouples for monitoring sodium outlet temperature of fuel SA. It comprises a fixed plate housing the thermo wells, a mobile plate housing the sleeves, which guide the sodium jet from the fuel SA and a translation mechanism.

## Challenges on the way

The mobile plate is kept 15 mm above the heads of SA during reactor operation, 75 mm during FH and 0 mm (resting on the heads of SA) for accurate measurement of temperatures during experiments. The translation of the mobile plate is done manually through a ball-screw assembly. The fuel SA outlet temperature is continuously monitored to confirm non-plugging of any SA.

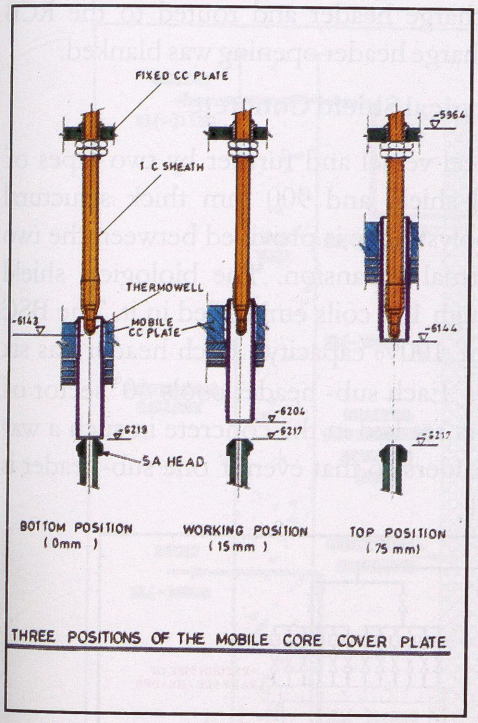
In July, 1995, when the CCPM was being lowered to 15 mm position after FH operation, it could not be lowered below 22 mm. After several investigations, it was decided to jack it down after confirming that there was no obstruction in the space below Core Cover Plate and the SA. The space between the mobile plate and SA heads was scanned over an angle of about 90° using under sodium ultrasonic scanner and it confirmed no obstruction. CCPM was mechanically jacked down with a force of 480 kg and got released. On repeated trials to check its free movement, the CCPM again got stuck at 22 mm. Jacking down with a force of 780 kg released the CCPM from 22 mm position and operation of CCPM was thereafter normal. Although the exact cause of the seizure could not be ascertained, it was felt that the seizure could probably be due to some obstructions in the inter-seal space of CCPM.

However, during subsequent FH campaign in July, 1996, the CCPM got stuck at 80 mm and could not be restored in spite of several efforts. Investigations carried out further revealed that the seizure was in an inaccessible location. It was decided to apply a jacking force up to 2 T (maximum allowed) below the point of suspected seizure. A step in the bore of the command tube was identified for this purpose. A collapsible metallic ram was developed and introduced into the command tube after removing the CCP. Jacking was tried in step loads of 500, 1000, 1500 & 2000 kg, monitoring the movement of ram and command tube by using dial gauges. But the CCPM could not be moved down. Pending confirmation of any non-interference between the thermowells in the fixed plate and sleeves in the mobile plate, no further attempts were made.

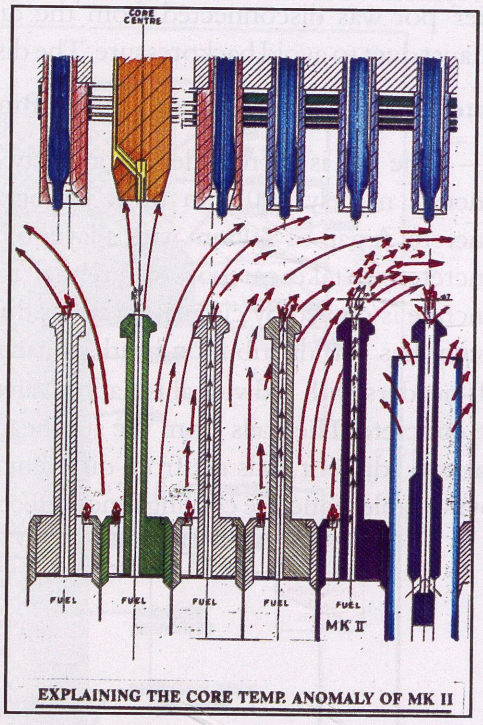


*Controlled Jacking down of CCPM*

Though in the current position CCPM does not pose any problem for normal operation or FH directly, the temperature measurements of the sodium from the fuel SA in the outer rings (especially in 3rd &4th rings) are not accurate. In the stuck position, the hot streams from the central SA glide over the cooler streams coming from the peripheral SA due to buoyancy effect and mask the true temperature readings. These temperatures are used for detecting the plugging of fuel SA and hence, an issue of safety concern. A lot of theoretical calculations and complex modelling had to be carried out to estimate the degree of inaccuracy in plugging detection. It was finally ascertained that sufficient margin was available to SCRAM the reactor safely in case of flow blockage in fuel SA.




*Thermocouple sleeves of CCPM at different positions*




*Schematic of mixing of sodium streams when CCPM at 80 mm*

All fresh SA are flow tested before loading into the reactor to rule out the possibility of loading a blocked SA. The probability of any extraneous material entering the core is considered to be remote. Nevertheless, it was also decided to supplement the plugging detection by periodic measurement of SA flow after every core change using the Eddy Current Flow meter developed by FRTG. This is currently in practice.

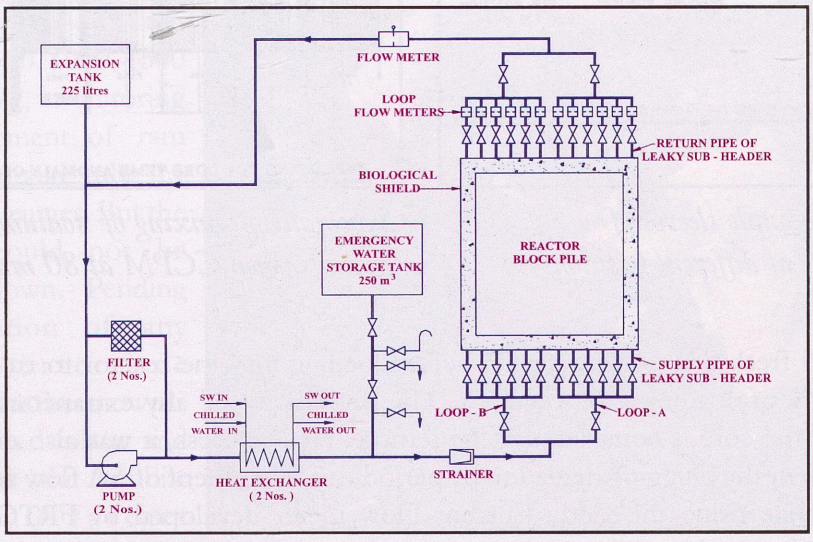
## Challenges on the way

### Mercury backing up from relief pot

In December 2005, when the primary sodium storage tank was pressurised for priming the fill & drain EM Pump for refilling the RV after its partial draining for internal inspection, it was noticed that the storage tank relief pot was relieving at a lower pressure. The investigation revealed that about 2 litres of Mercury backed in to the storage tank argon header which was in connection with the relief pot. The mercury was recovered and put back into the relief pot and relieving pressure got restored to initial value of 380 mb. The cause for backing of Mercury was due to passing of NRV in relief pot discharge line resulting in dynamic pressure of CRD compressor acting on the downstream side of relief pot. The discharge line from the relief pot was disconnected from the discharge header and routed to the RCB exhaust duct to avoid backpressure. The discharge header opening was blanked.

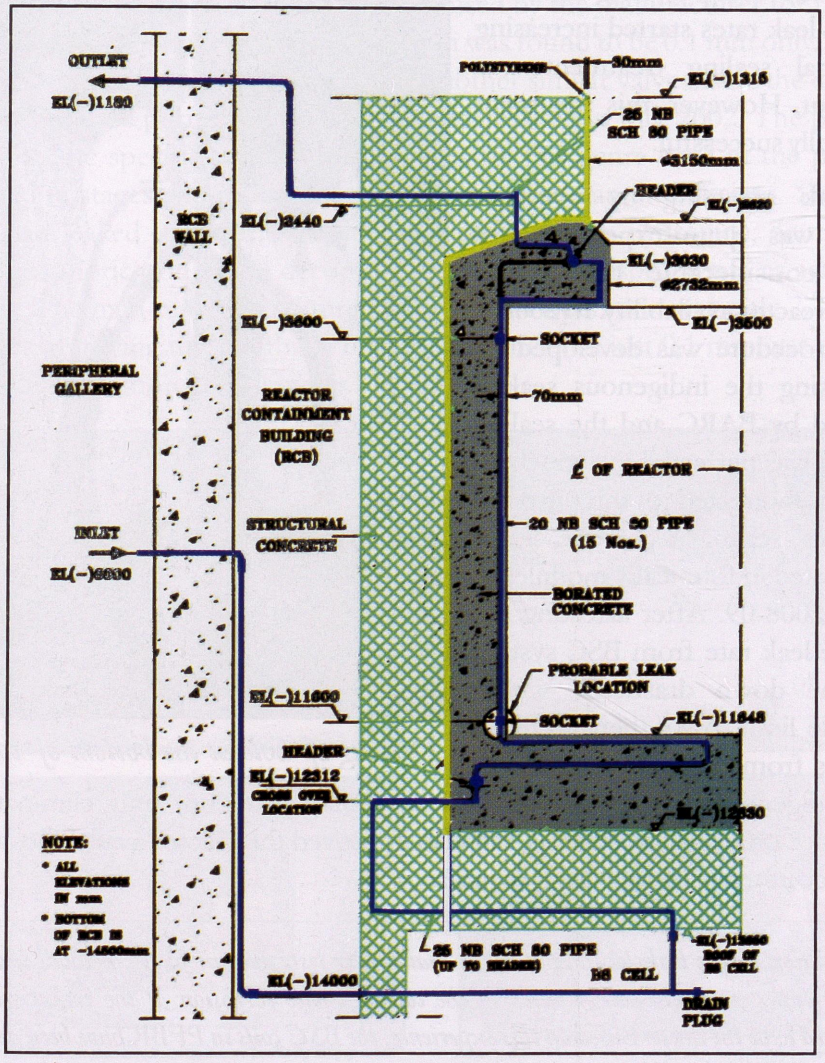
### Water leak from cooling coils inside Biological Shield Concrete

The RV is surrounded by a safety steel vessel and further by two types of concrete namely 600 mm thick biological shield and 900 mm thick structural concrete. A gap of 30 mm with a layer of polystyrene is provided between the two concretes to take care of differential thermal expansion. The biological shield concrete is cooled by circulating water through 180 coils embedded in it. The BSC system has two distribution headers; each of 100% capacity. Each header has six sub-headers with individual isolation valves. Each sub- header cools 60° sector of the concrete. The coils from the sub-headers are laid in the concrete in such a way that two adjacent coils are from different headers so that even if one sub-header is not available, concrete cooling is not affected.




*Schematic of the BSC loop*

In August, 2000, high inventory loss from BSC system was observed. Also, water seepage was observed from the structural concrete surrounding the manhole cover. When one of the sub-headers in header-A (A3), which cools 60° sector in the southwest portion of the concrete was isolated, the leak stopped. The leaky sub header was isolated. About 2 $m^3$ of water collected inside A1 cell (gap between steel vessel and biological concrete) was drained and reactor operation continued. There was no increase in biological shield concrete temperature as coils from header-B cooled the affected sector. In May, 2001, again inventory loss in BSC system and water seepage was observed. Reactor, which was operating at low power, was shutdown and investigation was carried out. This time the leak was from another coil




*Cross section of A1 cell with cooling coil*

(B5). During the investigation, water was found to be coming out of the isolated coil (A3) also. It was hence suspected that the leaky water from any of the coils could be stagnating in the interspace between the structural concrete and biological shield concrete and corroding the other coils. The leaks were arrested chemically by overseas agencies. To avoid stagnation of leaky water in the interspace, drain holes were drilled on the structural concrete, both at the bottom and on the sides.



*Drilling of hole at the bottom of concrete*

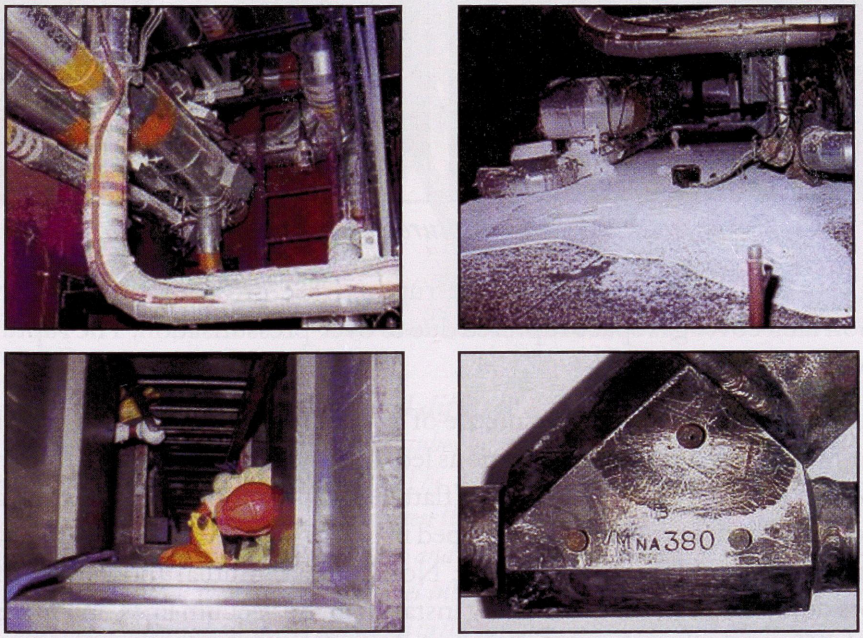
However after one year, leaks of small magnitude started appearing in some of the coils. During October, 2005, the leak rates started increasing and global sealing treatment was carried out. However, this treatment was not fully successful.

Leak arresting by overseas agencies was quite expensive and causing considerable time delay affecting reactor availability. In 2008, a sealing procedure was developed by FBTR using the indigenous sealant developed by BARC and the sealing technique was perfected in a mock up loop specially erected for this purpose. Using this sealing technique, leaks were arrested in four leaky modules in the year 2008-09. After arresting the leaks, the leak rate from BSC system has come down drastically within acceptable limit. Also, there were no new leaks from the coils which were sealed indigenously. These efforts by leak sealing team not only eliminated the dependence on overseas agencies, but also improved the reactor availability and our confidence in arresting such leaks in future.

*Identification of the leak location by water manometer tests and air purge methods indicated that the leaks are from socket welds in the coils. Crevice corrosion at the socket welds is considered to be the cause. Based on this experience, the BSC coils in PFBR have been designed and erected with butt welds.*

## Primary Sodium Leak

In April, 2002, there was an incident of leak of 75 kg of primary sodium from the purification circuit into the inerted purification cabin housing the circuit. The leaked sodium had frozen on the cabin floor and pipelines. The frozen sodium was cut manually through the man-hole at the top of the cabin using long tools. The sodium was then scooped out through another man-hole made at the lower side of the cabin for this purpose. The entire operation was carried out under inert atmosphere and from outside the cabin. Entry to the cabin was then made wearing air-masks to remove the residual sodium sticking to the pipelines. The cabin was then de-inerted and inspected. Leak was from the body of a 20 mm size bellows-sealed valve through one of the three blind holes used by the manufacturer for machining the valve body. The thickness at this location was found to be 0.1 mm only. The leaky valve was replaced and the blind holes in another similar valve inside the cabin were plug welded. The purification system was normalised in July, 2002. The problem is generic to the specific make and the valves of this genre used in the plant were inspected in stages and rectified wherever thickness was insufficient. The sodium which had leaked out in the incident was converted to hydroxide, neutralised by orthophosphoric acid in the decontamination pit and disposed off as active liquid effluent. The man-rem expenditure for this entire operation was only 2.25 person-mSv. It is gratifying to note that a material thickness of just 0.1 mm was enough to hold sodium for more than 15 years.




*Frozen sodium on the pipeline and floor of purification cabin*

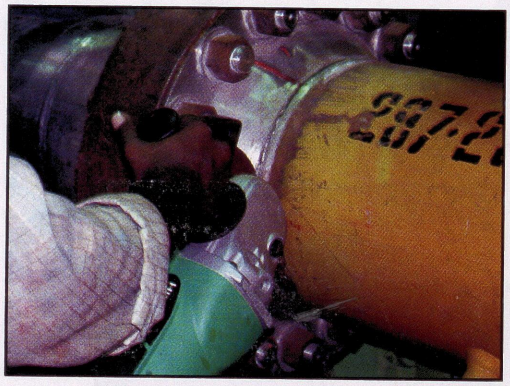
## Other Incidents in Sodium Systems

There have been four incidents of pin-hole leaks from the bellows of sodium valves. These were detected by the spark plug type leak detectors on the valves. The valves were replaced.



*Primary argon line choked with sodium & oxides*

There were a few occasions when the hot argon line connecting the primary sodium capacities with the primary sodium storage tank got choked with sodium and sodium oxides. Entry was made into the B-cells to identify the location. The choked lines were cut and replaced. The choking was due to improper laying of heaters on the nozzles of the storage tank. Heaters have now been provided on the tank shell surrounding the nozzle and the problem has not recurred.



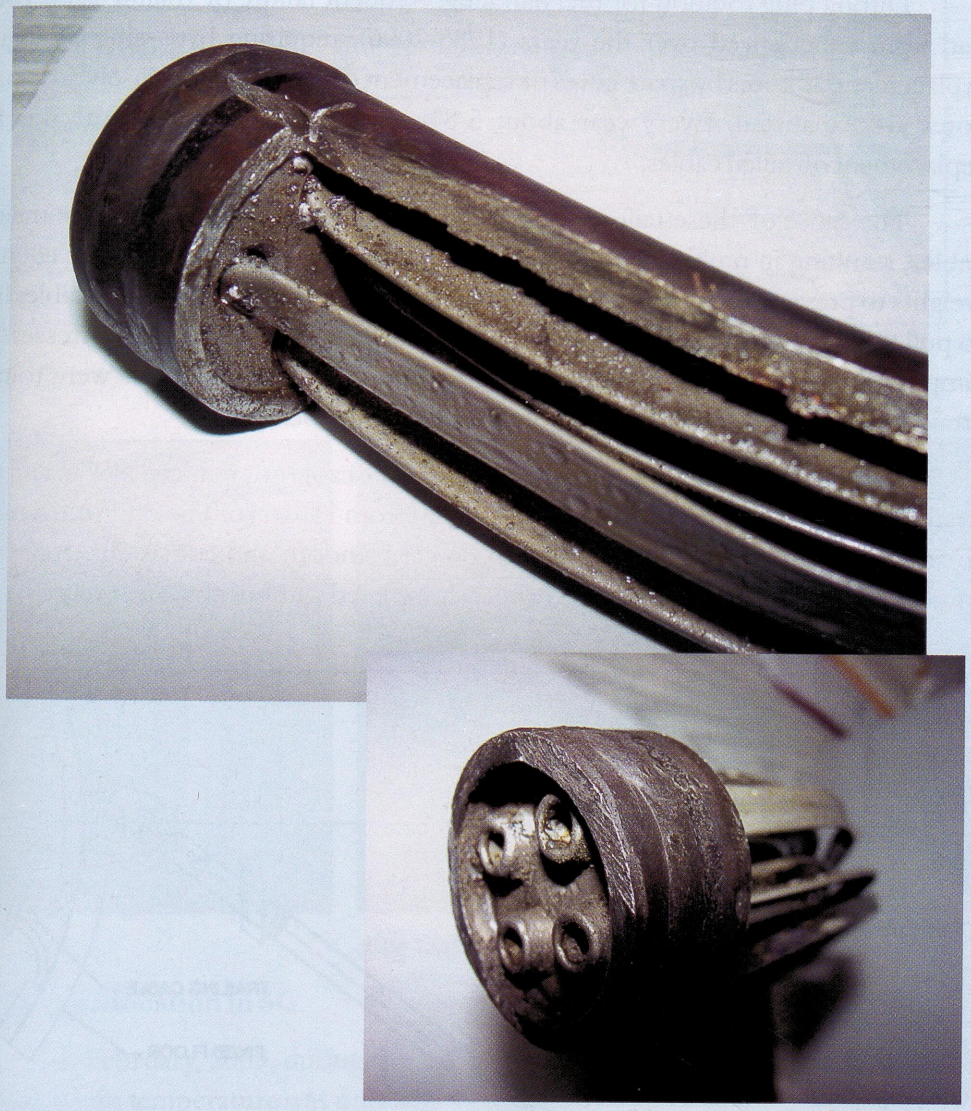


*Replacement of failed rupture disc in expansion tank*

There was an occasion when the rupture disc provided on the west loop expansion tank cover gas space ruptured due to over-pressurisation. The rupture disc was replaced.

There was, in 2006, a jig-saw puzzle of low flow through one of the triplicated loops of SGLDS. Extensive investigations led to the surprising finding that the nickel tubes of the diffuser had collapsed and flattened. Sodium had entered the vacuum line through a pin hole which had developed in one of the tube-tube sheet joints. The diffuser and vacuum line were replaced. Non-contact, Mutual Induction type leak detectors developed by FRTG are being installed in the vacuum lines downstream of all nickel diffuser shells to detect sodium leak from the nickel tubes.

*The flattened tubes of nickel diffuser*



**Nickel diffuser**

*The detector consists of four nickel tubes of 7.2 mm OD placed in a stainless steel shell of 42.4 mm OD & 35.9 mm ID. The assembly is bent by 90 degrees in order to accommodate the differential expansion between the nickel tubes and the outer (shell) envelope. The thickness of the nickel tubes is 0.3 mm and the total surface area of the nickel tubes is about 300 $cm^2$. Sodium flows inside the nickel tubes and $H_2$ diffuses through thin wall of the nickel tubes to the other side, which is connected to the vacuum system.*

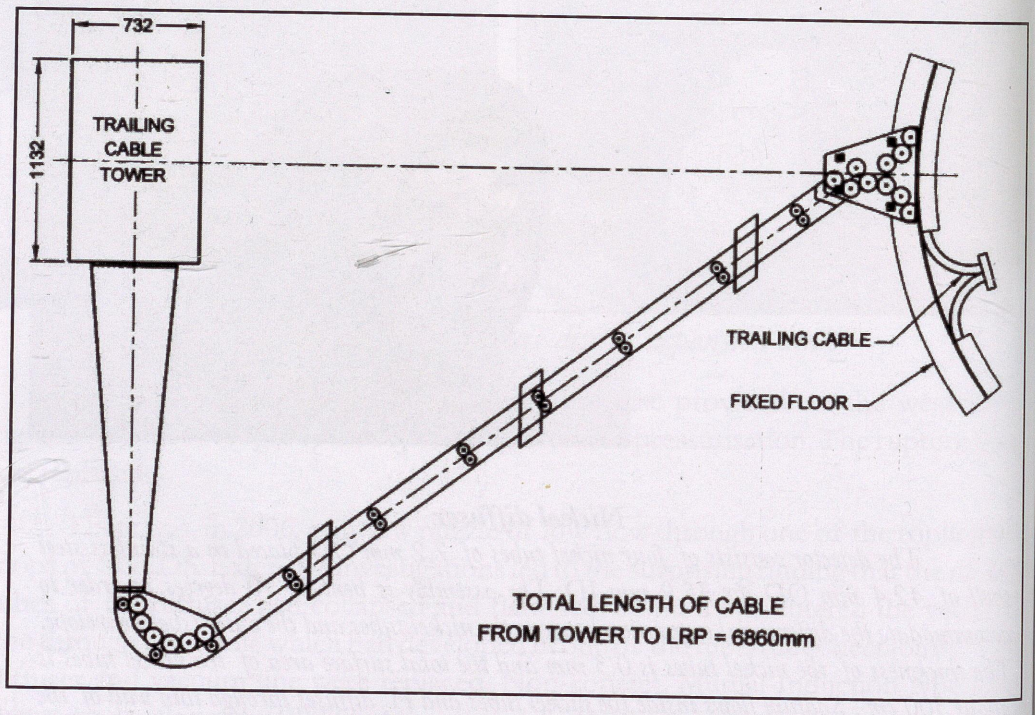
## Challenges on the way

### Failure of trailing cable cores

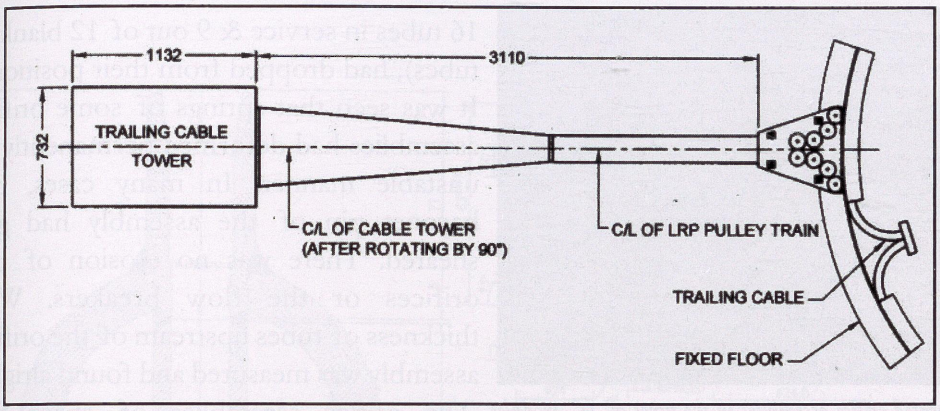
During plug rotation for fuel handling, frequent failure of trailing cable cores had been experienced over the years (1985-2006), requiring bypassing of logics, replacement of cores by spare cores or replacement of the cables themselves if spare cores are exhausted. Every year about 5-8% of the reactor down time was for replacement of failed cables.

The cause of these failures was found to be the tortuous path of the trailing cables, resulting in multiple bends and large number of pulleys with heavy counter weights to prevent slackening of the cables. During the winding cycles, the cables had to pull the counter-weights through this tortuous path, resulting in large stresses. To simplify the layout, the cable tower was rotated by 90° and the cables were routed straight from LRP to cable tower eliminating two sharp bends.

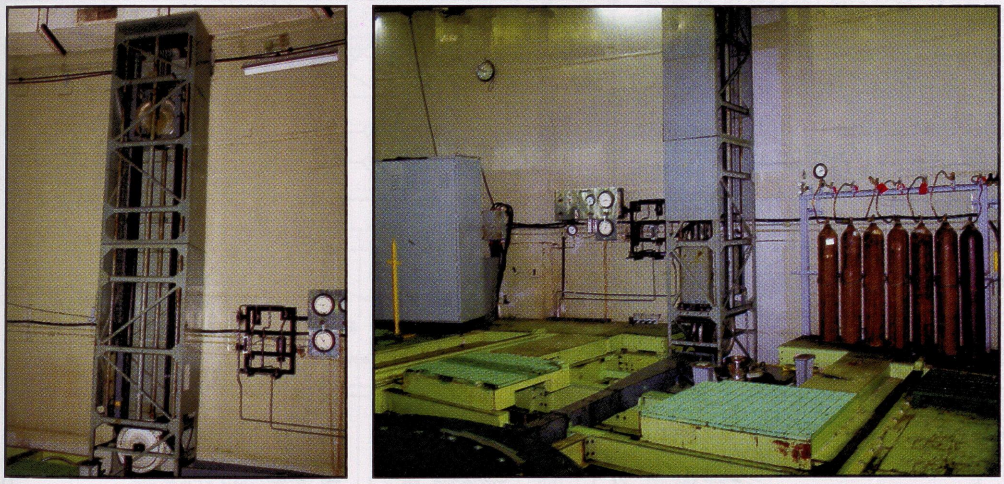
With this modification, the total length of the cable got reduced by 7 m and the total number of cable guide pulleys got reduced from 156 to 76. The cable movement to and from the tower was observed to be very smooth and consistent. After the modification, ten FH campaigns have been completed without any core failure.




*Original Trailing Cable Routing*



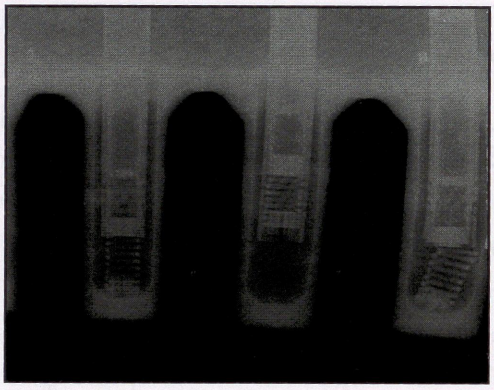

*Modified Trailing Cable Routing*



*Cable tower modification*

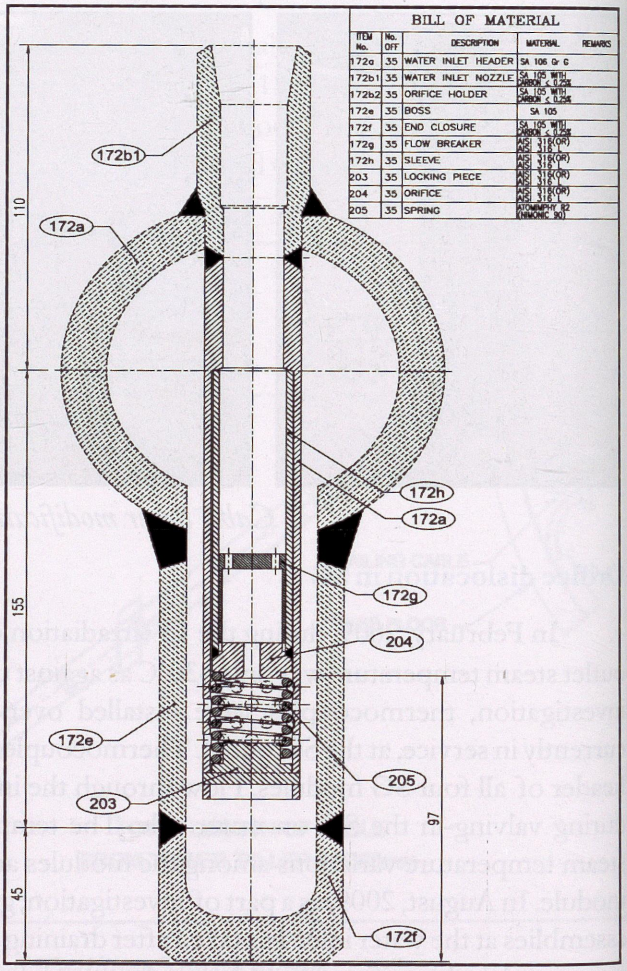
## Orifice dislocation in SG

In February, 2009, during the 15th irradiation campaign at 18.6 MWt, the bulk outlet steam temperature was only 430°C as against the estimated value of 450°C. For investigation, thermocouples were installed over the surfaces of the 16 tubes, currently in service, at the SG outlet. Thermocouples were also installed on the outlet header of all four SG modules. Flow through the individual modules was measured during valving-in the SG on water side. The temperature measured later showed steam temperature variations among the modules and among the tubes of the same module. In August, 2009, as a part of investigation, gamma radiography of the orifice assemblies at the water inlet was taken after draining sodium from the SG modules. It was seen that the orifice assemblies of 22 tubes, out of the total 28 tubes (13 out of

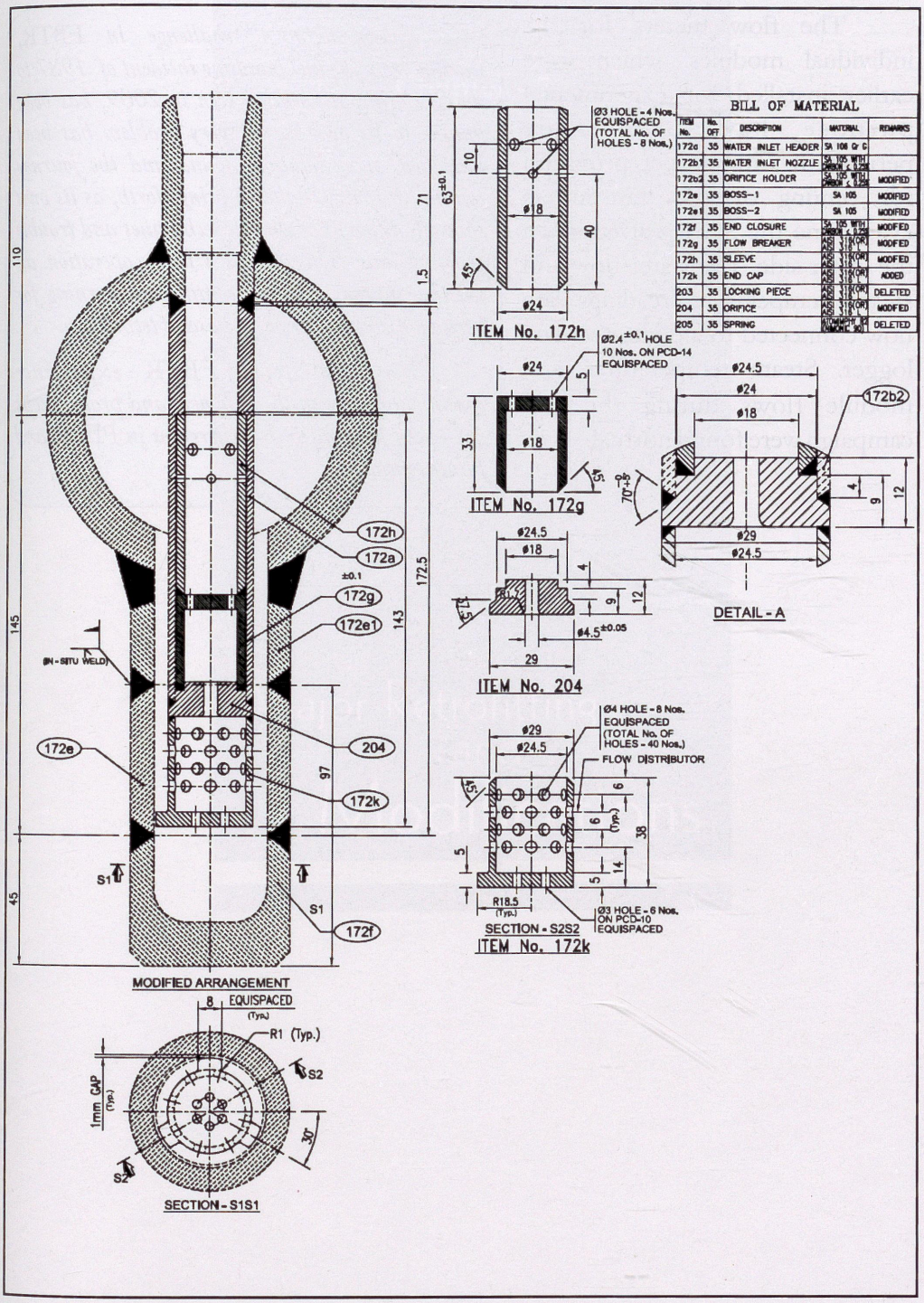


*Dislocated orifice assemblies in the tubes of SG modules*

16 tubes in service & 9 out of 12 blanked tubes), had dropped from their positions. It was seen that springs of some orifice assemblies had deformed permanently in unstable manner. In many cases, the bayonet pin of the assembly had got sheared. There was no erosion of the orifices or the flow breakers. Wall thickness of tubes upstream of the orifice assembly was measured and found alright. The orifice assemblies of spare SG modules were found intact. Based on the fact that the orifices had fallen even in the blanked tubes, it is obvious that the incident had taken place earlier to blanking and had come to light because of operation at high temperature. Spring loaded orifice assembly holder arrangement had been provided by the designer to enable replacement of the orifice, if needed. It is suspected that during depressurisation of SG, the downward thrust on the orifice assembly might have caused the dislocation of orifice holders and springs. The orifices were replaced with a welded design before the starting of 16th irradiation campaign. The replacement of the orifices was completed by December, 2009, leading to a downtime of about six months.




*Original spring loaded design*




*Modified welded design*

## Challenges on the way

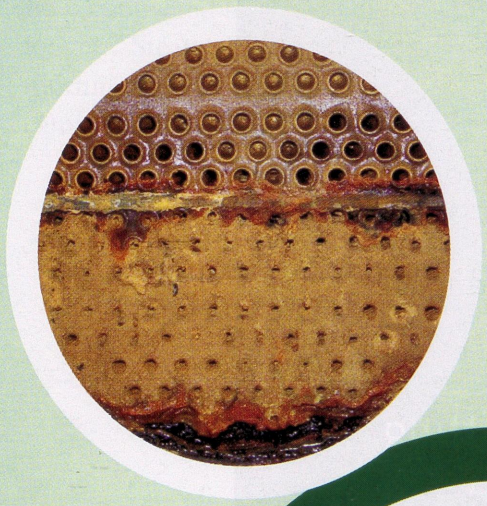
The flow meters for the individual modules, which were earlier installed for experimental purpose, have been made permanent with proper provision for venting of the transmitters every time the SG is valved-in on the water side. The water flow and steam temperature readings are now connected to a dedicated data logger. Steam temperatures and module flows during the 16th campaign were found normal.

*Every technical challenge in FBTR, starting from the fuel handling incident of 1987 to the SG orifice dislocation seen in 2009, has been unique in its own way. Every problem has been identified, right solutions found and the journey continued. Every challenge brings forth, as its own spin-off, different diagnostic techniques and trouble shooting aids. This makes FBTR operation an exciting journey towards continuous learning for perfection and mastery of a nascent technology.*

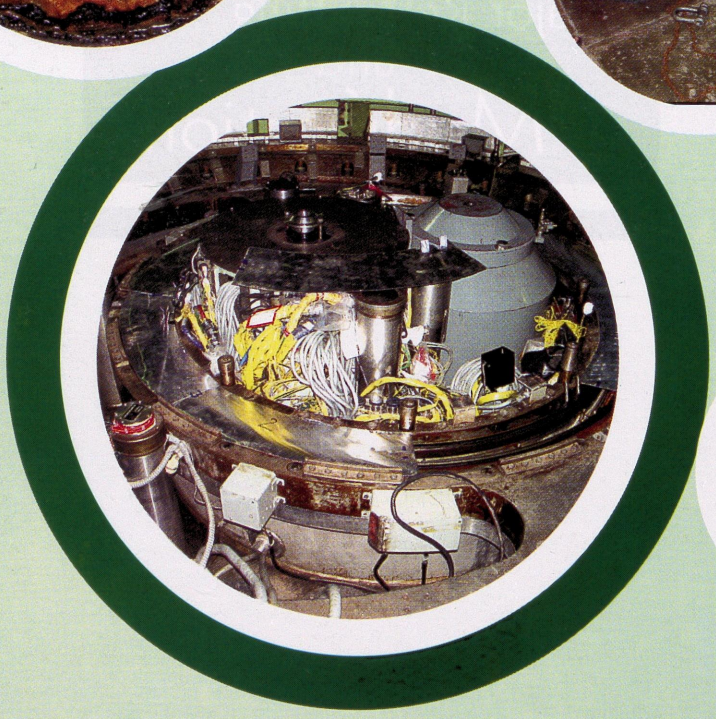
*Feedback from FBTR experience continuously goes to the designers and proper steps are taken to avoid such recurrences in PFBR and future reactors.*

Chapter - 7

# Major Retrofitting and Modifications









# Major Retrofitting and Modifications

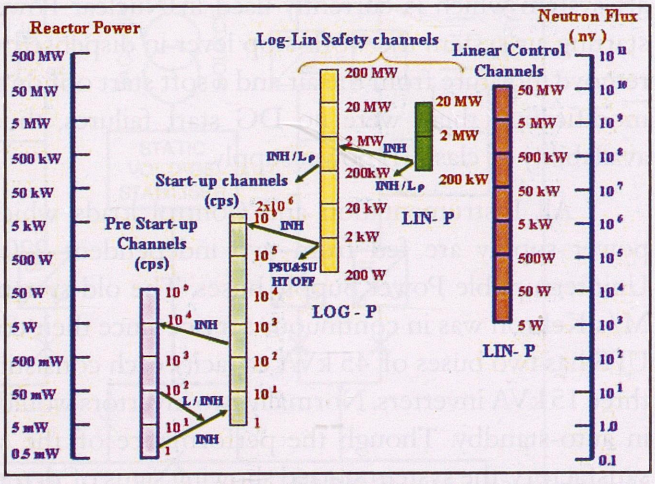
## Major Retrofitting and Modifications

Over the years, several major retro-fitting measures and modifications were carried out based on the operating experience. These were required for improving plant safety, reliability, operator convenience and to take care of obsolescence. Some of the modifications were to overcome the deficiencies observed during the commissioning stage and some were based on the review of incidents in the plant. The retro-fitting and modification measures together accounted for about 16% of plant outage since the first valving in of water into the SG. A total of 565 modifications have been carried out in the plant since 1992.

Most of the major refurbishments were carried out during the period 1997-99. Additional refurbishment activities have now been taken up as a XI five year plan project on Life Extension of FBTR.

### Neutronic Channels

The old neutronic channels were replaced with new generation channels in 1998 to overcome the problems of obsolescence of components and difficulties in maintenance. The new neutronic channels were designed and developed with the state of the art technology and all the prototype channels were designed, developed, fabricated and tested at Electronics Division, BARC. The know-how of these channels was subsequently transferred to M/s. Electronics Corporation of India Limited for production and installation at FBTR. The noise pick-up was overcome by optimising the gain and disc bias in the channel along with improved grounding. The channels are designed with minimum interconnections in the cabinets and therefore, spurious faults originating on account of loose connections were eliminated.




*Operating ranges of Neutronic Instrumentation*

In addition, optical isolation was provided instead of the pulse transformers in the earlier design, making it immune to noise pick-up. The modular design helps in carrying out the on-line maintenance with ease and less down time. The calibration and testing of the channels are more user friendly compared to the previous channels as there is no requirement for drawing out the chassis, thus avoiding the possibility of the cable connectors getting disconnected. During the period from 1998 to 2010, there were no component failures except a few power supply modules (+15V). This was overcome by replacing the modules with modified design.

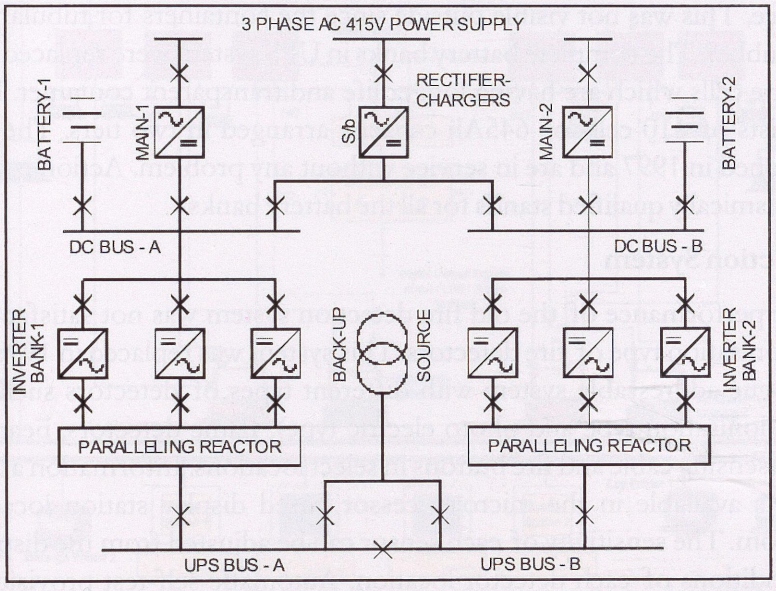
## Electrical Power Supply

Off-site power is fed from two independent feeders. Initially, one of the feeders was from MAPS and the other from Tamilnadu Electricity Board. The availability of the latter was poor and therefore, the second feeder also was changed over to an alternate feeder from MAPS. Both the feeders are paralleled at the 6.6 kV level, thus improving the availability of class IV power supply.

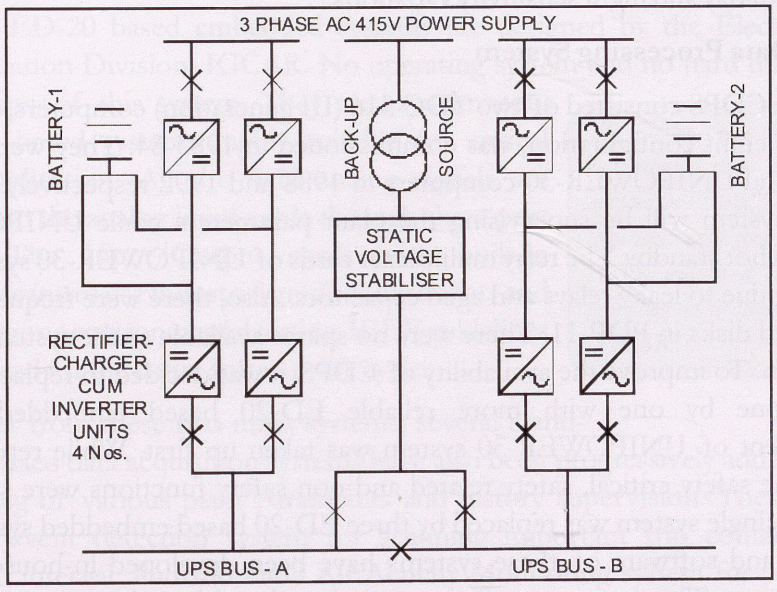
Several modifications have been carried out on the Diesel Generator systems (class-III power supply). These include modifications to avoid inadvertent draining of the diesel day tank, provision of fire water supply for cooling during service water shutdown, replacement of centrifugal pumps with self priming pumps for fuel transfer and replacement of starting air compressors with indigenous make. In the recent years, there were a few occasions when the DG failed to start during periodic surveillance checks. This was investigated in detail and found to be due to the ingress of moisture and dust into the ports of the start-air valves and sluggishness of the work-stop lever. The components in starting air system were obsolete and therefore, the entire starting air system was completely refurbished in 2009 with a state-of-the-art system which is currently used in Nuclear Power Plants in India. In the new starting air system the work-stop lever in dispensed with. It also has a provision to remove moisture from the air and a soft start option for testing purposes. After this modification there were no DG start failures, thus considerably improving the availability of class-III power supply.

All Instrumentation and Control loads which require uninterruptible AC power supply are fed from two independent 220 V, AC, single phase, 50 Hz Uninterruptible Power Supply buses. The old system supplied in late seventies by M/s Keltron was in continuous service since their commissioning in 1980. The old UPS has two buses of 45 kVA capacity each consisting of one rectifier-charger and three 15 kVA inverters. Normally two invertors would be in service and the third one in auto-standby. Though the performance of the system in the initial years was satisfactory, the system started showing signs of deterioration in the nineties. Due to ageing and unsatisfactory transient performance in parallel operation, the old system

needed replacement. The availability and reliability of the UPS system is important for a nuclear plant and therefore, it was replaced by a state-of-art system. The new system has integrated rectifier/charger and inverter in a single module. Each UPS bus has two redundant rectifier/charger and 22.5 kVA inverter connected in parallel. The system was commissioned in June 1998 and is in trouble free operation, thus enhancing the availability of class-II power supply.




*Old UPS system*




*New UPS system*

### *Major Retrofitting and Modifications*

Each UPS battery bank consisted of 100 tubular lead acid batteries of 550 Ah capacity, arranged in a single row, commissioned in 1981. The performance was good up to 1986 without requiring any battery replacement. However, during 1987-91, 90 cells were replaced out of 200 cells in both the UPS banks. The replacement of cells was necessitated since some of the cells were not meeting the capacity during 10 hour discharge test and for some cells shedding of the active material from the plates was taking place. This was not visible outside since the containers for tubular cells were made of rubber. The complete battery banks in UPS system were replaced in 1991 by Plante` type cells which are having longer life and transparent container. Each bank now consists of 110 cells of 645Ah capacity arranged in two tiers. The cells were commissioned in 1997 and are in service without any problem. Action is on hand to provide seismically qualified stands for all the battery banks.

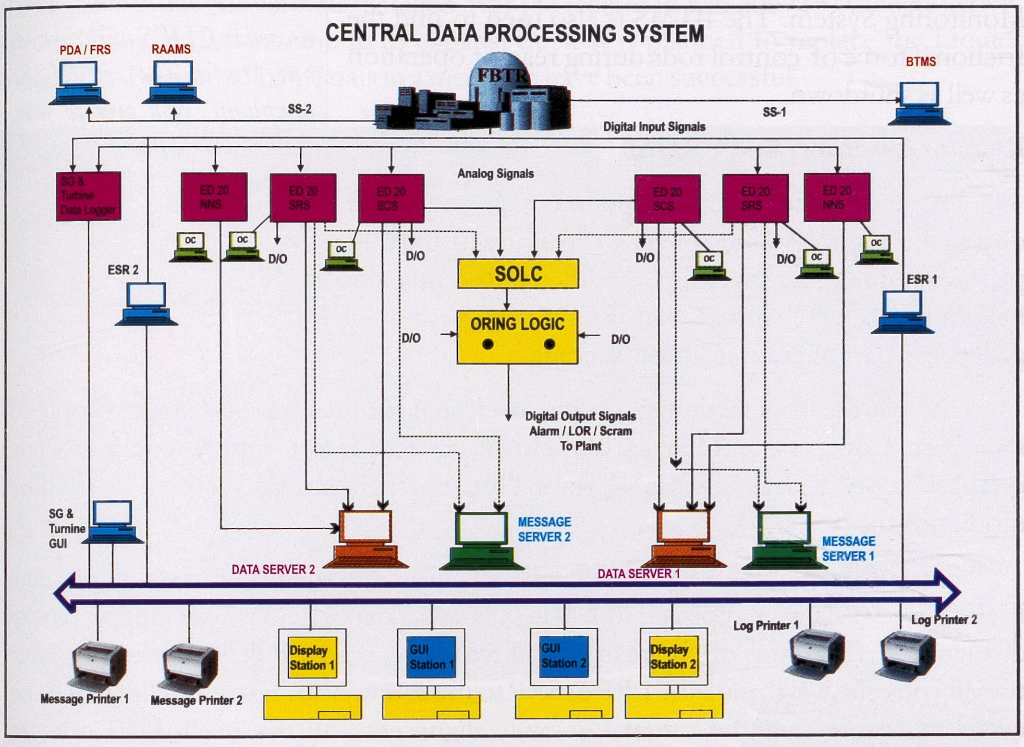
### Fire Detection System

The performance of the old fire detection system was not satisfactory and it had only ionisation type of fire detectors. This system was replaced in 1998-99 with a new analogue addressable system with different types of detectors such as smoke detectors (ionisation type and photo electric type), flame detectors, beam masters, linear heat sensing cable and fire buttons in select locations. Information about all fire detectors is available in the microprocessor based display station located in the control room. The sensitivity of each sensor can be adjusted from the display station to suit conditions of each detector location. Automatic self-test provision ensures that the sensor is functional and calibrated. The system also provides separate program for day and night sensitivity variations.

### Central Data Processing System

The CDPS consisted of two TDC-316 (III generation) computers, connected in fault tolerant configuration, was commissioned in 1983-84. They were replaced PDP-11 and UNIPOWER-30 computers in 1988 and 1992 respectively. Normally PDP-11 system will be supervising the plant parameters while UNIPOWER-30 system is a hot standby. The relay multiplexer cards of UNIPOWER-30 system failed frequently due to leaky relays and aged capacitors. Also, there were frequent failures of the hard disks in PDP-11. There were no spares available as the system is out of production. To improve the availability of CDPS, it was decided to replace both the systems one by one with more reliable ED-20 based embedded systems. Replacement of UNIPOWER-30 system was taken up first. While replacing this system, the safety critical, safety related and non safety functions were segregated. Thus, the single system was replaced by three ED-20 based embedded systems. The hardware and software of these systems have been developed in-house to avoid obsolescence. These systems were commissioned and have been in service since

2007. The PDP-11 system also was replaced by three identical ED-20 subsystems in 2009 and its performance evaluation is in progress.




*Architecture of CDPS*

The ED-20 based embedded systems are designed by the Electronics & Instrumentation Division, IGCAR. No operating system and no hard disk are the main features of this system. They are asynchronous VME bus based systems highly suitable for critical control application. All the field analog signals are connected to the analog input cards through three port isolators. The Input/Output cards with built-in diagnostic features use industrial grade components and have undergone environmental testing. IEEE guidelines were followed for the entire software life cycle.

*The design of the ED-20 subsystems is same as that being adopted for PFBR. The operating experience with these systems in FBTR will go a long way in providing the feedback for its use in PFBR.*

Apart from these two main systems, several stand-alone PC based data acquisition systems have also been progressively added for real time logging of various plant parameters and history supervision. These include sequential event detection system for scanning important trip contacts at 20 millisecond interval, Radiation and Air Activity Monitoring system for processing the signals from Radiation monitors, SG & Turbine data logger for logging signals

from steam/water and turbine systems, data logger for secondary sodium system and Block Pile Temperature Monitoring System. The BTMS is also used to find the frictional force of control rods during reactor operation as well as shutdown.

**Y2K compliance**

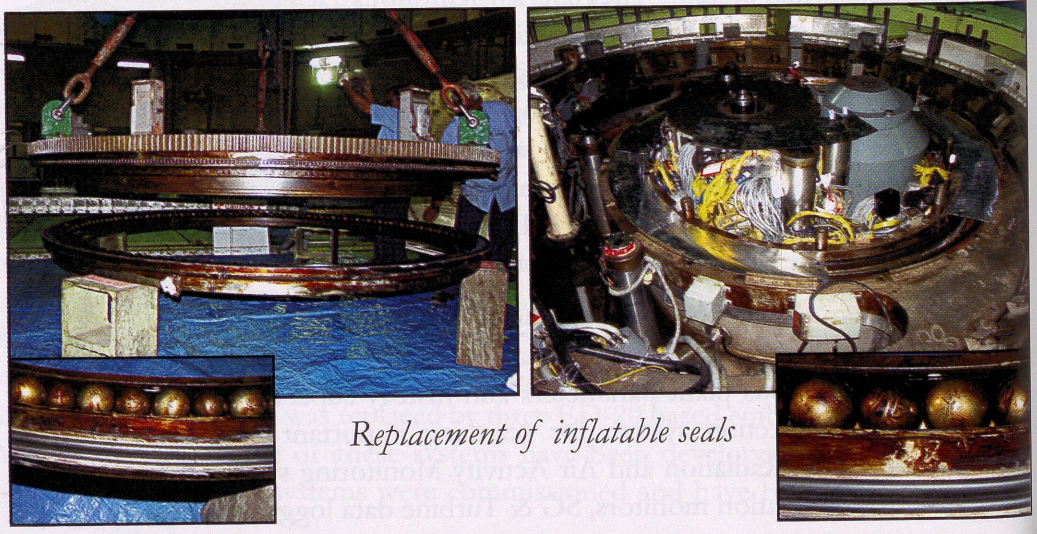
*Modifications that were needed in CDPS were carried out to make it Y2K compliant. Roll over to year 2000 was smooth in both systems of CDPS and successfully demonstrated Y2K compliance.*

## Reactor Assembly and CRDM

A permanent Helium injection circuit was added during the commissioning phase to take care of the grid plate shift due to temperature asymmetry. Recently, the helium banks were shifted from RCB to the peripheral gallery as a part of seismic house-keeping.

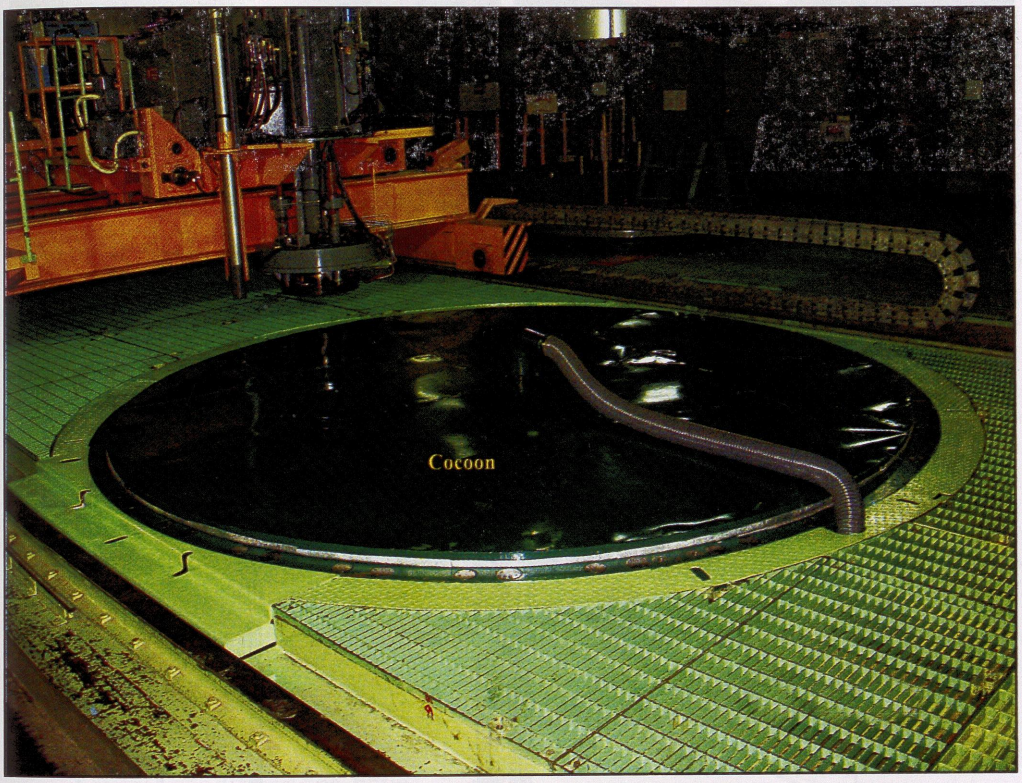
As per the original design, provision existed for the inter-seal argon supply to the Core Cover Plate Mechanism. However, the argon supply could not be established due to leaky inflatable seals. This shortcoming was rectified in 1996 by modifying the circuit.

The inflatable seals on the LRP and SRP failed during the commissioning stage itself. These seals were replaced in 2004. This was a major activity warranting partial dismantling of the top of the reactor and removing the LRP & SRP bearings after transferring the weights of the LRP & SRP to the screw jacks. Extreme care had to be taken to ensure that the original as-built alignments of the reactor do not get disturbed. Though the inflatable seals were successfully replaced, leak-tightness could not be achieved. This was because some of the elastomer seals in the inter-seal space are inaccessible for replacement. Nevertheless, this experience has given us the confidence to undertake major activities on the reactor.



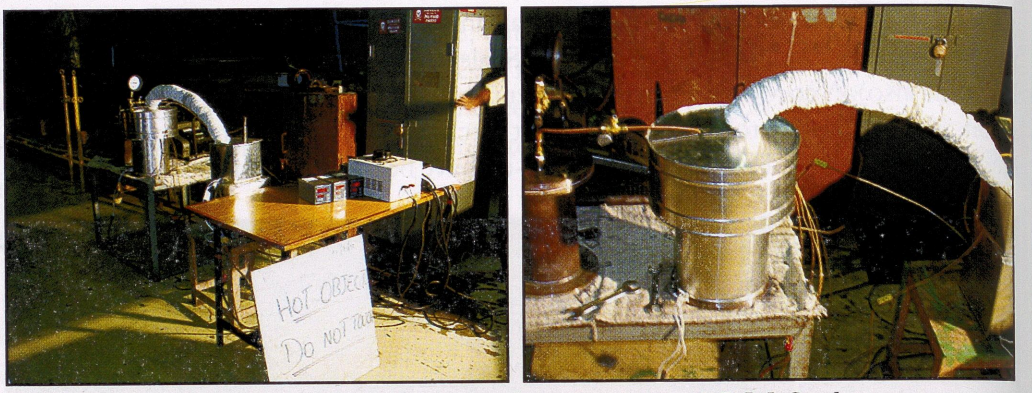
*Replacement of inflatable seals*

There has, of late, been deterioration in the leak-tightness of the reactor vessel cover gas, due to ageing of the LM Seals. (This phenomenon was observed in Rapsodie also). A cocoon was installed over the reactor in Feb 2009 to evacuate active argon, if any, directly to the RCB exhaust duct. It is planned to replace the Liquid Metal (Tin-Bismuthalloy). Trials in a mock-up have been successful.

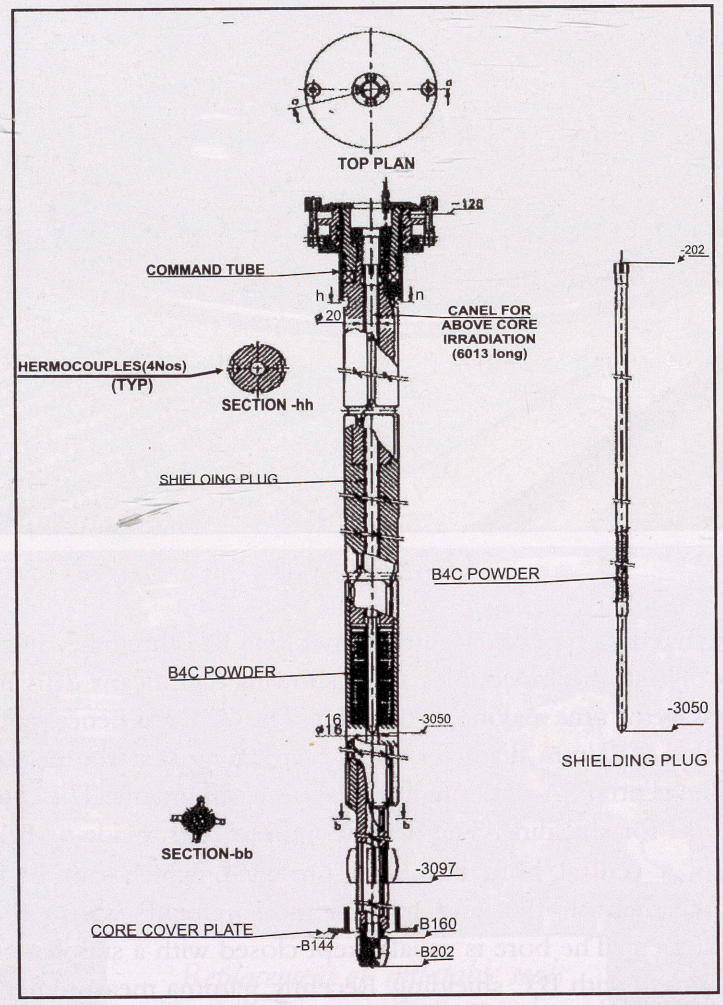



*Cocoon over the pile*

The original design of the Central Canal Plug had three thermocouples of 1 mm diameter. These thermocouples were showing significant drift in the signal values with passage of time making it unreliable. The CCP was hence redesigned with 4 thermocouples of 2 mm diameter, after confirming that the response time is adequate for the central SA outlet temperature measurements. The original design had borated steel for shielding. The new design has been made with $B_4C$ powder shielding. Also, a central bore has been provided, which can be utilised for experiments like neutron flux and gamma measurements above the core, foil activation studies etc. The bore is usually kept closed with a supplementary shield plug of 2.7m length with $B_4C$ shielding. Recently, gamma measurement above the core was carried out using this provision.




*Mock-up for removal and replacement of LM Seal*




*Modified Central Canal Plug*

There were a few bellows failures in CRDM. In the case of silicon bellows, the defective ones were replaced in-situ. Three lower parts of CRDM have also been replaced so far due to failure of metallic bellows. One control rod was removed from the reactor for carrying out PIE.

There was an interesting incident of reverse rotation of the CRDM motor on failure of one of its phase windings. When the command was given to the mechanism to lower, it started rising. Analysis of the power supply circuit indicated that such possibility exists due to the tapping of brake power supply across the Y&B phase of the motor after the overload relay. This resulted in unbalanced 3-phase power supply to the motor through the brake power supply transformer causing reverse rotation. Modifications were carried out in the CRDM power supply circuit after safety clearances.

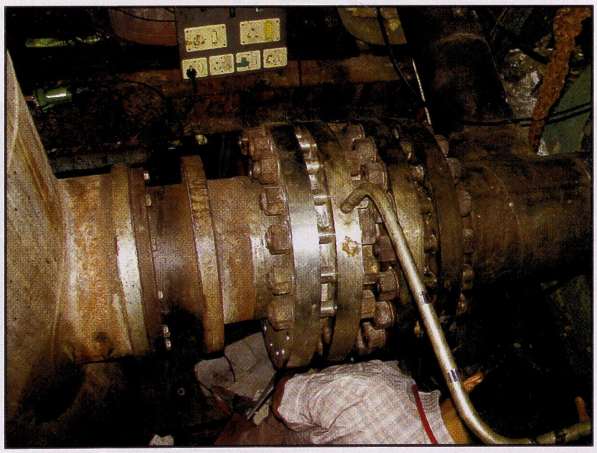
All the rectangular type power & signal disconnectable connectors of rotating plugs were replaced with circular connectors due to frequent problems of shearing/bending of the central locating pins.

### Steam-Water System

Nearly 100 engineering changes have been carried out in the steam water system. The major one was the replacement of the contact type LP heaters by non-contact type, which resulted in a large improvement in plant availability. Both the imported MBFPs were replaced by indigenous make of better design and also due to non availability of spares. DLPs were replaced by new pumps of more rugged and maintenance-friendly design.

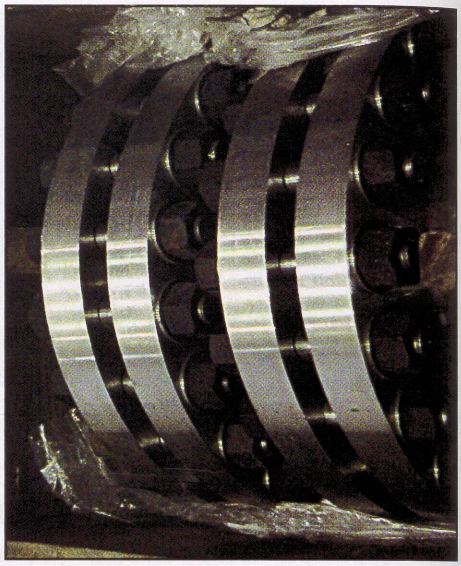
### Sodium Systems

Several modifications have been carried out for avoiding the choking of vapour traps, hot argon lines and solenoid valve ports. Wire-type leak detectors were installed on the bends on the steam generator shells to detect any sodium leak in the incipient stage. The leak detection (spark plug and wire type) circuit of sodium leak detectors is modified to facilitate on-line monitoring of unsafe-failures such as any discontinuity in wiring.



*Old rupture disc assembly*

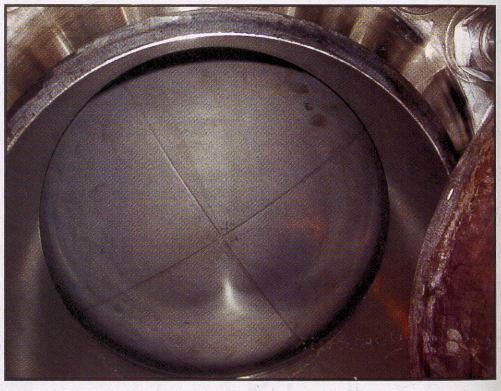
The 'knife-edge' type double rupture disc assemblies in the west SG inlet and outlet headers were replaced by 'scored' type double rupture disc assemblies as part of periodic surveillance. Old rupture disc assembly removed from the loop was tested for rupture. The rupture pressure was as per the original design intent, indicating no degradation due to sodium service for nearly two decades.



*New Rupture disc assembly*

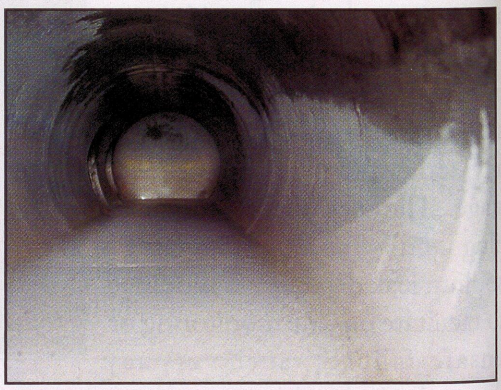


*Removed 'knife-edge' type rupture disc assembly*



*'Scored' type rupture disc assembly*



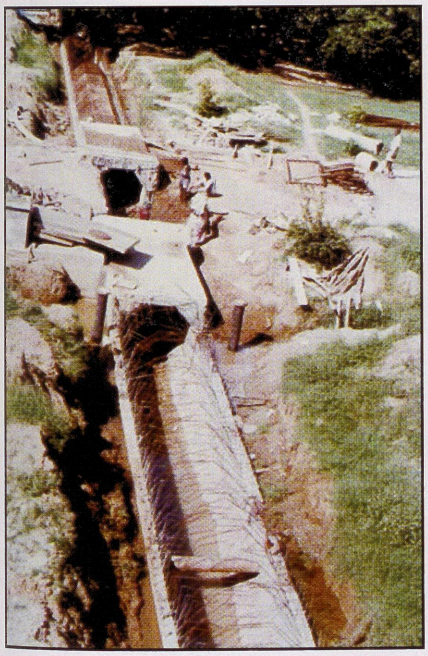


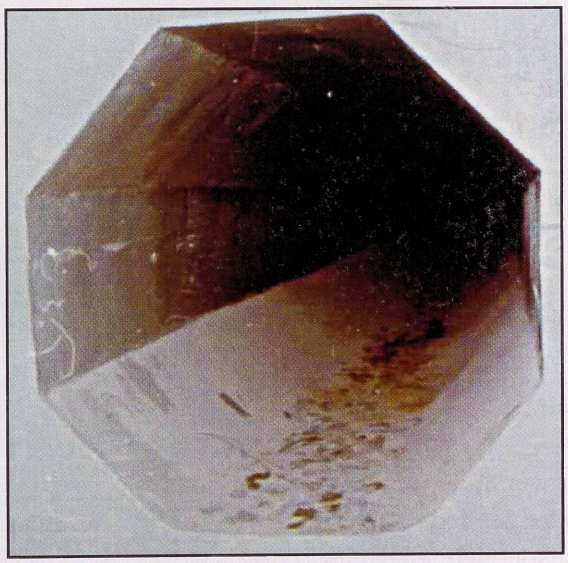
*Solid sodium seen in SG sodium header during rupture disc replacement*

## Cooling Water System

Originally, the outlet water from cooling tower was routed back to service water pit (from where cooling water pumps take suction) by gravity through 700 NB carbon steel pipes. Due to mismatch between quantity of water pumped and quantity of water returned, there was frequent overflowing of cooling tower basin and service water pit low level problems. In order to overcome this, the sizing of the return line was increased by constructing a concrete conduit of octagonal shape with 1.8m diameter inscribed circle with three inspection chambers. All the issues associated with the return line got solved with this modification.

The replacement of the rupture disc assemblies in the SG was a major intervention in the sodium system extending to about three months in 2008. The perfect execution of the job, without any incident of sodium fire has provided ample demonstration of our capability to intervene in sodium systems in a big way. In fact, sodium purity on completion of the job, was found unimpaired, confirming that all the precautions taken

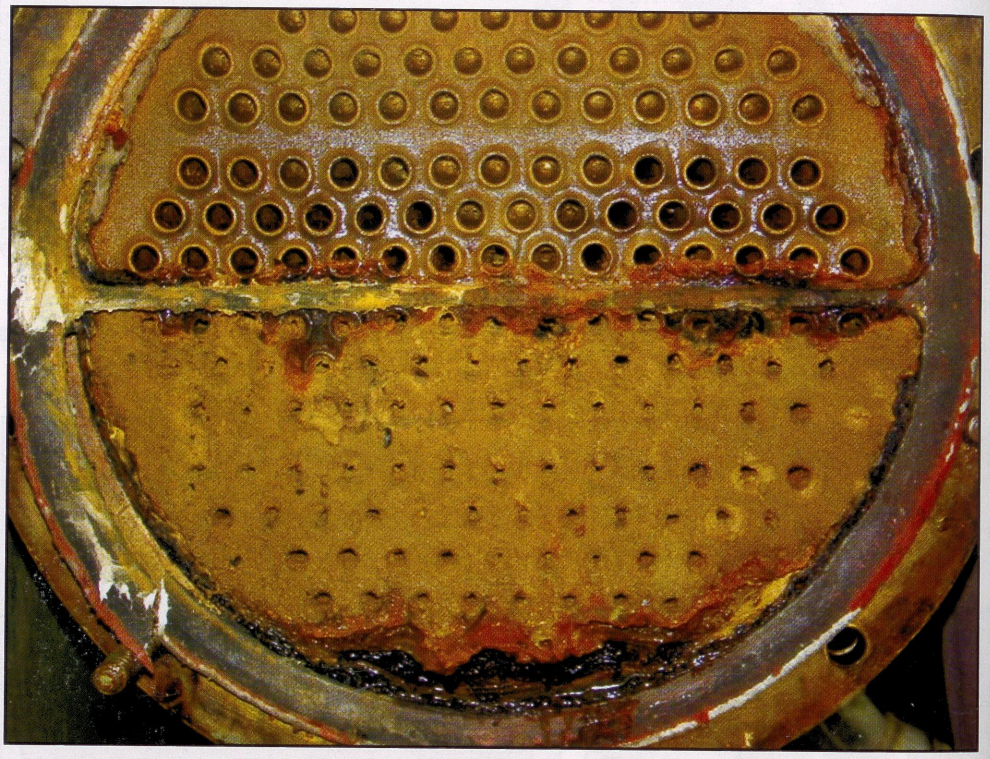




*The Octagonal Concrete channel connecting the cooling tower and service water pit*

In the initial years, the service water system was plagued with severe corrosion problems. Pin-holes developed frequently, calling for interim measures to arrest the leaks using rubber gaskets with clamping. Several remedial measures were taken to improve the condition. A side-stream filtration unit was added to the cooling tower to

avoid pick-up of dust, sand and organic vegetation while passing through the cooling tower. Chlorination system was replaced. All the pipelines of smaller diameters were replaced completely. Large diameter lines were chemically de-scaled. The thermofluid heat exchanger tube bundles were changed over to admiralty brass and one PHEC heat exchanger to cupro-nickel. A programme of chemical control by proprietary chemicals is in place, which has effectively avoided all problems of corrosion and deposition.



*Corrosion and deposits observed in a carbon steel heat exchanger*

Treated wood was used as fill material and drift eliminators of the cooling tower. Due to poor aging resistance of the wood, frequent replacement of the fills was needed involving large downtimes. The decay of the wood resulted in more suspended solids in cooling water, leading to chocking & deposition in various heat exchangers, strainers and pipelines. The wooden fills were replaced with PVC 'V' bars supported on glass reinforced plastic grids. The existing wooden drift eliminators were changed with cellular PVC drift eliminators, reducing the drift losses and the frequency of choking of strainers & heat exchangers considerably indicating the cleanliness of the cooling water system. There has also been a drastic improvement in the heat removal capacity of the cooling tower.



*Refurbishment of Cooling Tower with PVC fills*

### Fire water system

The existing system comprises of buried cast iron pipes. Over the years, these pipes have developed cracks and there have been a few instances of pipe burst. The identification and replacement of the failed spool is found to be time consuming, resulting in violation of the Technical Specification. The entire fire water system is being replaced by over-ground system of carbon steel pipes.



*Over-ground Fire water system*

## Major Retrofitting and Modifications

### Chilled water and AC &Ventilation

The chilled water system has been augmented by replacing the old chillers by three state-of-the-art chillers: two of 150 TR and one of 350 TR capacities. Air handling units are being replaced by compact, double-skin units progressively.

### Nitrogen plant

Initially, FBTR was having a nitrogen production plant of 65 $Nm^3/h$ capacity working on cryogenic principle. Many problems were faced in the operation of this plant in the initial years and nitrogen cylinders were brought from outside to meet the requirements. To overcome this problem, a nitrogen plant based on pressure swing adsorption principle was procured and commissioned. On successful operation, the plant capacity was augmented to $65m^3/h$.

### New delay tank in liquid effluent system

To meet the regulatory requirements, a new delay tank of 150 $m^3$ capacity (underground concrete tank with two inter connected compartments of $75m^3$ capacity each) was constructed and commissioned at the south east corner of FBTR complex in 2003. Each compartment is provided with a separate pump with common discharge header to discharge the effluent to CWMF. The pump house is located just above the delay tank. The inlet pipe lines to the FBTR delay tank are branched off from existing FBTR-RML lines near south east corner of RML building. Two bore-holes are provided for sample collection. With the commissioning of this tank, the problem of liquid effluent storage and the dependence on RML delay tank is eliminated. However, the RML delay tank will be used in case of emergencies.

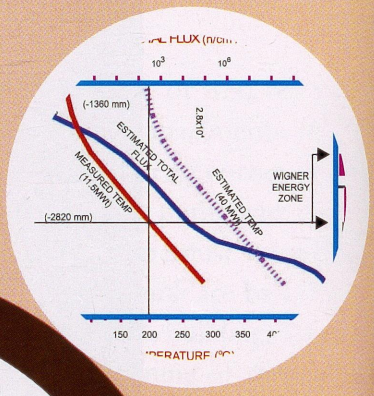
### Augmentation of ventilation system of SGB & TB

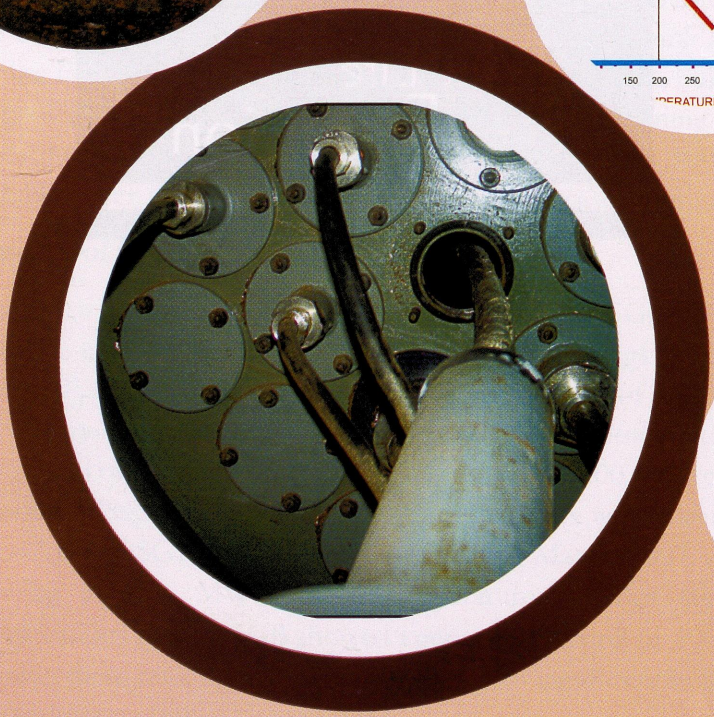
In order to bring down the ambient temperatures in SGB&TB, it was decided to augment the existing ventilation system of these buildings. The TB&SGB were originally ventilated only by means of three tube axial fans of 85000 $m^3/h$ capacity each. The augmented system comprising three supply blowers of 125000 $m^3/h$ capacity each and twelve wall mounted exhaust fans of 20000 $m^3/h$ capacity each was commissioned in September, 2006. The supply blowers were located in the newly constructed TB and SGB supply fan rooms. During March, 2007, when the reactor was operating at 16.6 MWt power, temperature readings taken at different locations/elevations in TB&SGB showed a general reduction of $10^0C$ in the ambient temperature in these buildings. This helped in improving the cooling for MBFPs and Secondary sodium pumps to a large extent.

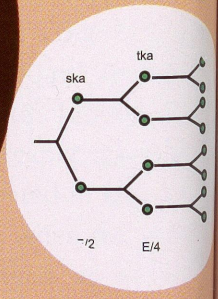
Chapter - 8

# Plant Life Extension




$10^3$
$10^6$
(-1360 mm)
2.8x10
ESTIMATED TOTAL FLUX
MEASURED TEMP (11.5MW)
ESTIMATED TEMP (40 MW)
WIGNER ENERGY ZONE
(-2820 mm)
150
200
250
300
350




tka
ska
E/4

# Plant Life Extension

## Plant Life Extension

Exercises for assessing the residual life of non-replaceable and difficult-to-replace components of the reactor and heat transport systems started in 2002, three years ahead of the assumed design life of 20 years. It may be remarked here that there does not exist a designated name plate design life for FBTR, which is of vintage design. This is due to the limitations of the design codes available when Rapsodie was designed. Fast reactor component codal design lives are generally based on creep and fatigue considerations. In the sixties, nuclear codes for high temperature creep design were yet to evolve. Rapsodie reactor components were all design checked as per the Boiler & Pressure Vessel Codes which had provision for high temperature design only up to 100,000 hours, which corresponds to about 11.5 Effective Full Power Years. With about 60% plant availability factor (which is expected of a test reactor), the design checked life would be around 20 calendar years. Thus, it had been tacitly assumed that FBTR life is 20 years. In fact, the stresses in sodium systems are so low that the actual allowable life would be much longer. The piping had been designed as per Power Piping codes for 7000 thermal cycles, large enough for the start-up & shutdown cycles of a test reactor for more than 20 years.

## Degradation mechanisms

### Creep and Fatigue

The stress levels in FBTR components are so low that the actual allowable life from creep would be much longer. During the FBTR project days, ASME brought out what are called `code-cases' for dealing with nuclear components operating in the creep range, i.e. above 427°C (800°F) for austenitic stainless steel and 371°C (700°F) for ferritic steels. These code cases also enabled calculation of the synergic effects of creep and fatigue to define the creep-fatigue limit of any component. All the nuclear components of FBTR were checked as per the code cases for creep of 100,000 hours and fatigue of 2000 thermal cycles. The creep-fatigue damage was found to be far below the allowable limits. Hence, FBTR life limit does not come from creep-fatigue damage. The creep clock may be said to have started ticking only since December, 2008 as the operating temperatures up to the 14$^{th}$ campaign were below the creep range of SS 316. Preliminary assessment shows that the nuclear systems have exhausted less than 10% of the original design life from creep-fatigue considerations.

Certain localised areas where hot sodium mixes with cold sodium are prone to what are called 'striping' stresses. Four such critical areas were identified, one each on each of the primary and secondary loops. In the secondary, the system has been modified. Regarding the primary, calculations show that stresses are within limits for the residual life of the plant.

## Loss of Corrosion Resistance

Prolonged operation of stainless steel above 500°C can lead to 'sensitisation' due to the formation of chromium carbides, deteriorating the corrosion resistance of stainless steels. Reactor grade sodium is extremely compatible with steel. Even sensitised stainless steel is compatible with reactor grade sodium. Hence corrosion is not a major issue for sodium cooled reactors, if the sodium purity is well maintained. The sodium purity in FBTR has always been maintained below a plugging temperature of 110°C (i.e. oxygen < 1 ppm). Periodic visual inspection of the RV internals is carried out using a periscope every two years. These inspections have shown nothing abnormal. Subassemblies and fuel pins which have stayed in the core for more than 20 years have been found to retain the original shining appearance and no evidence of corrosion by sodium has been seen.

The chemical composition of SS 316 used in FBTR is modified with restriction on carbon. The SS 316 materials have been procured with test for sensitisation to ensure absence of sensitisation in Heat Affected Zone. The operating history till the end of the 14$^{th}$ irradiation campaign has also been below the range of sensitisation. Hence there is no concern from this angle for FBTR sodium systems. In order to confirm this, replica micrographs were taken from the high temperature secondary systems and have shown practically virgin micro-structure. There has only been a slight discolouration of the external surfaces, which is common for stainless steels at high temperature in contact with air.

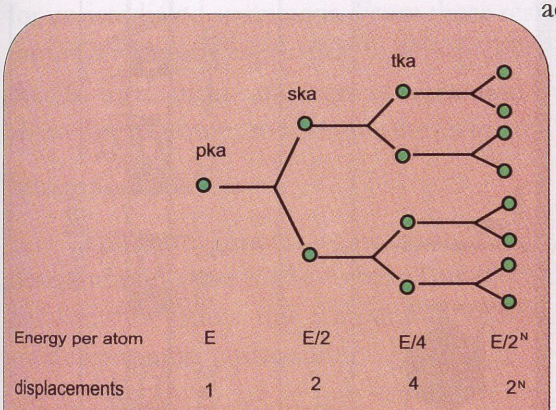
## Embrittlement due to Thermal Ageing

The other life limiting factor may come from embrittlement due to thermal ageing. It is known that austenitic stainless steels tend to lose their impact strength due to prolonged ageing at above 500°C. The mechanism of embrittlement is both carbide precipitation and sigma phase formation. The chemical composition chosen for FBTR is controlled to avoid/minimise embrittlement. Whereas the ASME code does not call for impact test for austenitic stainless steels, all the SS raw materials used in FBTR have been tested for impact strength in the original condition as well as after accelerated ageing simulating the end-of-life state. The weld metals are more susceptible to embrittlement due to the transformation of delta ferrite into sigma phase. The welding specification for FBTR imposes delta ferrite limit between 2 &

as against minimum of 5 specified by ASME Sec-III-NB. The welding electrodes have also been qualified by impact testing after accelerated ageing simulating end-of-life state. Hence, there is no concern of embrittlement. Again, the operating temperatures till the end of the 14th irradiation campaign have been low enough to be of any concern of thermal embrittlement.

## Irradiation Embrittlement

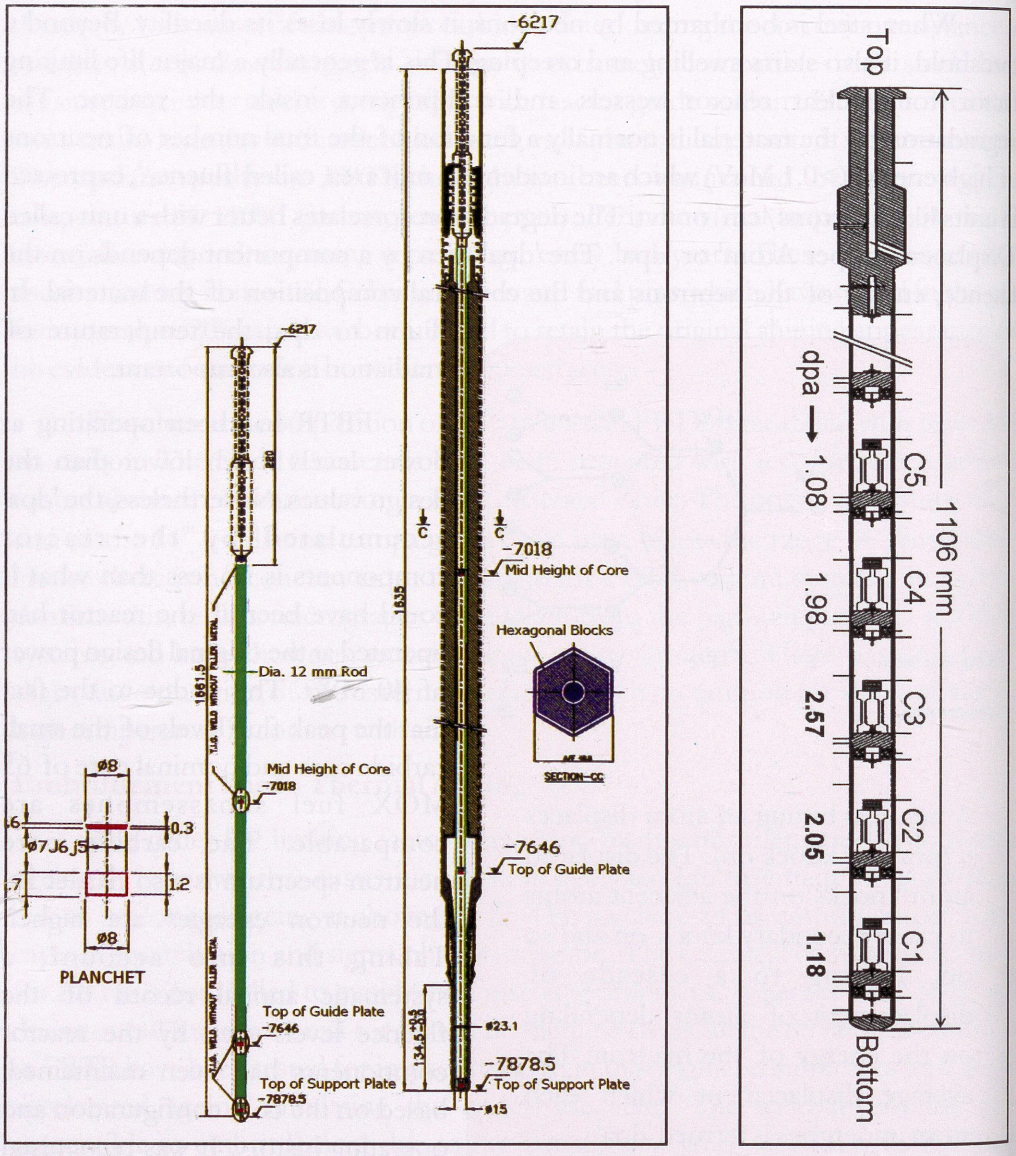
When steel is bombarded by neutrons, it slowly loses its ductility. Beyond a threshold, it also starts swelling and creeping. This is generally a major life limiting factor for nuclear reactor vessels and components inside the reactor. The degradation of the material is normally a function of the total number of neutrons of high energy (>0.1 MeV) which are incident on unit area, called 'fluence', expressed in units like neutrons/$cm^2$ or nvt. The degradation correlates better with a unit called 'Displacement per Atom' or 'dpa'. The 'dpa' seen by a component depends on the fluence, energy of the neutrons and the chemical composition of the material. In addition to dpa, the temperature of irradiation is also important.




A neutron hitting an atom displaces it (primary knock on). The displaced atom knocks on the adjacent atoms to cause secondary knock-on and so on, leading to a cascade of displacement of atoms depending on the energy of the neutron. The average displacement which each atom undergoes is termed 'dpa'.

FBTR has been operating at power levels much lower than the design values. Nevertheless, the 'dpa' accumulated by the reactor components is no less than what it could have been if the reactor had operated at the original design power of 40 MWt. This is due to the fact that the peak flux levels of the small carbide core and nominal core of 65 MOX fuel subassemblies are comparable. The carbide core neutron spectrum is also harder i.e. the neutron energies are higher. Taking this into account, a systematic annual record of the fluence levels seen by the reactor components had been maintained, based on the core configuration and operating history. It was recognised that the life limiting component

from neutron induced degradation would be the Grid Plate. Designers modelled the various scenarios which would impose the fluence limit. These include loss of ductility, overall deformation of the Grid Plate due to irradiation creep and upward curving due to differential swelling across the thickness. It was concluded that the limiting fluence comes from the allowable residual ductility. Permanent structures should have a residual ductility of at least 10% as per the practices followed in USA. Based on literature, the corresponding fluence limit was taken as $2.0 \times 10^{22}$ n/cm$^2$,




*The special SA for flux measurement at Grid Plate location*

giving a residual life of 11 Effective Full Power Years for the Grid plate. It is worth noting here that unlike other reactors, the core of FBTR has been evolving over the years, and the reactor power also keeps on changing with the core configuration and burn-up status. Hence, the term 'Effective Full Power Year' for FBTR connotes the number of years of operation at the peak design LHR of 400 W/cm, since the flux is proportional to the peak LHR. It is not linked with a specific power level.

In order to validate the calculations of flux, a special carrier with Np foils was loaded in the reactor. The capsule extended up to the Grid Plate level. The threshold neutron energy for fission in Np is above 0.1 MeV, and since most of the damage comes from neutrons with energy range above this level, Np was considered the best choice. The flux at the Grid Plate, as calculated from the fission rate in Np, was found to be about 14% higher than the theoretical estimates.

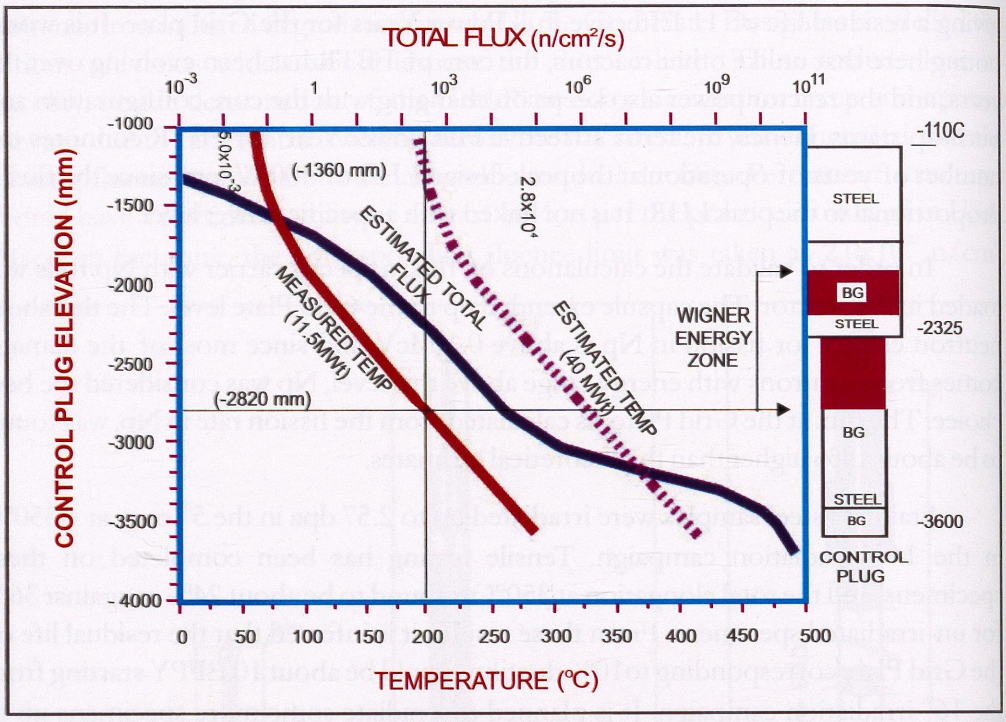
Stainless steel samples were irradiated up to 2.57 dpa in the 5$^{th}$ ring, at ~350°C in the 14$^{th}$ irradiation campaign. Tensile testing has been completed on these specimens, and the total elongation at 350°C is found to be about 24% as against 36% for un-irradiated specimens. From these results, it is inferred that the residual life of the Grid Plate corresponding to10% ductility would be about 10 EFPY starting from the 16$^{th}$ irradiation campaign. It is planned to irradiate some more specimens up to about 5 dpa in order to fine tune the estimates.

### Steam Generators

The steam generators are made of 2.25Cr-1Mo-Nb stabilised ferritic steel. The design temperature for SG is based on 480°C steam conditions. The operation of SG has been far below this temperature. Enough residual life is hence available. The modules are also replaceable.

### Wigner Energy in Graphite

The top shields of the reactor are provided with steel and borated graphite as shielding materials. Graphite, when bombarded with neutrons, tends to accumulate energy if the temperature of graphite is less than 200°C. This is called Wigner energy. If the temperature of irradiation exceeds 200°C, concurrent annealing takes place and Wigner energy is not stored. If graphite with accumulated Wigner energy is heated to above 200°C, an auto-catalytic release of all the accumulated energy will take place which may increase the graphite temperatures to very high levels. It was required to assure that Wigner energy release will not take place when the reactor temperature is increased close to the design values because the reactor had been operated at lower reactor outlet temperatures (430°C) with corresponding lower shield temperatures up to the end of the 14$^{th}$ irradiation campaign. Calculations were




*Wigner energy analysis of Borated Graphite layers of the control plug*

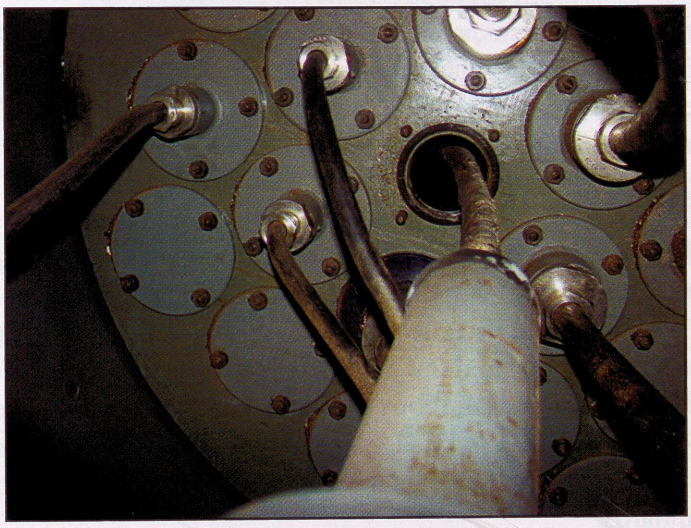
done to estimate the energy accumulated at different layers, based on the temperatures and flux levels for the cumulative operating history. These revealed very low levels of energy accumulation, assuring no concern of any Wigner energy release.

## Maintenance & upgradation of other components

All systems have been well maintained by periodic preventive maintenance as per schedule and their healthiness ascertained through mandatory surveillance checks. Many systems and components have been upgraded or replaced due to ageing and obsolescence, or for improving reliability and safety. Typical examples are the neutronic channels, fire alarm systems, the computer sub-systems of CDPS, Station batteries, UPS system, Cooling tower, MBFP & DLP. Carbon steel tube bundles in important process heat exchangers have been replaced with Cupro-nickel or admiralty brass to minimise corrosion/deposition problems. Most of the service water lines have been replaced. The wooden fills of the cooling tower were replaced with PVC fills. The wooden drift eliminators were changed with cellular PVC drift eliminators. The underground cast iron piping of fire water system is being replaced by over-head system of carbon steel piping. Civil engineering structures are



*The failed underground Cast Iron pipe of Fire water system*



*RCB cable penetration joint*

periodically inspected and maintained. Biennial leak tests of the RCB have shown no degradation of leak-tightness with time. The RCB isolation dampers which required frequent maintenance were replaced with high performance butterfly dampers of fail-safe design.

One of the usual concerns in the life extension of nuclear reactors is the deterioration of PVC/rubber insulated electrical and instrumentation cables due to radiation. In FBTR, all the cables in high radiation zones like B-cells are Mineral

Insulated, SS sheathed cables. There are no PVC/rubber insulated cables inside the cells. Such cables inside the RCB are in areas surrounding the radioactive cells, where the ambient radiation levels are very low (< 10μGy/h) and hence, there is no concern of radiation induced degradation. To find out the extent of the degradation due to normal ageing, cables removed from RCB penetrations were tested and found to have a residual life of more than 25 years.

TG is rated for 16.4 MWe, with steam at 120 bars & 480°C. The turbine had been mostly operated at low power levels using steam at 420°C only. Hence, BHEL had recommended inspection of the turbine internals. A major overhauling of the turbine was hence taken up in 2004. Mild erosion marks were noticed on the tip of the last stage blades. Replica was taken at two more severely eroded locations and it was confirmed that the erosion was not serious. Nevertheless, considering that FBTR turbine is a unique one of its kind, spare blades for the last stage are being purchased.

## Periodic Safety Review

As per the AERB norms, a major periodic safety review is mandatory. The PSR documents covering aspects like ageing management, equipment qualification, safety track record and revised safety analysis have been submitted to AERB and are under regulatory review.

## Seismic Re-evaluation

Rapsodie was located in a highly seismic zone, and was designed for high seismicity. The reactor vessel was designed for 0.2 g (H) and 0.4 g (V). The internal structures of the RCB were designed for 0.25 g (H) and 0.5 g (V). Other buildings were designed for 0.1 g (H) and 0.2 g (V). At the time of construction of FBTR, Kalpakkam site was under seismic zone no. 1 as per IS 1893 and considered to be free from seismic events. Hence, a full fledged seismic analysis of FBTR systems was not carried out, though the civil buildings were designed for low seismic coefficients as per Indian Standards.

International Atomic Energy Agency guide on seismic re-evaluation of existing Nuclear Power Plants states that an assessment of seismic capacity of an existing operating facility can be prompted for the following:

(a) Evidence of greater seismic hazard at site than expected before, owing to new or additional data and/or to new methods.

(b) Regulatory requirements, such as periodic safety reviews, to ensure that the plant has adequate margins for seismic loads.

(c) Lack of anti-seismic design or poor anti-seismic design.

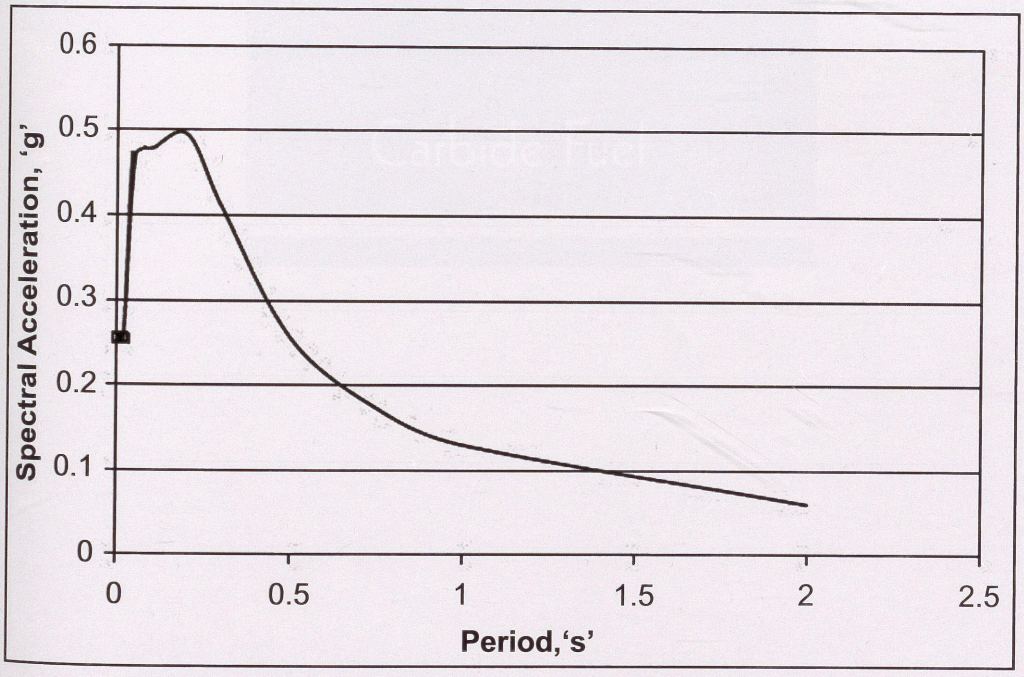
(d) New technical findings such as vulnerability of some structure (e.g., masonry walls) or equipment (e.g., relays), other feedback and new experience from real earthquakes.

Design Basis Ground Motion parameters for PFBR site at Kalpakkam is at a higher level of seismicity than the one adopted in the design of FBTR. Moreover, the seismic analysis methodologies have made rapid strides over the years. The design criteria also have undergone changes. Hence a seismic re-evaluation of FBTR was carried out as a research project of IGCAR and SRI.

As per the IAEA guide, for older plants, the seismic re-evaluation has to address the following safety functions:

- safely shutdown the plant
- maintain the plant in safe shutdown condition
- long-term decay heat removal
- containment/confinement of radioactive inventory

A Task Force was formed in November, 2004 for the seismic re-evaluation of FBTR. Event trees and fault trees were prepared to identify the list of structures, systems and components required to be qualified for meeting the above objectives.



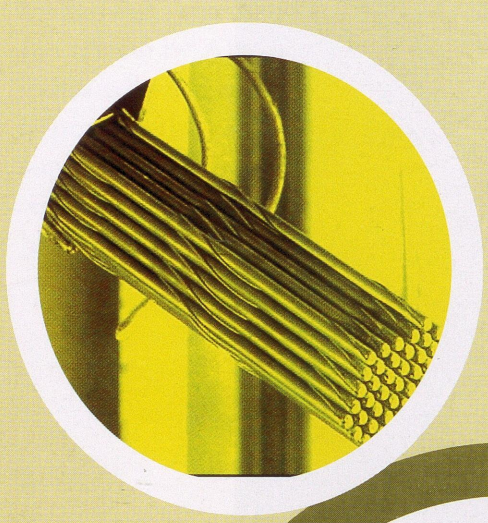

*Review Basis Ground Motion used for Seismic Re-evaluation of FBTR*

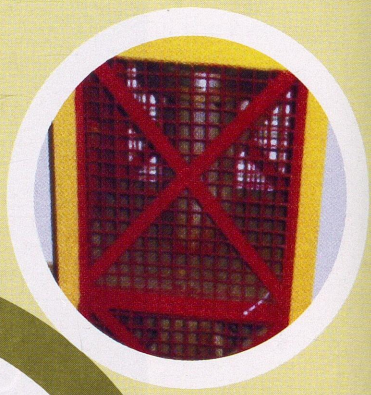
The list boiled down to 11 structures and 389 equipment & components. Major civil structures like RCB, Steam Generator Building, Service Building, Control Building, Cooling Tower and RCB isolation wall were analysed for the Review Basis Ground Motion.

The primary and secondary sodium systems along with several auxiliary systems were analysed for the floor response spectra generated from the civil structure analysis. Many of the smaller components were qualified based on generic data and a few were qualified by shake-table test at IGCAR. Plant walk-down was carried out twice. The entire research project was completed by January, 2008, and the review by AERB is in advanced stage of completion. The analyses have not indicated the need for any major retro-fittings except strengthening the brick walls, improving the cable trays in RCB and providing additional supports to the RCB supply air ducting. Most of the latent weaknesses were identified during the plant walk-down rather than in the analysis. It is planned to complete all the easy fixes by March, 2011 and complete the other major activities by March, 2012.

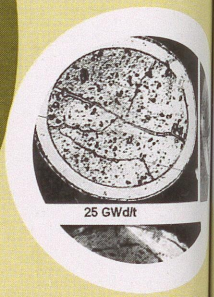
Chapter - 9

# Carbide Fuel








25 GWd/t

# Carbide Fuel

## Carbide Fuel

The high Pu carbide fuel of FBTR is a story of a challenge converted to opportunity. The original selection of the fuel for FBTR was 70% $UO_2$ (enriched to 85% $U^{235}$) -30% $PuO_2$, already a proven fuel for Rapsodie reactor. However, the non-availability of $U^{235}$ forced us to study an alternate fuel of composition 76% $PuO_2$-24% natural $UO_2$ (MOX). But, the preliminary metallurgical investigations carried out on this fuel composition revealed its non-compatibility (volume expansion due to sodium-fuel interaction product in case of a clad breach) with sodium coolant. Though, this non-compatibility issue could have been addressed by opting for lower O/M ratio (1.80), it would have lowered the thermal conductivity of the fuel resulting in an undesirable two phase microstructure.

Thereafter, plutonium rich mixed uranium-plutonium carbide fuel became a promising alternative choice. Compared to the MOX fuel, mixed carbide fuel has a higher heavy atom density, thermal conductivity & breeding ratio and excellent compatibility with sodium. Mixed carbide fuel, considered as an advanced fuel for LMFBR, was then developed in Radiometallurgy Division, BARC. The work started using very small quantities to begin with and a fabrication flow sheet was developed scaling it up to Kilogram level production. A fuel fabrication laboratory was set up taking into consideration of all safety requirements to develop and manufacture mixed carbide fuel for FBTR. It was decided to use plutonium rich mixed carbide of composition $(Pu_{0.7}\text{-}U_{0.3})C$ as the driver fuel (MK-I) for the compact core of FBTR. At that point-of-time, there was not many information available on the Pu rich mixed carbide fuel, especially with regard to its thermo physical properties. Therefore, it became necessary to generate data indigenously. As this composition was first to be used in LMFBR, stringent specifications of fuel were evolved based on the fabrication feasibility and quality control. The first fuel shipment to the FBTR site was made in September, 1984. Subsequently, in 1996, MK-II carbide fuel $(Pu_{0.55}\text{-}U_{0.45})C$ was also made and loaded in the core.

## Out-of-pile studies

The initial mixed carbide fuel specifications were formulated based on a conservative approach since there was no data/information available in the literature for the Pu rich fuel composition with regard to thermal properties, creep, swelling, fuel-clad-coolant chemical compatibility, irradiation behaviour and reprocessing.

Extensive research and development works were carried out to generate sufficient out-of-pile data of this fuel composition to gain confidence in using it as a fuel for FBTR and steer the safety clearance.

The issues of Pu rich carbide fuel compared with that of U rich carbide are lower solidus temperature & thermal conductivity, presence of Oxygen impurity leading to gas phase carburisation of the clad and Pu loss during fabrication. Apart from this, Carbon stoichiometry plays a very important role in determining the extent of Fuel–Clad Chemical Interaction and Fuel–Clad Mechanical Interaction. A hyperstoichiometric fuel is preferred to avoid the formation of metal phase M (Pu+U) during irradiation leading to low melting eutectic alloy of Fe and Ni (from SS-316 cladding). Moreover, C/M ratio decreases with burn-up and in extreme case may lead to formation of metal phase.

Some of the important thermophysical properties, like solidus temperature, thermal expansion, thermal conductivity and thermal toughness (hot hardness) were measured. Solidus temperature of MK-I fuel was estimated by heating/cooling of the fuel sample to different temperatures near or above the approximate melting temperature followed by metallographic examination. Alternately for MK-II fuel, it was carried out using a dilatometer by measuring thermal expansion of the sample as a function of temperature (a sharp transition occurs at the solidus temperature when the pellet loses its shape). The solidus temperatures of MK-I and MK-II fuels were 1875 and 1920°C respectively. The solidus temperature was used in the estimation of the maximum permissible LHR of these fuels.

The average value of the coefficient of linear thermal expansion for MK-I and MK-II fuel between 300 and 1800 K as measured by dilatometer were $13.8 \times 10^{-6}$ $K^{-1}$ and $11.6 \times 10^{-6}$ $K^{-1}$ respectively. MK-I fuel has higher coefficient of thermal expansion than that of MK-II fuel. This was attributed to higher Pu content in the MK-I fuel.

Thermal conductivity was determined from the experimentally measured thermal diffusivity data by laser flash method and subsequently multiplying it with heat capacity of UC–PuC solid solution and density. Thermal conductivity increases with increase in temperature but decreases with Pu content resulting in lower thermal conductivity of MK-I up to about 1100 K, but higher beyond this temperature. The higher conductivity beyond 1100 K is attributed to the contribution from electronic heat transfer by PuC which is inherently a defect structure ($PuC_{1-x}$). At the average working temperature, both MK-I and MK-II fuels have almost similar thermal conductivity. Hence, the thermal performance of these fuels in terms of LHR, etc. may not be much different even considering the marginal difference in the solidus temperatures of the two fuels.

Hot hardness data of fuels can predict the Fuel Clad Mechanical Interaction behaviour resulting from fuel swelling. Swelling of fuel leads to fuel–clad gap closure. Further swelling results in development of back stress which causes restrained swelling, i.e. creeping of the fuel within the available pores. Extent of creep deformation can be determined qualitatively by hot hardness data which is much simpler needing small sample size. The data showed hardness of both MK-I and MK-II fuels decreases with increase in temperature; MK-II having higher hardness at all temperatures. MK-I fuel indicates sharp decrease in hardness at 1123 K (0.52 Tm; where Tm is the solidus temperature) and beyond this the fuel becomes very soft. Hence, for the average volumetric temperature of 1553 K, the fuel becomes very plastic minimising the possibility of any stress generated by FCMI.

Fuel–clad-coolant chemical interaction is a key issue limiting the life of a fuel pin in a reactor. For carbide fuel, this occurs by clad carburisation by solid state (direct contact) 'C' transfer or gas phase carburisation by CO. The extent of clad carburisation depends on the 'Carbon Potential' of the fuel and partial pressure ($P_{CO}$) generated by the reaction of dissolved 'O' with MC. However, Pu rich carbide has some advantages over its uranium rich counter parts. First, PuC and Pu rich MC have some range of carbon stoichiometry, unlike UC and U rich MC which are line compounds. Secondly, higher quantities of $M_2C_3$ phase and 'O' impurity can be accommodated in MC without the risk of significant clad carburisation. In the two phase field of MC/ $M_2C_3$, $M_2C_3$ is richer in plutonium and for Pu rich MC, $M_2C_3$ is virtually $Pu_2C_3$. $Pu_2C_3$ phase has lower carbon potential than that of $U_2C_3$ at all temperatures and with SS-316 (clad material) up to 978 K. Theoretical calculation based on thermodynamic data of free energies of formation of PuC, PuO and $Pu_2C_3$ predicted that both 'C' potential and $P_{CO}$ decreases with increase in Pu. Hence, MK-I and II are less carburising than U rich fuel. Theoretical calculations have shown that for MK-I and MK-II fuels containing 6400 ppm of 'O', the $P_{CO}$ values are too low to cause any significant gas-phase carburisation of the cladding up to clad temperature of 1000 K.

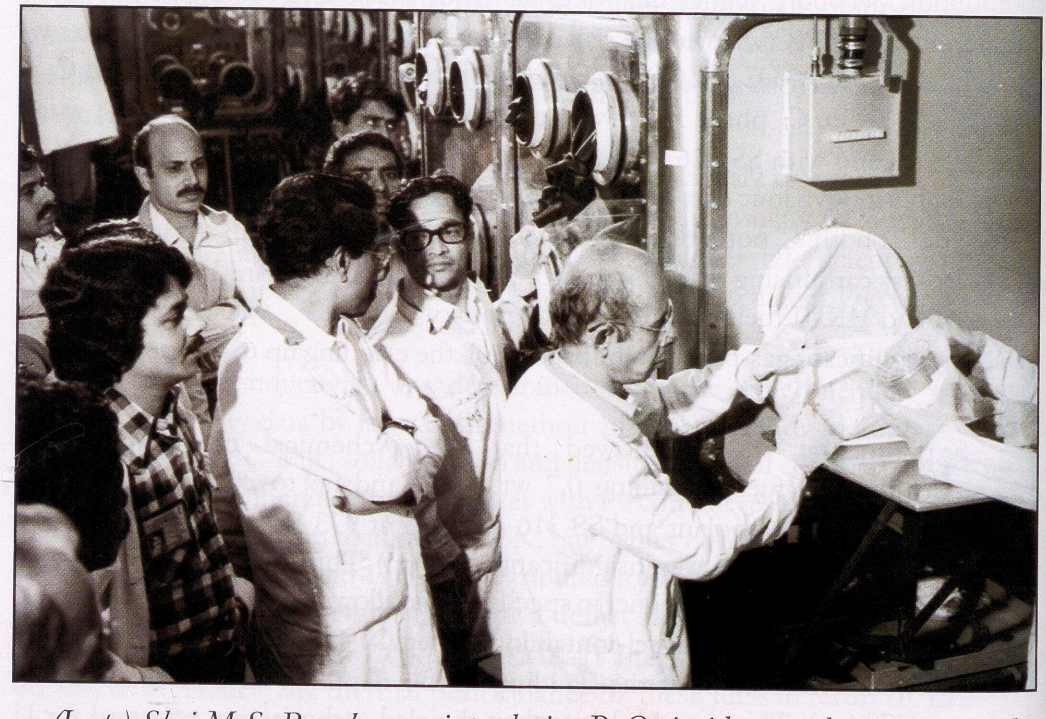
Experimental studies showed that the chemical compatibility of hyperstoichiometric fuel containing 0.7 wt % 'O' and up to 20 wt.% $M_2C_3$ was excellent with sodium coolant and SS-316 cladding at 973 K for 1000 hours. The depth of clad carburisation was insignificant (<12 μm). This observation provided a valuable support to the fuel designer in specifying the allowable 'O' and $M_2C_3$ content. Tests were also carried out for fuel containing higher 'O' (1 wt %) and $M_2C_3$ (60 wt %) which showed carburisation of the cladding up to a depth of 25% of the cladding thickness (370 μm). This is much lower than the clad carburisation reported for U rich mixed carbide containing 15% $M_2C_3$ and less than 0.2 wt % 'O' when 70% of clad

wall thickness was affected. The experimental result tend to confirm the reported theoretical prediction that relatively high 'O' and $M_2C_3$ content could be tolerated in hyperstoichiometric MK-I fuel because of its low 'C' potential activity and $P_{CO}$ even at the operating fuel central line temperature of 1550 K.

The fuel-clad-coolant compatibility experiments were carried out in a two-compartmented- capsules containing fuel pellets inside SS-316 cladding tube in one compartment and sodium in the other. The capsules were heated at 883 K for 2000 hours. Subsequently, sodium was leached out from the pellet and the cladding material. The clad and the fuel were observed to be in good physical condition. Metallographic examination revealed very narrow reaction layer near clad-coolant interface.

## Fabrication and Quality Control

Sufficient experimental back-up gave the confidence to start the fuel production in RMD, BARC. The entire operation from powder processing to pin making has to be carried out in a train of leak tight glove boxes under high purity nitrogen atmosphere, as the carbide is extremely pyrophoric and susceptible to hydrolysis.



*(Late) Shri M.S. Ramkumar introducing $PuO_2$ inside a newly commissioned train of glove boxes meant for FBTR fuel production*

The glove box atmosphere has to be continuously monitored for oxygen and moisture. Accordingly, a process flow sheet for the fabrication of mixed carbide fuel pellets was developed and it is now well established. This involved preparation of carbide clinkers by vacuum carbothermic reduction of oxide-graphite mixture followed by processing of crushed/milled clinkers into sintered pellets by powder metallurgical route. The vacuum and temperature for carbothermic reduction are optimised in order to minimise plutonium losses by vaporisation and also to have oxygen, nitrogen, carbon and $M_2C_3$ content of the fuel within the specified range. Sintered pellets are analysed for U, Pu, O, N, C, $M_2C_3$ & common impurities and subjected to thorough visual, dimensional and linear mass check for acceptance. Fuel pellets which are not conforming to chemical as well as physical acceptance are recycled either via controlled oxidation route or by direct processing (i.e. crushing/milling). MC pellets accepted after chemical and physical QC procedures are stacked to length and encapsulated into SS-316M clad tubes along with other components, including UC insulation pellets. Fuel pins are decontaminated and then subjected to stringent quality checks before wire wrapping. This is followed by numbering of fuel pins for identification before dispatch. Fuel pins are loaded inside specially designed 'Bird cages' and despatched to IGCAR in SNM transport carrier.



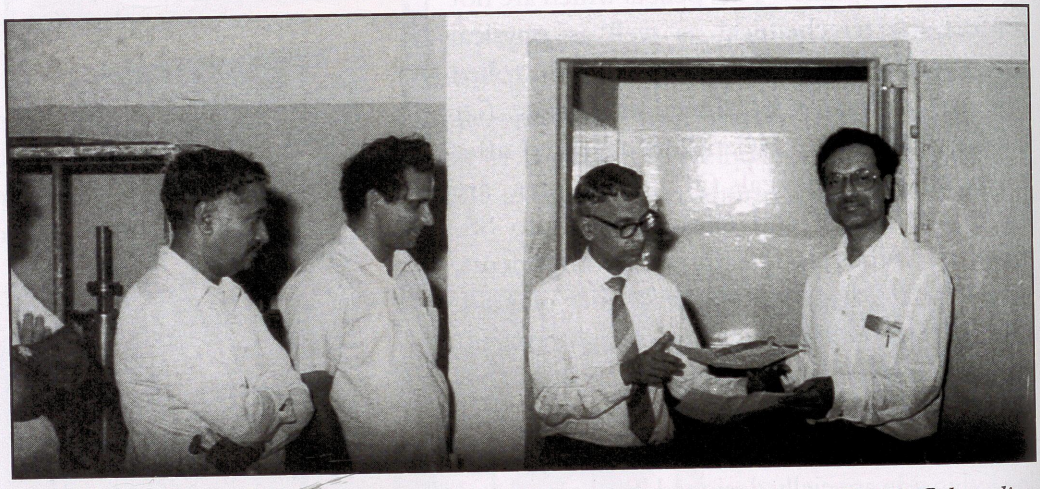
*'Bird cage' for fuel pin transportation*

Fabrication of MC fuel for FBTR manifested in the plant scale production of plutonium based fuel in the country.

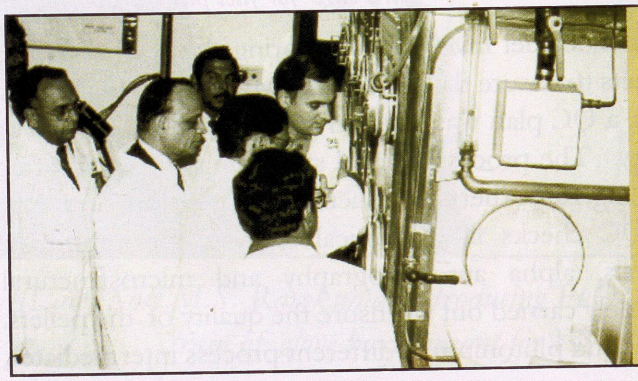
The QC of the mixed carbide fuel involved monitoring the feed material, intermediate and finished products to ensure the fuel quality within the precincts of the specified limits. Accordingly, a QC plan was evolved incorporating the process control steps wherever necessary. The process control steps are carried out after major processes, like powder mixing, carbothermic reduction, compaction, sintering, etc. The major process and QC checks like chemical analysis for impurities, composition and phase analysis, alpha autoradiography and microstructural characterisation of fuel are routinely carried out to ensure the quality of the pellets. For the rapid analysis of uranium and plutonium in different process intermediates,

X-ray fluorescence method was employed. The feedback from process control data is needed to control the process by applying necessary correction in the fabrication parameters. Lack of previous experience in the country about the QC of such fuel has necessitated the development of various destructive and non-destructive techniques, specifically for the evaluation of the fuel pellets and fuel pins.

Large number of fuel pins with MK-I and MK-II compositions have been fabricated successfully on campaign basis and delivered to FBTR. Based on the experience gained over a period of time, certain modifications in the fuel specifications were also introduced. An additional fuel fabrication facility has been commissioned incorporating novel fabrication methods, automated laser based fuel pellet inspection system and laser welding for the end-cap closure.



*(Late) Dr. P.R. Roy, the then Head, Radiometallurgy Division, BARC handing over the documents of the first consignment of FBTR fuel pins to (late) Shri C.V. Sundaram, Director, RRC at the FBTR site, Kalpakkam. (Dr. C.K. Mathews & Dr. G. R. Balasubramaniam are also seen)*



*Former* ***Prime Minister (late) Shri Rajiv Gandhi*** *keenly observing a Semi Automated FBTR fuel pellet inspection machine inside a glove box*

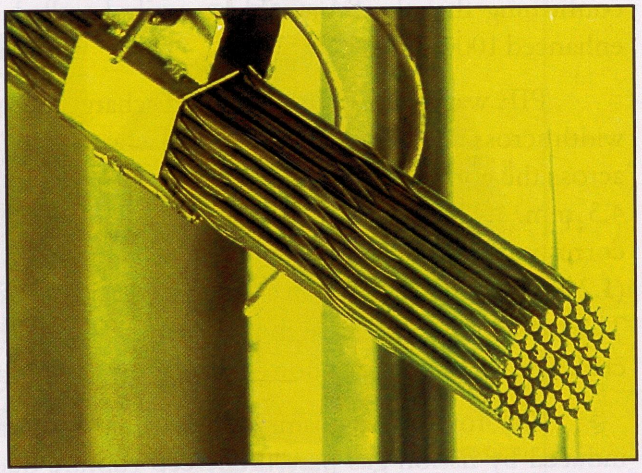
## Design Parameters

The usual criterion adopted in fuel design is to avoid melting of fuel at the centre of the pellet so that the consequent compaction of core leading to increase in reactivity could be prevented. The melt heat rating is estimated using the melting point of the fuel, thermal conductivity, temperature gradient across the fuel-clad gap and the coolant inlet temperature. The design safety limit is worked out from the melt heat rating by providing adequate margin for uncertainties in fuel physical properties and fuel clad gap. The allowable peak LHR is then defined by providing adequate operating margin for over-power trip and inaccuracies in the trip settings. An operating limit of 250 W/cm LHR was initially recommended in 1982 based on conservative estimates of fuel properties and hot spot factors. The attainable burn-up at this LHR was set conservatively at 25 GWd/t. This being a unique fuel without any available in-pile irradiation data, the reactor itself served as an irradiation facility for ascertaining the fuel performance in terms of LHR and burn-up.

## Evolution of LHR

In order to have a more realistic assessment of the allowable LHR of MK-I fuel, out-of-pile tests were carried out at CG. The fuel pellets were heated by electrical resistance method up to 620 W/cm in a specially designed set-up. At 313 W/cm, the pellet developed circumferential cracks and started touching the clad. The fuel-clad gap was fully closed in a few hours at 390 W/cm. However, there was no sign of fuel melting even at 620 W/cm. These results gave the confidence of operating this fuel safely at 320 W/cm, as against the original estimate of 250 W/cm. Hence, the LHR of the fuel was raised to 320 W/cm in 1993.

The central SA was discharged in 1996 at a burn-up of 25 GWd/t. The PIE indicated the clad diameter to be within the original tolerance limits and there was no clad deformation. Metallographic examination indicated near closure of the fuel-clad gap by fuel cracking. Fission gas release was <1%. Two opposing effects come into play as regards heat transfer from pellet to clad during the irradiation of the



*Fuel Pin Bundle inside the hot cells after sodium cleaning for PIE*

fuel. Irradiation induced swelling closes the gap between clad & pellet and thus, aids heat transfer. On the other hand, the release of fission gases from the pellet dilutes the helium bond gas between clad & pellet and reduces the thermal conductivity. Based on the above observations, it was estimated that the operating LHR of the MK-I fuel could be safely increased to 400 W/cm beyond a burn-up of 35 GWd/t because of the beneficial effect of closure of the gap and lesser fission gas release from the pellet at this burn-up.

### Progressive Enhancement of the burn-up limit

PIE at 25 GWd/t revealed the average fuel swelling rate as 1.2 % per atom percent burn-up. The fuel swelling rate is similar to the values reported in literature for low Pu carbide fuels (20% PuC). The earlier limit on burn-up was based on higher swelling rates assumed for Pu rich fuel. A reassessment of the burn-up limit led to an upward revision of the attainable burn-up to 64 GWd/t.

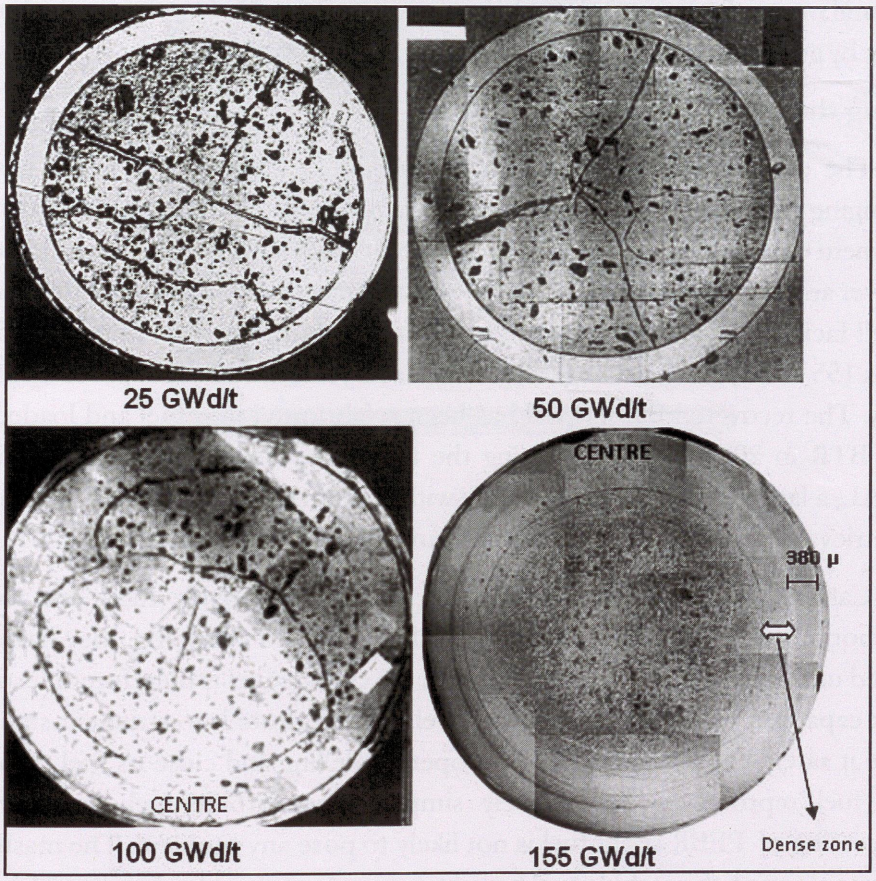
Subsequently, one SA was discharged at a burn-up of 50 GWd/t in 2000. PIE indicated no sign of fuel-clad mechanical interaction. The fuel-clad gap was about 20-30 microns, which could accommodate further swelling of fuel before Fuel-Clad Mechanical Interaction could start. No restructuring of the fuel was noticed. The micro structure indicated that the swelling is accommodated by filling up of porosity. The radial cracks in the fuel pellet were still present. The linear fuel swelling rate was estimated as 1.5%. There was no significant increase in the hardness of the clad. It also validated the swelling laws taken by the designer based on PIE results at 25GWd/t and reiterated the fact that LHR can be safely increased to 400W/cm. There was no measurable dilation of the hexagonal wrapper. Six SA in the first ring were lifted by 200 mm and lowered back and the lifting forces were normal, confirming that the SA could be easily handled. Hence, the target burn-up was enhanced 100 GWd/t.

PIE was carried out on a SA discharged at a burn-up of 101.5 GWd/t. The width across the flats of the hexagonal wrapper had increased by 0.34 mm (0.7%) and across the corners by 0.23 mm (0.5%). Head-to-foot misalignment was found to be 4.3 mm. The fuel pins had elongated by 0.6-2.0 mm, the maximum increase corresponding to 0.4%. Diameter of the fuel pins had increased by 0.06–0.08 mm (1.2-1.6%). The increase in stack length varied from 3.23-8.35 mm (1.01%-2.61%). The maximum gas release measured was 14%. There was no carburisation of the clad. The uniform elongation of the clad at 430°C was 3%.

The four mechanisms which could limit the burn-up are: creep rupture induced by fission gas pressure, creep rupture due to FCMI, fuel matrix porosity exhaustion and dilation of hexagonal wrapper. Designers estimated that the limits

based on clad rupture due to fission gas pressure and FCMI are high. The burn-up at which the available porosities could be fully exhausted was estimated to be between 107.7 and 129.5 GWd/t and the behaviour of the fuel after exhaustion of the porosities is not known. The burn-up limit corresponding to wrapper dilation equal to the minimum available inter-subassembly gap was estimated as 155 GWd/t. Beyond this, the inter-wrapper pressure could lead to difficulties in fuel handling. The force required to discharge the SA was measured at the end of each campaign. The results showed an increasing trend with burn-up, but the peak load has always been below the high tension trip setting of the gripper hoist mechanism of the discharging flask. So far, 16 MK-I SA (976 pins) have reached such a high burn-up without any clad failure.

PIE was carried out on a SA at 155 GWd/t. Metallographic examination of the fuel-clad cross section at the centre of the fuel column revealed a distinct zone devoid




*Micrographs of MK-I Fuel at Different Burn-up levels*

of fuel porosities near the outer diameter of the fuel. The porosity exhaustion is attributed to the creep of the fuel due to FCMI. Complete closure of the fuel-clad gap and circumferential cracking at the centre as well as the end of the fuel column indicated that the entire fuel column has entered the restrained swelling regime. The dimensional measurements on the hexagonal wrapper indicated a maximum dilation of 0.65 mm at the centre of the fuel column region. Considering a nominal gap of 0.75 mm originally available between the fuel SA, the present gap available is about 0.1 mm. Though the clad and wrapper ductility is considered adequate from the handling point-of-view, it was decided to set this burn-up as the operating limit of MK-I fuel. One SA was taken to 165 GWd/t and it also endured this burn-up without any failure. However, as a safe measure, it has been decided to set the allowable burn-up to 155 GWd/t.

The limitation on burn-up arises not so much from the fuel per-se as from the structural materials of the fuel SA. It will hence, be possible to enhance the burn-up further by going in for more swelling resistant clad and wrapper.

## Closing the fuel cycle

The closing of the fuel cycle with carbide fuel that is rich in Pu is a very challenging task. R&D on process development on flow sheet, dissolution and the equipment development on laser dismantling of SA, single pin chopper, electrolytic dissolver and centrifugal extractor led to the construction and commissioning of the hot cell facility, CORAL. Spent fuel with different burn-up from as low as 25 to as high as 155 GWd/t and cooling times as low as 2 years have been reprocessed in this facility. The recovered Pu material has been refabricated into fuel and loaded back into FBTR in 2009, formally closing the fuel cycle. Based on the experience of CORAL, a larger demonstration facility with sodium decontamination, dismantling and retrieval of pins is under construction and will become operational by 2011-12.

Carbide fuels are more difficult to handle than oxide fuels due to their pyrophoric nature. At all stages of fabrication, PIE and chopping, they have to be handled under inert cover. Experience with the carbide fuel is therefore, a testimony to our capability to design ab-initio a novel fuel, fabricate it to stringent standards, handle it safely, validate it by PIE & proper modelling and close its fuel cycle. The oxide fuel reprocessing is relatively simpler than carbide fuel and therefore, reprocessing of PFBR spent fuel is not likely to pose any problem. The mastery of the technology of the carbide fuel has put us in a good stead for handling the power reactor MOX fuels.

Chapter - 10

# Regulatory Experience


Atomic Energy Establishments in India
SRINAGAR (Jammu & Kashmir)
Nuclear Research Laboratory(BARC)
GULMARG (Jammu & Kashmir)
High Altitude Research Laboratory (BARC)
NARORA (Uttar Pradesh)
Narora Atomic Power Station (NPCIL)
TURAMDIH (Jharkhand)
Uranium Mine (UCIL)
KOLKATA (West Bengal)
*Variable Energy Cyclotron Centre
*Regional Radiation Medicine Centre (VECC)
*Radio-pharmaceutical Laboratory (BRIT)
*Saha Institute of Nuclear Physics
AHMEDABAD (Gujarat)
Institute for Plasma Research
KASAN (Delhi)
Seismic Array Station (BARC)
DELHI
Radio-pharmaceutical Laboratories (BRIT)
BANDUHURANG (Jharkhand)
Uranium Mine (UCIL)
BARODA (Gujarat)
Heavy Water Plant (HWB)
MOUNT ABU (Rajasthan)
Gurushikher Observatory for Astrophysical Sciences (BARC)
DIBRUGARH (Assam)
Radio-Immunoassay Centre (BRIT)
KAKRAPAR (Gujarat)
Kakrapar Atomic Power Station (NPCIL)
ALLAHABAD (Uttar Pradesh)
Harish-Chandra Research Institute
RAWATBHATTA (Rajasthan)
*Rajasthan Atomic Power Station 1-4 (NPCIL)
*Rajasthan Atomic Power Project 5&6 (NPCIL)
*Heavy Water Plant-Kota (HWB)
*RAPPCOF-Kota (BRIT)
NARWAPAHAR (Jharkhand)
Uranium Mine (UCIL)
JADUGUDA/BHATIN (Jharkhand)
*Uranium Corporation of India Limited (HQ)
*Uranium Mill & Mines (UCIL)
INDORE (Madhya Pradesh)
Raja Ramanna Centre for Advanced Technology
INDUS- 1&2
TARAPUR (Maharashtra)
*Tarapur Atomic Power Station (NPCIL)
*Tarapur Atomic Power Project 3&4 (NPCIL)
*WIP / SSSF (BARC)
*Power Reactor Fuel Reprocessing Plant (BARC)
*Advanced Fuel Fabrication Facility(BARC)
*General Services Organisation
HAZIRA (Gujarat)
Heavy Water Plant (HWB)
TALCHER (Orissa)
Heavy Water Plant (HWB)
D2EHPA Plant
BHUBANESWAR (Orissa)
Institute of Physics
NASIK (Maharashtra)
KRUSHAK (BARC)
CHHATRAPUR (Orissa)
Orissa Sand Complex /
Thorium Plant (IRE)
THAL (Maharashtra)
Heavy Water Plant (HWB)
MANUGURU (Andhra Pradesh)
Heavy Water Plant (HWB)
MUMBAI (Maharashtra)
*Nuclear Power Corporation of India Ltd.(HQ)
*Indian Rare Earths Ltd. (HQ)
*Heavy Water Board (HQ)
*ISOMED Plant (BRIT)
*Bhabha Atomic Research Centre
*Radiation Medicine Centre (BARC)
*BRNS / HBNI / NBHM (HQ)
*TIFR / TMC / AEES (HQ)
*DPS / DCS&EM (HQ)
HYDERABAD (Andhra Pradesh)
*Electronics Corporation of India Ltd.(HQ)
*Nuclear Fuel Complex (HQ)
*Jonaki Laboratory (BRIT)
*Centre for Compositional Characterisation of Materials (BARC)
*Atomic Minerals Directorate for Exploration & Research (HQ)
GAURIBIDANUR (Karnataka)
Seismic Station(BARC)
BANGALORE (Karnataka)
Radio-pharmaceutical Laboratories (BRIT)
NAVI MUMBAI (Maharashtra)
*Board of Radiation & Isotope Technology (HQ)
*Radio-pharmaceutical Laboratories (BRIT)
*Radiation Processing Plant (BRIT)
*Labelled Compounds Laboratory (BRIT)
* Radiation Equipment Production Facility (BRIT)
*Beryllium Plant (BARC)
*Electron Beam Centre (BARC)
*ACTREC (TMC)
MYSORE (Karnataka)
Rare Materials Project (BARC)
CHENNAI (Tamil Nadu)
*Institute of Mathematical Sciences
KALPAKKAM (Tamil Nadu)
*Madras Atomic Power Station (NPCIL)
*Bharatiya Nabhikiya Vidyut Nigam Ltd.(HQ)
*Indira Gandhi Centre for Atomic Research
*PFBR Project (IGCAR)
*Nuclear Desalination Demonstration Plant (BARC)
*Kalpakkam Fuel Reprocessing Plant (BARC)
*General Services Organisation
KAIGA (Karnataka)
*Kaiga Generating Station 1&2 (NPCIL)
*Kaiga Atomic Power Project 3&4 (NPCIL)
TUTICORIN (Tamil Nadu)
Heavy Water Plant (HWB)
UDYOGMANDAL (ALWAYE) (Kerala)
Rare Earths Plant (IRE)
PALAYAKAYAL (Tamil Nadu)
New Zirconium Sponge Plant (NFC)
CHAVARA (Kerala)
Minerals Separation Plant (IRE)
KUDANKULAM (Tamil Nadu)
Kudankulam Atomic Power Project (NPCIL)
KOLLAM (Kerala)
Low Radiation Research Laboratory (BARC)
MANAVALAKURUCHI (Tamil Nadu)
Minerals Separation Plant (IRE)
BRNS : Board of Research in Nuclear Sciences
HBNI : Homi Bhabha National Institute
NBHM : National Board for Higher Mathematics
SSSF : Solid Storage Surveillance Facility
WIP : Waste Immobilisation Plant
AEES : Atomic Energy Education Society
ACTREC : Advanced Centre for Treatment, Research & Education in Cancer
TIFR : Tata Institute of FundamentalResearch
TMC : Tata Memorial Centre
DCS&EM : Directorate of Construction, Services & Estate Management
DPS : Directorate of Purchase & Stores
Research & Development Organisations
Public Sector Undertakings
Industrial Facilities
Grant-in-aid Organisations
Service Organisations

# Regulatory Experience

## Regulatory Experience

FBTR project was started well before the creation of the Atomic Energy Regulatory Board. Land had been acquired by DAE, north and south of MAPP, beyond the 1.6 km exclusion zone. The availability of land, low population density, assurance of water supply, low seismicity and proximity to MAPP which would facilitate sharing of infra-structural facilities and exchange of knowledge base were the major reasons for the choice of Kalpakkam for RRC. In those pre-AERB days there were no formal mechanisms for safety clearance of nuclear projects. Only site clearance was required from DAE-Site Selection Committee. With MAPP already cleared at Kalpakkam, there was no difficulty in getting the Kalpakkam site cleared for FBTR. Since cooling water requirement was not very large, the terminal sink was a Cooling Tower. Hence, unlike MAPS, it was not required to locate the reactor close to the sea-line. The highest spot within the campus of RRC was chosen for FBTR. The site was at a distance of about one km from the sea. Site clearance was accorded by the Site Selection Committee of the DAE.

Rapsodie was located in a highly seismic zone, and was designed for high seismicity (Mercali scale of 10). At the time of construction of FBTR, Kalpakkam site was falling under seismic zone no. 1 as per IS 1893 and hence, considered to be located in an area free of seismic events. However, during April-May, 1966, a few tremors were noticed in Tambaram, about 30 km from site. All these appeared to be only between 3.9 and 5 on Richter-Gutenberg scale of magnitude. This area is hence considered well outside the active seismic zone. However, considering the importance of reactor installation and the fact that recurrences of tremor cannot be ruled out completely in future, civil structures were designed as per the Indian Standards using appropriate seismic coefficients.

### Initial Safety Review

FBTR was started as a self-regulated project. The designers and project personnel had undergone training at Rapsodie. The project also had the benefit of the agreement between India and France, by which, French engineers were available for consultation. Realising the need for independent review of the design and safety aspects of FBTR, a Safety Evaluation Working Group was constituted by RRC. This committee reviewed in detail all the design and safety aspects of the reactor, especially where changes from Rapsodie design were involved.

### *Regulatory Experience*

With the start of commissioning activities, the RRC-CWMF Safety Committee (later became IGCAR Safety Committee or IGCAR-SC) was constituted and held its first sitting on January 30, 1982. The first FBTR item discussed by the committee was in its fourth sitting on November 30, 1982, when the safety report on purification of commercial grade sodium to reactor grade sodium in the purification loop at Engineering Hall was reviewed. From then on, all the commissioning aspects, including sodium transportation to FBTR, were meticulously reviewed by the RRC-CWMF Safety Committee. The recommendations of the Safety Committee were put up for approval to the DAE-Safety Review Committee (DAE-SRC).

The 65 SA MOX core design of Rapsodie was used for the Preliminary Safety Analysis Report of FBTR prepared in 1982 and submitted to the Safety Committee and DAE-SRC. With the shift from MOX fuel to the carbide core, the concerned chapters of the PSAR were modified for the new core and issued as an addendum to the PSAR and the safety clearance was sought.

The shift from mixed oxide to mixed carbide fuel led to major discussions within the safety committees. The Pu-rich mixed carbide driver fuel selected for FBTR was essentially an untested fuel designed conservatively using results from out-of-pile testing and the published international experience with mixed carbide fuel of low Pu content. In addition, the neutronic parameters of the core with Pu rich fuel differed substantially from that with the mixed oxide fuel. DAE-SRC, understanding the difficulties in getting enriched uranium, was considerate in clearing this untested driver fuel for FBTR, based on out-of-pile data and the conservatism in design. The Technical Specifications document for 10.6 MWt was prepared for FBTR based on the similar documents available for RAPS (Rajasthan Atomic Power Station) and MAPS. It was reviewed and approved by DAE-SRC after several extended meetings at Kalpakkam.

### AERB Regulatory Control

The Atomic Energy Act was enacted by the Indian Parliament in 1948 and upgraded in 1962. In pursuance of this Act, the AERB was set up in 1983 to lay down nuclear safety standards and regulations for radiation safety in the whole country and for industrial safety within DAE and to ensure compliance. AERB is independent of DAE, being directly responsible to the Atomic Energy Commission (AEC). DAE-SRC was absorbed in AERB and became the Safety Review Committee for Operating Plants (SARCOP) and is entrusted with the task of reviewing the safety performance of operating plants and to recommend modifications or approve continuance of operations.

Operating plants like FBTR are subjected to a three-tier review. The first is by the Station Operations Review Committee (SORC). The first meeting of SORC was

held on March 14, 1986. To date, SORC has met 380 times- i.e. on an average more than once a month. Safety related issues are subjected to the second-tier review by the IGCAR-SC constituted by SARCOP. Deliberations and recommendations of this committee on the plant operations status and with a bearing on safety are brought to the notice of plant management, Director, IGCAR and SARCOP to ensure improvement in the safety status and compliance. As of date, the IGCAR-SC has held 260 meetings (each of 1 to 2 days) at an average of 9 meetings per year. Most IGCAR-SC meetings have at least one agenda item pertaining to FBTR. Deliberations in IGCAR-SC include regular briefing on FBTR status, review of core configuration changes for every campaign, safety clearance for each campaign, major test programmes including procedures and test results, modifications in safety critical and safety related systems, discussion of Safety Related Unusual Occurrences/Significant Events, planned violation of Technical Specifications, etc.

The third-tier review is by SARCOP. Usually SARCOP deals with safety significant events and modifications in safety critical and safety related systems. FBTR still does not have a fixed core and has not reached the equilibrium core envisaged earlier. One major consequence of this is that the reactor power and physics characteristics keep changing with every fuel handling campaign. Limiting Safety System Settings defined in the Technical Specification document need modification for every campaign. Hence, IGCAR-SC and SARCOP clearances are obtained on campaign to campaign basis. Also, stage-wise clearances for the enhancement of the allowable burn-up of the MK-I fuel were accorded by SARCOP, based on PIE studies.

## Radiation Safety Practice and Experience

The plant health physicist is independent of the plant management and reports to Head, Radiation Safety Section, who is professionally qualified with long years of field experience. The independence of this group is further assured by the reporting of Head, RSS to Director, IGCAR through Head, Radiological Safety Division and Director, Safety Group, independent of the operating plants and facilities.

FBTR operation has been benign in terms of radiation exposure causing minimum concern to the safety authorities. The general radiation level in RCB during operation is 1 μGy/h. On top of the reactor, maximum streaming fields of 15 μGy/h is seen at spots. During the initial power rising, higher radiation levels, if any, noted had been reduced by local shielding augmentation. Since 1986 up to 2009, the cumulative collective dose was ~149.03 person mSv. There has been no release of $I^{131}$, $Sr^{90}$ and other particulates so far due to reactor operation.

## Incidents

There is a well-established procedure for reviewing any incident in FBTR. First Information Report (FIR) is issued by the shift in-charge and, if it is of safety significance, is reviewed immediately in SORC and also intimated to the higher safety committees (IGCAR-SC and SARCOP). After detailed investigation and proposing remedial measures to prevent recurrence, Significant Event Report (SER) is filed for review by the committees. The root cause of the incident is identified and remedial action prescribed to prevent recurrence is implemented after due safety review.

There were several incidents in the last 25 years, after first criticality, which required investigations, root cause analyses and safety review before repairs or modifications, if any, and resuming operations. It would be difficult to narrate here all the events along with their safety significance and the final resolutions. Most of the events are below scale (incidents of no safety significance) in the International Nuclear Event Scale reporting. A few events fall under INES scale-1 (anomaly). One event in 1994 (SG got isolated on water side and LOR did not take place on low feed water flow) was rated as scale-2 (incident).

## Modifications in Safety Systems

During the design, safety review, commissioning and initial operations of FBTR, a conservative and safe approach was adopted which led to the introduction of a number of automatic safety actions and trips. This was justified since FBTR is a prototype first sodium cooled reactor in the country with unique Pu-rich mixed carbide fuel and operating with a fast neutron spectrum. However, based on the actual operating experience over the years, many of the automatic safety actions were found to be redundant. To improve the plant availability, the safety actions were analysed in depth taking into account the operating experience and proposals were put up to IGCAR-SC and SARCOP for their modification or deletion.

There are several instances to show the extreme caution with which the safety issues of a nascent technology were initially addressed, both by the operators and regulators, and the open-mindedness and pragmatic outlook to review them based on actual experience feedback. For example, with the 3 second interlock imposed by SEWG and DAE-SRC, the CRDM motor was getting heated up excessively due to frequent start-ups. More important, the probability of uncontrolled withdrawal of control rod increases considerably due to possible welding of raising contactors. Also, in the steaming phase of the steam generator, the steam water system was operating in the two phase regime for quite some time, resulting in the erosion of internals of the control valves. A proposal was hence put up later to delete this interlock. It was also proposed to manually inhibit the reactivity trip during power

rising and to raise the control rod level discordance LOR limit to 40 mm from 20 mm. A SARCOP member personally participated in the experiments conducted for clearing these proposals and provided valuable suggestions on the logic modifications.

## Licensing & Qualification

Although an experimental reactor, FBTR has all the complexities of a nuclear power plant. A requisite complement of staff must be available in each shift on round-the-clock basis to carry out the functions effectively. The shift-staff consists of two engineers, four supervisors, nine operators, one health physicist, one chemist and three maintainers. Whereas qualification requirement is a must for all, the licence for plant operation is mandatory for shift engineers and supervisors only.

The licensing procedure for shift personnel includes induction training on basic engineering principles, study of system manuals & equipment catalogues and on-the job training in the field. Certification of adequate job knowledge and experience in the relevant field is done through check-lists, written examination, walk-through test and final interview with the Licensing Committee. All licensed operating personnel require renewal of licence every three years. The licensing committee for Level III operating personnel & above and also for Control Room supervisors/assistants is constituted by AERB and for others by Director, IGCAR.

## Periodic Safety Review

Within the period of operating license of a nuclear power plant, AERB grants authorisation for operation for specified periods. For renewal of authorisation after the specified period, a comprehensive safety review is required considering cumulative effects of plant ageing, irradiation damage, results of in-service inspection, system modifications, operating experience, status and performance of safety & safety support systems, revisions to applicable safety standards, etc. The first periodic safety review of FBTR is in progress. After the preliminary review of the documents by AERB, the plant has revised most of the chapters and submitted. The Technical Specifications for reactor operation, which is the rule book for the plant, has been revised twice. The third revision is currently under regulatory review.

## Safety Culture in FBTR

FBTR has inherited the self-regulating culture followed during the project phase. The management of IGCAR encourages transparency in reporting, reflecting the scientific temper ingrained in the Centre. In fact, many safety issues raised by SC & SARCOP are based on station monthly reports and SORC minutes rather than from the formal incident reporting system.

### Regulatory Experience

With the designers and the experimentalists sitting just outside the reactor, FBTR enjoys the benefit of the analytical prowess of the design group and the resources of the Reactor Engineering Halls and other facilities in IGCAR, in addition to the resources of BARC.

FBTR has also benefited greatly by the wisdom and experience of the regulatory bodies. The fuel handling incident of 1987 was a setback for FBTR, but the then chairman of IGCAR-CWMF SC, himself played a leading role in all the activities related to normalising the reactor, personally guiding the several teams which were working towards investigation, analysis and restoration. After the incident of dropping of Capsule Transfer Gripper in 1989, SARCOP stipulated the formation of a separate in-plant committee to review all pile handling procedures. This has been a rewarding stipulation for the plant, as all handling operations are done with well-written procedures in the form of check-lists, and all handling operations on the reactor since then have been incident free. Many of the precautions recommended by the review committee have percolated to major handling operations even outside the pile, and have become part of the personnel's psyche.

Members of the SARCOP standing committee on Reactor Physics participated in the experiments to verify the self-limiting and reversible nature of the reactivity transients observed during the Zr-Nb irradiation at 8 MWt in 1998-99, for according clearance for reactor operation. Even recently, when there was an incident of violation of shutdown margin, the members of this committee reviewed it in a consultative mode of understanding and resolving the problem. SARCOP and SC also suggested a few experiments.

*Shakespeare, while talking of the quality of mercy, says 'it blesseth him that gives and him that takes'. So is the regulatory process for FBTR, India's first fast reactor. While IGCAR in general and FBTR in particular have benefited from the interaction with the regulators, the regulators have also benefited by regulating FBTR.*

### Symbiosis

The operating experience feedback from FBTR has provided AERB the knowledge base required to take on the challenges of regulating future breeder reactors. Some of the senior officers in AERB had their earlier stint in FBTR. The rich experience and knowledge of these officers has been a supplement to the fast reactor knowledge base of AERB. The seismic re-evaluation of FBTR to current standards has been taken up as a joint research project between IGCAR & Safety Research Institute of AERB. This exercise is likely to set the regulatory standard for seismic re-evaluation of old plants in India.

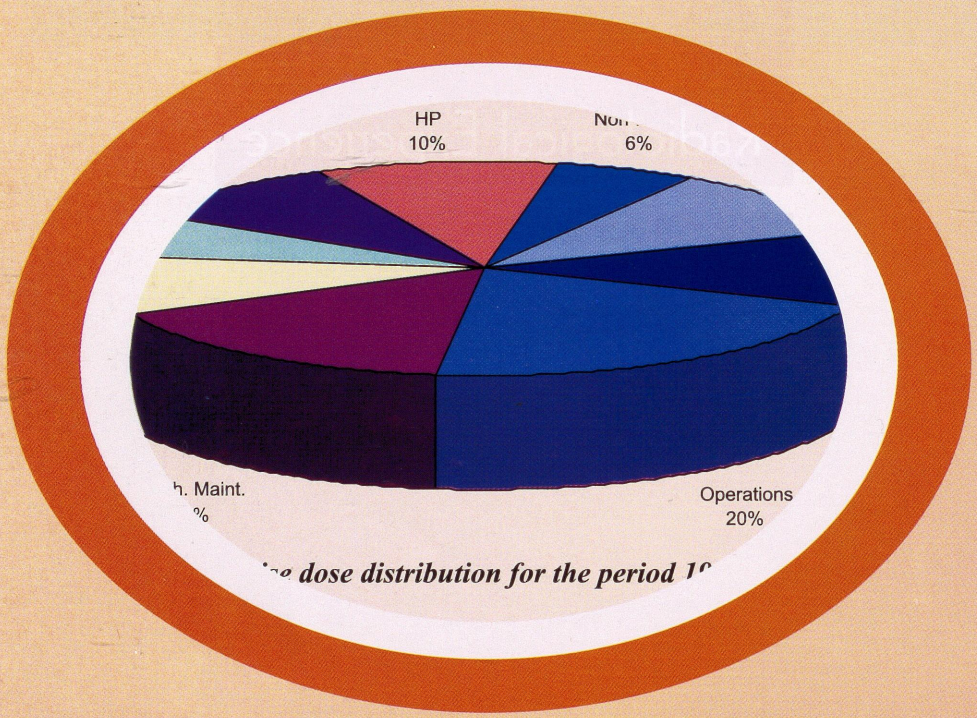
| Major Events reviewed by SC & SARCOP | |
|---|---|
| **Year** | **Event** |
| 1984 | 2.5 litres of NaK leaked from secondary sodium cold trap |
| 1985 | Control Rod partial drop incident |
| 1986 | CR rising even after release of raise command button |
| 1987 | Fuel handling incident |
| 1987 | 1 litre Na leaked from secondary circuit while sampling |
| 1988 | 2 litres of NaK leaked from NaK bubbler of argon purification circuit |
| 1989 | Dropping of capsule transfer gripper |
| 1989 | Core Temperature Supervision not operational for 15 minutes during reactor operation |
| 1990 | Non-availability of DG sets for 1.25 hour during offsite power failure due to draining of diesel from day tank (Station Black-out) |
| 1990 | Ingress of moisture into the internal circuit of Preheating & Emergency Cooling System |
| 1991 | Sodium leak from the bellows seal valve of East SGLDS circuit |
| 1992 | Station black-out for 15 minutes |
| 1992 | MBFP seizure |
| 1993 | Reactor operation for 2.25 hours without core temperature supervision and Fine Impulse Test |
| 1993 | Non-availability of both secondary sodium pumps |
| 1993 | Feed water leak from SG inlet sub-header |
| 1993 | Feed water leak from SG orifices |
| 1994 | Failure of LOR on Ward Leonard trip |
| 1994 | Positive reactivity transient |
| 1994 | SG got isolated on water side and LOR did not take place on low feed water flow |
| 1995 | Reactor SCRAM on Positive reactivity transient |
| 1995 | Core Cover Plate Mechanism malfunction (stuck in Fuel handling position) |
| 1995 | Station black-out for 2 minutes |
| 1996 | CCPM permanently stuck in Fuel Handling position |
| 1996 | Inoperability of CRDM#E |
| 1996 | Core Temperature anomalies of MK-II Fuel SA |
| 1998 | Reactivity phenomenon at 8 MWt |
| 1999 | Inoperability of CRDM#E |

## Regulatory Experience

| Year | Event |
|---|---|
| 1998 | Feed water leak from thermowell in SG |
| 2000 | Biological shield cooling system water leak |
| 2001 | Non-availability of offsite power for 36 hours |
| 2002 | 75 kg primary sodium leak into Purification Cabin |
| 2002 | LOR on 'SG water leak to sodium' signal (found spurious) |
| 2003 | CR rising on getting lower command |
| 2004 | Release of pot plug while handling active control rod with secondary flask |
| 2005 | LOR on 'SG water leak to sodium' signal (found spurious) |
| 2005 | Cable fire in East SGLDS cabin |
| 2005 | Leak from Bellows seal valve in secondary west |
| 2006 | Sodium leak in Ni diffuser of SGLDS |
| 2007 | Biological shield cooling system water leak |
| 2007 | Choking of primary sodium hot argon line |
| 2007 | Failure of Expansion Tank Rupture disc |
| 2007 | Inoperability of CRDM#D |
| 2009 | Detection of dislocation of orifice assemblies in SG modules |
| 2009 | Shut down margin extrapolated too low after fuel handling |

Chapter - 11

# Radiological Experience




*dose distribution for the period 1*

# Radiological Experience

## Radiological Experience

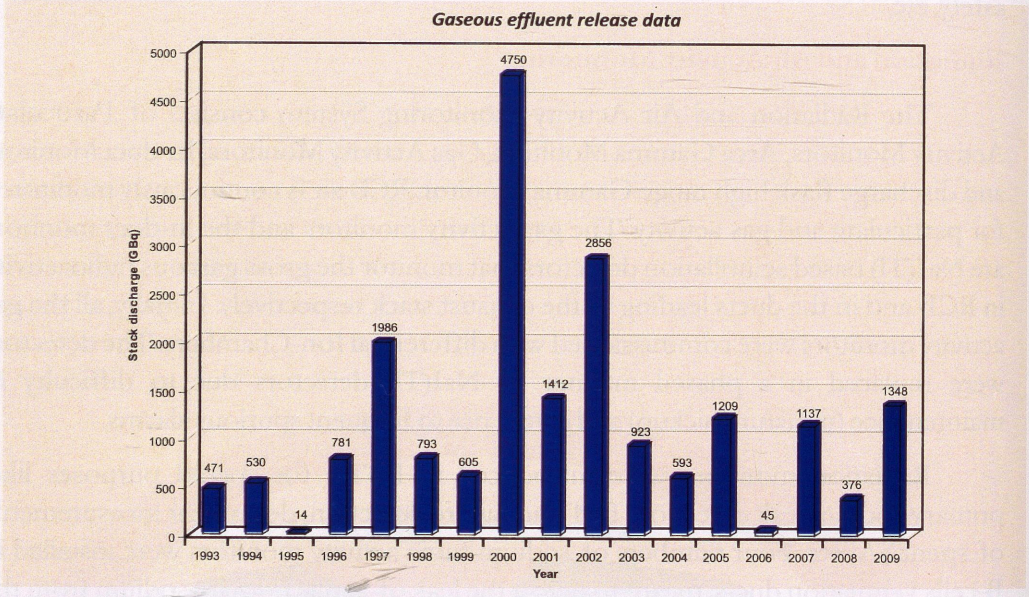
Radiological experience gained during operation of a nuclear reactor is an important safety aspect. The experience includes shielding requirements, monitoring of air activity, personnel monitoring, waste disposal, environmental safety, etc.

### Radiation and Air Activity Monitoring

The Radiation and Air Activity Monitoring System consists of Particulate Activity Monitors, Area Gamma Monitors, Gas Activity Monitors, In-duct Monitors and discharge flask high range Gamma Monitor. RCB air is continuously monitored for particulate and gas activity. The gas activity monitors and the in-duct monitors are NaI(Tl) based scintillation detectors that monitor the gross gaseous radioactivity in RCB and in the ducts leading to the exhaust stack respectively. Initially, all the gas activity monitors were commissioned with differential Ion-Chambers. The detectors were replaced in a phased manner by NaI(Tl) detectors due to difficulty in maintenance (moisture pickup) and giving rise to frequent spurious alarms.

Radiation monitors were introduced in FBTR for special purposes like, primary sodium leak detection, fuel clad failure detection, decay heat measurements of spent SA, etc. Four numbers of sodium fume activity monitors were installed in B-cells ventilation ducts, meant to detect the leak of active primary sodium from the pipelines into the shielded concrete cells. Based on the operating experience, these detectors were relocated in the duct away from B-cells and provided with lead shielding around the detector to reduce the background. The gamma energy threshold was slowly raised from 200 keV to 1 MeV to ensure the detection of only $Na^{24}$ gamma. The increase of threshold energy to 1 MeV helped in reducing the counts in the Compton continuum due to other radionuclides and detecting the intended $Na^{24}$ peaks alone (1.369 MeV and 2.753 MeV) without compromising much on the sensitivity factor. These modifications helped in obtaining better signal-to-noise ratio and also avoiding the spurious alarms. A gas activity monitor ($Na^{24}$ activity monitor) to detect sodium aerosol in the double envelope (filled with nitrogen gas) of RV and a particulate activity monitor for detecting primary sodium leak in purification cabin were introduced. During reactor operation, the increase in activity as registered by the $Na^{24}$ activity monitor could be correlated to $Ar^{41}$ produced from trace amount of $Ar^{40}$ impurity present in the nitrogen.

Reactor cover gas activity is monitored continuously using two gas flow ionisation chambers in CRD system (whose discharge is routed stack) for the early detection of fuel pin failures. Sensitivities of these monitors were estimated by measuring the concentration of $Ar^{41}$ in the cover gas vis-à-vis the monitor reading. In the nineties, CRD system was not in continuous service and also its flow rate was erratic. Of late, CRD system is in continuous service with improved flow control. An optimum flow rate of 20 to 50 cc/s is maintained to avoid the release of $Ne^{23}$ produced by $Na^{23}(n,p)Ne^{23}$ reaction which could otherwise lead to inadvertent higher discharge through stack.



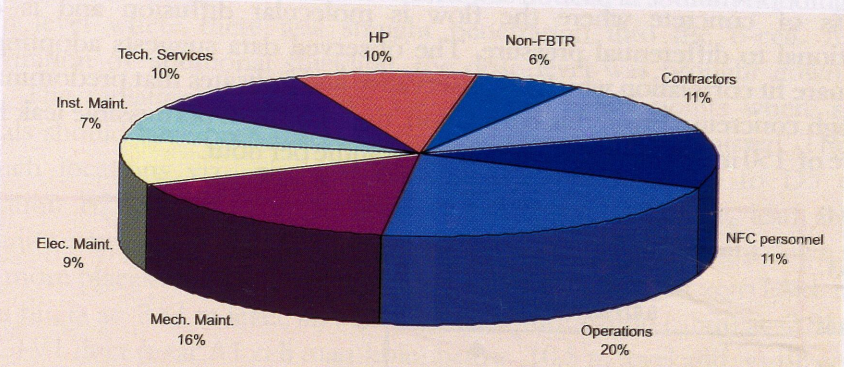

The discharge flask is provided with a high range gamma monitor to prevent any inadvertent removal of irradiated fuel SA with more than 400 W of decay power.

The readings of all the installed radiation monitors are fed to a dedicated industrial PC. The calculation of gaseous effluent discharged through stack on the basis of readings of the exhaust induct monitors of RCB has been incorporated.

## Personnel Monitoring

All occupational workers are provided with personnel monitoring services in the form of Thermo Luminescent Dosimeters. All these workers undergo routine annual whole body monitoring and periodic medical examination as stipulated by Atomic Energy Regulatory Board. There has been no significant case of internal exposure. The man-rem expenditure for FBTR has been consistently low. The exposure data as well as group-wise dose distribution are being recorded. The man-rem expenditure during the normalisation of reactor after the FH incident in 1987

was 26.45 person-mSv. The cumulative collective dose during the period from 1991 to 2009 was 97.37 person-mSv with the maximum of 10.5 person-mSv in 2009. The highest individual dose of 4.25 mSv was received in 2003 during the inspection of sodium service valves located in the primary sodium cells in the aftermath of minor sodium leak incident in purification circuit. The annual average per-capita dose is 0.017 mSv. Major contributors to personnel exposure are due to inspection work in the primary sodium cells, fresh fuel assembly operations and retrieval cum disposal of leaked radioactive sodium from primary purification cabin.



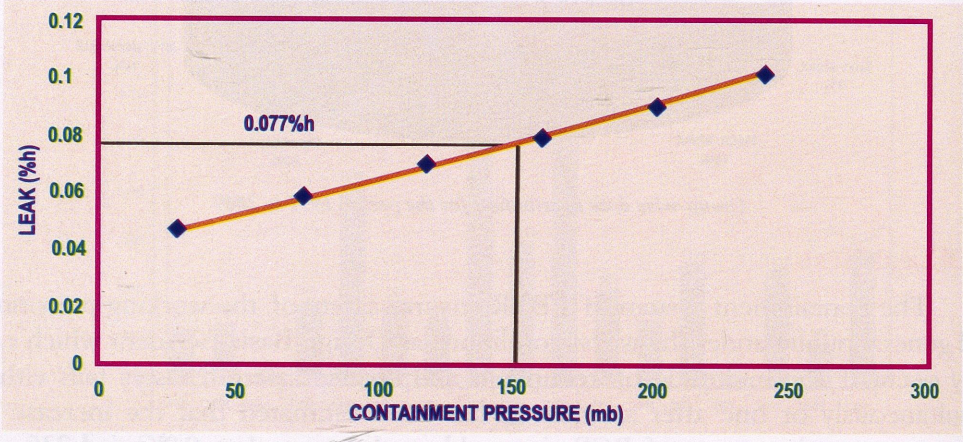

***Group-wise dose distribution for the period 1991 to 2009***

## RCB Leak Test

The containment system of FBTR ensures safety of the working personnel and general public under the worst conditions as Design Basis Accident which can only occur if the multitude of redundant and diverse safety features fails either simultaneously or one after another. It has been estimated that the increase in temperature and pressure of RCB air would not be more than 94°C and 235 mb respectively as a result of any of the three Design Basis Accidents namely TOPA, LOCA and LOFA. RCB is of reinforced concrete construction of 25 m diameter with 900 mm thick containment wall. The base raft is 14.5 m below ground level and the dome is 18.5 m above. Below the ground level, the containment building is divided into various shielded cells to house reactor vessel, primary sodium systems and their auxiliary systems. Apart from pipe line and cable penetrations, there are five major penetrations namely material airlock, personnel airlock, supply duct, exhaust duct and effluent duct.

Integrated leakage rate test of RCB is required to be carried out biennially as per ANS 56-B, 1981 standard using 'Absolute method'. This method stipulates the determination of leakage over 24 hours by means of measurement of direct pressure, temperature and humidity during the test and the application of gas laws. So far containment integrity tests have been conducted 13 times during the course of 25 years with very good results. After completing RCB construction works, the first proof test was conducted in September, 1983 at the design pressure of 235 mb and

the estimated leakage rate was 0.054% volume per hour as against the specified leakage rate of <0.1% volume per hour. The normal practice being not to subject the building to design pressure repeatedly, it was decided to carry out periodic tests at a lower pressure of 150 mb. To establish a correlation between leak rate and containment pressure, a series of tests were conducted at various pressures in March, 1987. The two main leakage paths were through pipe/cable penetrations (about 56/300 Nos respectively) of RCB and through the thickness of RCC. The first path gives the orifice type of flow where the leakage is directly proportional to square root of differential pressure. The second type of leak is through zigzag paths in the thickness of concrete where the flow is molecular diffusion and is directly proportional to differential pressure. The observed data suggests adopting linear least square fit correlation which is conservative and indicates that predominant leak is through concrete. From this correlation, the maximum permissible leak rate at a pressure of 150 mb works out to be 0.077% volume per hour.




After 1987, periodic tests were conducted biennially and the leakage rate variation is between 0.034 and 0.054% volume per hour, well below the maximum permissible limit. This indicates that the integrity of RCB has been well maintained over the past 25 years. Conducting each test required about 10 to 15 days of preparatory work with reactor in shutdown state. The consistent lower leakage rate data further suggest that there is a scope for decreasing the testing frequency to reduce reactor downtime.

## Augmentation of Shielding

Immediately after the first criticality in 1985, meticulous radiation surveys were carried out at all accessible locations in RCB at 10 kWt. This helped in spotting streaming locations of concern, locations where the shielding was not incorporated by oversight during commissioning stage and locations that needed augmentation of shielding. Concerted efforts were taken to augment the shielding expeditiously so that the radiation levels could be brought down to acceptable values for sustained high power operations.

The major streaming location was noticed over the sodium sampling plug and argon supply line above Over Flow Tank. The projected dose rate for 10.5 MWt at this location was calculated to be as high as 1 Gy/h. This streaming was due to the deficiency in the design of central straight-line thimble opening for cover gas supply/discharge. The shield plug was redesigned, fabricated and installed. Additional lead shielding weighing about 2.5 T was also provided over the OFT. Subsequently, the measured background dose rate above OFT at 17.4 MWt was less than 1 μGy/h with a narrow streaming of 50 μGy/h.

Similar streaming location of concern was noticed at sodium flooding line at -10.5 m level where there was straight penetration into the B-cell. After the augmentation of additional shielding, the dose level was brought down by two orders. The radiation surveys have also revealed a few points where shielding materials should have been placed during commissioning but missed. There were two such locations viz., nitrogen valve penetration from B3 to D5 cell and penetration B2D5S1 at -2.81 m level; at both these locations, lead shots and colemanite were introduced. Augmentation of lead shielding brought down the field at few more places such as i) above DND pits (east & west) at zero level, ii) above sodium filters at -2.81 m level, iii) above B6 cell (purification cabin) at -2.81 m level, iv) B1C1V1 duct (near A1 cell man-hole) below -10.5 m level and, v) B8 cell doors (east and west). Maximum neutron field over top of the pile was 20 μSv/h.

## Radioactive Waste Disposal

Generation of solid waste in FBTR is very minimal. Category-III solid wastes were disposed only once so far during 1994 and consisted of cut portions of damaged CRDM. The surface dose rate on the waste bag was 500 mGy/h and it was essentially due to $Co^{60}$. Category-II wastes were disposed only twice so far (SS cut pieces of CRDM with a maximum surface dose rate of 0.5 mGy/h due to $Co^{60}$ and activation of defective cables used for neutron detectors, containing 447 MBq of $Ag^{110m}$). The potentially active wastes are mostly the High Efficiency Particulate Activity filters removed from RCB exhaust.

Liquid effluent from decontamination facilities and condensates from air handling units located in potentially active areas are stored in separate SS tanks and pumped to Centralised Waste Management Facility for disposal. Before disposal, sample from the liquid effluent is collected and analysed for activity. The beta-gamma activity of the liquid effluent has so far been below the detection level (18 kBq/$m^3$). However, samples from the sediment accumulated in the liquid effluent tanks showed the presence of $Co^{60}$, $Cs^{137}$, $Mn^{54}$, $Cs^{134}$ and $Sb^{125}$. Radioactive sodium removed from primary purification cabin was treated and sent to CWMF as liquid waste containing cumulative activity of 188 MBq of $Na^{22}$.

Radioactive gaseous wastes are filtered and released to the environment through a 65 m tall stack. The major radioactive component expected to be present in

the gaseous effluent under normal operating condition is $Ar^{41}$. The cover gas argon is sampled continuously by CRD circuit to detect any clad rupture and is vented out through the stack. Continuous monitoring of the stack discharge is carried out with the help of induct and stack monitors. Besides, spot samples are periodically collected from different sampling points provided along the exhaust duct & the stack to identify the radionuclides and estimate their activities. Under failed fuel-pin condition, provision exists in the design to store the radioactive cover gas in four storage tanks for decay of the activity before release. Each tank is of 4 $m^3$ volume and can be pressurised up to 5 bar. The gaseous activity releases are negligibly low compared to the Technical Specification limit of 3330 TBq per annum for $Ar^{41}$ & fission product noble gases together. The integral release is consistent with reactor operation history; i.e., the release of $Ar^{41}$ per EFPD is 17.5 GBq till date indicating no uncontrolled release.

The maximum observed $Ar^{41}$ activity released through stack so far has been 4750 GBq in the year 2000 when the reactor was continuously operated at 13.4 MWt. If one rounds off this to 5000 GBq and further multiply it by a factor of three (since the maximum designed power level of FBTR is 40 MWt), it would total to 15 TBq and the dose to the public due to $Ar^{41}$ release would be only 0.000225 mSv against the apportioned limit of 0.05 mSv. Even if one adds the release of Fission Product Noble Gases due to fuel pin failure, if any, it is considered adequate to have a figure of 0.01 mSv for FPNG release including $Ar^{41}$.

The $I^{131}$ release is monitored continuously by a NaI(Tl) detector based gamma ray spectrometer installed at the exhaust duct in the stack. Special equipment for removal or control of iodine is not required for FBTR exhaust gas system because of the retention of iodine by the coolant sodium. Efficient iodine filter system provides adequate control of the release of radioactive iodine, in the event of postulated substantial clad failure. Estimation of $Sr^{90}$ and other particulate fission products are done by continuous filter paper sampling of stack exhaust air. There had been no release of $I^{131}$, $Sr^{90}$ and other particulates so far.

## Sodium Leak Incident and its Disposal

During the sodium leak incident in purification circuit, five out of six particulate activity monitors located in RCB showed increasing trend of air activity level coinciding with the first alarm annunciation by wire type leak detector. The release of sodium aerosols into the RCB atmosphere had come down after about six hours since shutdown. The maximum air activity during the incident was 0.1 DAC of $Na^{24}$ only. Effective radiological surveillance was provided during the normalisation period of three months. Moreover, based on the feedback obtained during this incident, the alarm threshold for Particulate Activity Monitor was lowered to 10 times the background value. The total dose expenditure during this entire operation was 2.25 person-mSv only. The spectral analysis of the sodium sample collected from

inside the purification cabin indicated the presence of $Hg^{203}$ (4342 Bq/g), $Na^{22}$ (2023 Bq/g), $Ag^{110m}$ (4.55 Bq/g), $Sb^{124}$ (4.44 Bq/g) and $Zn^{65}$ (4.19 Bq/g) isotopes. While the presence of $Na^{22}$ is expected, the mercury contamination had come long back to the primary sodium from the relief pot connected to the cover gas system. Presence of $Ag^{110m}$ and $Sb^{124}$ activation products were due to the impurities present in the sodium and the activation product $Zn^{65}$ was due to the presence of zinc in the paint used for coating of the ferritic steel components. The leaked out sodium removed from purification cabin was converted to sodium hydroxide in decontamination pit and sent as sodium phosphate to the liquid effluent tank. A dedicated Particulate Activity Monitor located near the decontamination pit did not indicate any increased particulate activity during the operation. About 188 MBq of liquid effluent was generated and disposed.

## Decontamination of components

During the two decades of operation of FBTR, decontamination of several components such as lower portion of CRDM, micro filters, sodium centrifugal pumps, irradiated fuel SA, etc. were carried out. Decontamination of the damaged CRDM was carried out to reuse part of the mechanism having lesser radioactivity and dispose the rest having higher activity. Intense gamma field of 500 mGy/h was measured on contact, which was essentially due to $Co^{60}$ activation in Stellite portion. The outer surface of the CRDM was cleaned by steam-nitrogen process to remove sodium deposits contaminated with $Co^{60}$. The liquid waste was of 2 MBq/$m^3$ activity. Decontamination of the guide tube was carried out several times in the decontamination pit.

## Study on Deposited Activity in Primary Sodium Pipelines

Measurements were undertaken to assess the deposition of radioactive corrosion and activation products in the primary sodium pipelines and various components, which are housed in B-cells. Such an assessment would guide health physicists in controlling the exposures to occupational workers of FBTR during maintenance work. In 2002, after draining sodium from the primary pipelines, the ambient gamma radiation levels in the B-cells came down by five times from 200 to 40 μGy/h. This was essentially due to the presence of $Na^{22}$ (78%) and soluble radioisotopes such as $Hg^{203}$ (10%) in the sodium coolant. Spectral investigation studies revealed that deposited activity in the primary sodium pipelines consists of $Mn^{54}$ (93%), $Na^{22}$ (2.5%), $Co^{60}$ (2%), $Co^{58}$ (1.5%), $Sb^{124}$ (trace), and $Zn^{65}$ (trace) isotopes. Presence of $Co^{58}$ and $Mn^{54}$ in the lines confirm that the corrosion and transport of core material activation products to the primary sodium pipelines.

## On-line Monitoring of Gaseous Effluent

An on-line GM tube based system was developed to estimate the specific activity of the gaseous effluent vented out through stack. In view of its low Minimum Detectable Activity (27 kBq/$m^3$), the system is useful in analysing gaseous activities

of lower levels. To estimate the $Ar^{41}$ activity in the cover gas without any uncertainty, the detector was calibrated with $Na^{22}$ liquid source. Necessary corrections for self-absorption were applied to arrive at conversion factors for gas measurements. This system is currently in service.

## DND Experiments with Special Perforated Sub-Assembly containing Natural Uranium

After loading the special SA in the reactor core for DND experiments, fission product particulate activity was observed in RCB during reactor operation. Also, during the course of conducting DND experiments, higher than normal ambient radiation levels in RCB, release of air-borne activity in RCB and increase in the readings of CRD monitors due to release of Fission Product Noble Gases into the cover gas were observed, though all these were not much of concern from the radiation exposure point of view. Even after removal of the special SA from the core, the air-borne activity was persisting in RCB during subsequent reactor operations and it was established that this was due to fission of tramp natural uranium present in primary sodium that had come from the special SA and not due to the failure of irradiated fuel or blanket assemblies. To reduce the air-borne activity in RCB, a cocoon has been installed over the Block Pile to direct any active cover gas leak in to the effluent.

## Environmental Aspects of FBRs

Environmental impact is an important global issue. The release of radioactivity from FBRs is negligible due to its design features. A blanket of inert gas and a protective envelope covering the primary sodium vessel & pipelines rules out the escape of primary sodium into the RCB atmosphere. Design features of FBRs also significantly minimise the generation of liquid and solid wastes. Generation of solid wastes in the form of spent ion exchange resins used in radioactive water purification in thermal reactors is not applicable to FBRs. By the simple technique of lowering the sodium temperature, oxide impurities are precipitated out in cold traps and the purified sodium is put back into circulation. These cold traps are designed to last for the entire life of the reactor.

The collective exposure of the occupational workers is consistently lower. The mining and milling operations are reduced by a factor of at least 50 resulting in considerable reduction in dose rates from these operations. The thermal impact on the environment is less for FBRs due to higher thermal efficiency. The low operating pressure of FBRs virtually precludes the possibility of a catastrophic failure of sodium pipes or vessels as an accident initiator. FBR designs are also characterised by negative feedback mechanism. The considerable quantity of sodium in the main vessel serves as a large heat sink by virtue of the thermal properties of sodium. The boiling point of sodium being relatively high, a very large amount of heat can be absorbed without leading to bulk boiling of sodium in the core. By separating the long-lived actinide group of elements from the High Level Liquid Waste and adding them to the fresh fuel loaded into FBRs relieves the problem of long-term storage of the vitrified waste.

# Summary

Like every journey through a new technology, the journey of FBTR has been an exciting game of ups and downs. Every challenge taught us a lesson and we returned stronger. More gratifying is that the overall progress has been one of upward travel.

FBTR was conceived as a springboard for the development of sodium cooled fast reactor technology in India. We will conclude this book by summarising the contribution, so far, of FBTR in fulfilling the aspirations of its founding fathers and defining its future missions.

## Feedback from Project Experience

When the project was initiated, except some of the project team leaders, all the others were novices. Most of them were fresh from the BARC Training School. To many, the dull, matted grey look of stainless steel was a surprise, since the only idea of stainless steel they had was from the polished stainless steel kitchenware. To many of the Indian industries who undertook fabrication of components for FBTR, welding and assembly of stainless steel components to the stringent quality levels and dimensional tolerances was a new experience. It was a phase of collective learning by all. From such an initial state of knowledge and experience, to have built a reactor with 80% indigenisation, to a quality level where the reactor components have withstood 25 years of operation in high temperature sodium, is no mean achievement. More important, the project team was quick to realise the greater challenges in the fabrication of components for large fast reactors like PFBR and put in place a Component Fabrication Technology Development Programme for PFBR. They pooled in their experience gained from FBTR. The successful fabrication of many of the large components of PFBR to exacting tolerances, with no major slippage in schedule, owes in large measure to the Technology Development programme spearheaded by the FBTR project engineers.

## Sodium Technology

There were many in the project team who had not seen sodium. The need for well-trained operators for FBTR and the need for facilities for testing of FBTR components and instruments in sodium was one of the major factors that led to the design and construction of small sodium loops in the Engineering Halls of IGCAR. A large rig for the purification of commercial grade sodium to reactor grade purity

was then put up, to meet the requirements of FBTR. The initial impetus given by FBTR led to the next stage where now large sized sodium rigs have been set up in bigger Engineering Halls for testing the large components of PFBR.

FBTR also gave the impetus to the growth of sodium chemistry in IGCAR. In addition to carrying out periodic chemical and spectral analysis of sodium samples from FBTR, the Chemistry Group has developed several sensors including electrochemical hydrogen meter, electrochemical carbon meter and hydrogen-in-argon meter for which FBTR became a major test bed. The electrochemical hydrogen meter developed by CG was installed in Phenix, based on its satisfactory performance in FBTR and SGTF.

## Front end of Fuel Cycle

The change over to the carbide fuel called for the ab-initio design of a novel fuel, characterising it, finalising its fabrication flow sheet and design, construction & commissioning of a carbide fuel fabrication facility with inert cover. This is a major contribution of FBTR to the front end of a fuel cycle involving Pu. RMD, BARC also characterised fuel of MK-II composition and fabricated pins for 13 SA. In addition, high Pu MOX (44% $PuO_2$) was characterised by RMD, BARC and safety clearances were obtained by IGCAR for the hybrid core. Annular MOX fuel pellets of the PFBR fuel dimensions were made by Advanced Fuel Fabrication Facility, Tarapur. Eight SA of this composition have been loaded in the core of FBTR. These have amply vindicated not only our ability to develop different types of fast reactor fuels, but also our ability to design fast reactor cores with different types of fuels.

The Chemistry Group of IGCAR played a role in characterising the carbide fuel and conducted out-of-pile tests in assessing the allowable LHR of the fuel. They conducted compatibility studies of fuels with sodium. Currently, laboratory scale facilities for making sol-gel fuel pins and sodium-bonded metallic fuel pins are in advanced stages of commissioning. Irradiated fuel pins and pellets are periodically sent to the CG for studies related to fission gas release, burn-up measurement, etc. A failed fuel localisation set-up, developed by CG for identifying the failed fuel by the age method, has been in service in the primary cover gas reject circuit. FBTR has thus, spurred the fuel chemistry activities of the Centre.

## Radiometallurgy

The Radiometallurgy laboratory of IGCAR is a unique facility. It handles fuels with very high burn-up levels (~18 atom %). The hot cells are provided with nitrogen cover as the SA and capsules sent from FBTR are contaminated with sodium and the fuel itself is pyrophoric in nature. In the initial years, RML fabricated the neutron source pins for FBTR by encapsulating active antimony pellets and

transferring the capsules to FBTR. In subsequent years, FBTR provided the wherewithal for the growth of Radiometallurgy programme by transferring fuel SA at different burn-up levels for PIE. Control rods, irradiation capsules and nickel SA have also been periodically sent to RML through the inter-facility transfer mechanism. Depending on the requirements of examination, RML has perfected need-based inspection techniques including Metrology, Eddy Current Testing, Gamma radiography, X-radiography, Neutron Radiography, etc. Techniques for sodium decontamination, remote cutting of SA & capsules, retrieval of pins & foils, cutting of fuel pins & extracting the pellets, puncturing the pins for fission gas studies and measuring the release fraction, etc. have been perfected by RML. Studies on the mechanical properties of wrapper and clad are also carried out in these hot cells. Thus, FBTR paved the way for the overall growth of Radiometallurgy programme in the Centre. Currently electron microscopy facilities are being added for characterisation of irradiated fuel and structural materials.

## Back-end of the Fuel Cycle

The reprocessing of fast reactor fuel is far more complicated than fuel from thermal reactors. The very high burn-up levels of the fast reactor fuels result in much higher radioactivity levels of the fuel being handled, with their attendant problems like higher background activity, degradation of solvent by radiolysis, etc. The pins are contaminated with sodium and cannot be directly dissolved in aqueous medium. Being pyrophoric, chopping has to be done under inert cover. R&D on process flow sheet development, dissolution and the equipment development on laser dismantling of SA, single pin chopper, electrolytic dissolver and centrifugal extractor led to the construction and commissioning of the hot cell facility, CORAL. Spent fuel with different burn-up levels from as low as 25 to as high as 155 GWd/t with cooling times as low as two years have been reprocessed in this facility. The mastery of the technology of the carbide fuel has given us the confidence in reprocessing the power reactor MOX fuels. Concurrent with reprocessing, R&D activities related to the recovery of minor actinides from low level solid wastes have also been initiated in CG.

## Metallurgy

During the project days, the Metallurgy Programme of RRC played a major advisory role to the project team on welding & heat treatment of stainless steels and hard-facing (stelliting) of FBTR components. In the wake of non-availability of SS filler wires as per the original specifications, they proposed a suitable alternative after extensive testing and qualification. The metallurgists also carried out extensive investigation on the welds made as per FBTR specification, with specific reference to delta ferrite, Heat Affected Zone and embrittlement due to high temperature ageing.

In the light of apprehensions raised about Stress Corrosion Cracking of stainless steels due to chlorides in the saline atmosphere at Kalpakkam, they put in place a programme for testing the base metals and welds for susceptibility to SCC and alleviated the fears. They also characterised the creep & fatigue resistance of the typical base metals and welds used in FBTR. The initial experience with FBTR gradually expanded into a major R&D programme for PFBR materials.

FBTR is used as a test bed for development of core structural materials. The objectives of these developmental activities are to meet the requirements of enhanced burn-up up to 200 GWd/t and metallic fuel SA. The Metallurgy and Materials Group has developed a 15Cr-15Ni-Ti modified austenitic SS clad tube that enables high burn-up. This alloy is used as cladding and wrapper for the PFBR fuel SA. The PFBR test SA, eight MOX SA and a few tensile specimens made of this alloy are currently under irradiation in FBTR. A variant of this alloy is under development. Also, modified 9Cr-1Mo ferritic steel with high-toughness and ferritic Oxide Dispersion Strengthened alloys with high-creep strength are being developed for use in future FBRs, including for use with metallic fuel. These materials will be irradiated in FBTR in the coming years. Instrumented capsules for irradiation are also under development by MMG.

## Electronics, Instrumentation & Devices

The Electronics and Instrumentation Group of IGCAR has developed the ED-20 based, real time, embedded computer systems for PFBR. Both the computer sub-systems of the Centralised Data Processing System in FBTR have been replaced by the ED-20 systems and are under evaluation. The feedback experience so far has been satisfactory, inspiring confidence in the design and its deployment in PFBR.

The Reactor Vessel displacement measurement system became obsolete. The Electronics and Instrumentation Group developed a new microprocessor based system, which has been performing very well.

FBTR has also been the test bed for the high temperature fission chambers developed by BARC for PFBR, the Kalman filter technique developed by BARC for confirming the dropping of DSR in PFBR and the acoustic technique developed by FRTG for the same purpose. The under-sodium ultrasonic device developed by FRTG was extensively tested, for the first time in sodium, in FBTR in the wake of the fuel handling incident and the CCPM incident. The refined version of this device will find its application in PFBR. The novel air ultrasonic system, BAT, developed by E&IG found its application in profiling the guide tube during the fuel handling incident. The Eddy Current Flow meter developed by FRTG is being periodically used in FBTR. It has been continuously modified based on operational experience. A

modified version of the sensor was sent to Phenix for end-of-life experiments. ECFM will also be deployed in PFBR.

## Growth of Analytical Strength

The changing core configuration of FBTR, with the progressive enhancement of the burn-up limit of the carbide fuel, has made an immense contribution to the growth of studies related to reactor safety, incident analysis, etc. FBTR also acts as a testing ground for validating the analyses through experiments. Problems like the core temperature anomaly due to the stuck Core Cover Plate led to the analysis of plenum hydraulics. The seismic re-evaluation of FBTR called for a complex modelling of the primary sodium system with its double envelope. The Civil Engineering Division carried out the seismic analysis and capacity assessment of several civil structures, including the Reactor Containment Building. Reactor physics and shielding codes developed in the Centre are being fine-tuned based on the measurements in FBTR. FBTR has thus made a significant impact on the analytical and computational strength of the Centre.

## Operational Experience

About 150 $m^3$ of sodium has been kept molten for more than 25 years in FBTR, of which about 84 $m^3$ is in continuous circulation in the primary and secondary loops with unimpaired purity. We have mastered the art of major interventions in sodium systems without any incident and even without contaminating the sodium by air ingress. The experience and confidence gained in FBTR has enabled us to leapfrog to a stage where currently about 1400 T of sodium has been charged into the storage tanks of PFBR without any incident.

There has been no clad failure upto a burn-up of 165 GWd/t, but measures to tackle any pin failure are in place. The four sodium pumps have cumulatively logged 650,000 hours of trouble-free operation. The SG modules have operated without any leak since the first valving-in of water. This is a credit not only to the ruggedness of the design and quality of fabrication, but also to the strict chemistry control of the feed water. Steam at the highest pressure and temperature among the nuclear power plants in India is being handled without incidents. The hundreds of modifications carried out in the plant testify to the ability of the plant personnel to listen to the pulse of the plant, recognise the deviations, analyse the causes, identify the correct solutions and engineer them to correct standards. We now have the ability to design and conduct safety related experiments in a reactor with a reactive core. We can design irradiation devices and safely handle them. The BSC coil leaks led to the development of indigenous chemicals and procedures for remote leak sealing. All

these testify to mastery in operating and maintaining a typical fast reactor power plant.

## Feedback to PFBR design

Starting from the NaK leak incident during commissioning, every significant event in FBTR has its prompt feedback to PFBR design. Every incident is studied by the designers for its applicability to PFBR. It is analysed for its implications on PFBR operation. Wherever required, the design is modified.

With the confidence obtained from the smooth manual power control in FBTR, manual control philosophy is retained for PFBR also. Design provisions have been incorporated in PFBR based on the fuel handling incident, inoperability of CCPM, sodium and NaK leaks and trailing cable core failures. FBTR operating experience in optimising the trip parameters and noise elimination in neutronic channels has been utilised to ensure availability of reactor without compromising on safety. The use of oil has been minimised in the auxiliary systems of PFBR based on a few oil fire incidents in FBTR. Improvements were carried out in CRDM power supply schemes based on the incident of reverse rotation of a CRDM in FBTR. The water leaks from the BSC coils led to design changes to ensure the integrity of BSC coils in PFBR.

The successful operation of FBTR, excellent track record of nuclear and radiological safety and the confidence in the design based on FBTR feedback experience were some of the main factors in according the project sanction for PFBR.

## Human Resources

FBTR is the cradle for trained human resources for operating the next few fast reactors. A few engineers from the O&M team, who had an initial stint in CIRUS, underwent training in Rapsodie. They came back and trained the engineers, supervisors and operators for FBTR. Most of them had on-the-job training in sodium handling in the engineering loops and participated in all the commissioning activities of FBTR. We thus have a pool of trained manpower at all levels for fast reactors. There was some attrition in the wake of the IT boom following the Y2K, but they were replaced by fresh batches of trainees. Experienced personnel from FBTR are also being redeployed elsewhere in the Centre.

The O&M personnel of PFBR, including engineers, supervisors and technicians have been undergoing both class room and on-the-job training in FBTR. A few qualified senior shift engineers from FBTR are also proposed to be deputed to PFBR for commissioning and the first few years of operation.

## Contribution to the Regulatory Framework

AERB has a rich experience in the regulation of PHWR. FBTR being the first sodium cooled fast reactor in India, regulatory guidelines had to be evolved. FBTR station documents, coupled with the operating experience, provided the basis for formulating the regulatory guidelines for fast reactors in India. A few experienced officers from FBTR were transferred to AERB and have been playing a major role in the PFBR safety reviews. In recent years, fresh engineers from AERB have also been undergoing training in FBTR, thus providing trained human resources for regulating the fast reactors in future.



*The planting of a banyan tree sapling by the former Prime Minister, (late) Shri. Rajiv Gandhi on the eve of FBTR criticality is highly symbolic. The banyan tree sends down aerial roots to the ground. On reaching the ground, they sustain themselves and become strong enough to be props for the tree itself. FBTR catalysed the growth of all facilities related to the various facets of fast reactor technology and is now supported by them, thus manifesting true symbiosis.*

## The Journey will continue...

The current residual life of FBTR has been estimated as 10 Effective Full Power Years. (The term 'Effective Full Power Year' for FBTR connotes the number of years of operation at the peak design Linear Heat Rating of 400 W/cm as the life of FBTR is controlled by the neutron dose on the Grid Plate. It is not linked with a specific power level).

One question often raised is why a reactor that has been designed for 40 MWt/13.2 MWe has been operated at 18.6 MWt/3.2 MWe only. The change-over to the carbide fuel with initial uncertainties on the recovery of Pu from the spent fuel called for a cautious approach in investing Pu. This resulted in the small carbide core rated for 10.6 MWt. This factor has continued to be one major influence in shaping the core and power evolution of FBTR. The policy of IGCAR has been to achieve the set missions of FBTR with minimum fissile inventory. Power generation is not the main agenda of FBTR. During the initial years following the valving-in of water into the SG, reactor was being operated for gaining operating experience. In subsequent years, it has been operated with specific missions, in order to reach specific goals.

By adopting the graded approach, we are now left with a reactor which has a residual life of 10 years, ready to be the work-horse for its subsequent missions for the development of the next generation of fast reactors. The pace of operation which has been intentional and also by coincidence due to the host of challenges faced and the modifications done has also been in tune with the pace of development of advanced fuels and core structural materials. It should be heartening to note that so much has been accomplished with so little fissile inventory.

## The Future Missions

One of the major missions of FBTR since 2003 was the irradiation of PFBR test fuel. The PFBR test SA also imposed a limitation on the operating LHR of the driver fuel. The PFBR fuel will be discharged at the end of the current campaign as it has been taken to the target burn-up. It will then be possible to raise the operating LHR of the driver fuel and raise the power to ~22 MWt. This will enable raising the sodium temperature to 500°C. Feedback from the 16$^{th}$ campaign indicates that there are no more major technical issues as regards to high temperature operation. Inputs from the pilot reprocessing plant CORAL are encouraging with regards to our ability to close the fuel cycle. DFRP, the regular reprocessing facility for FBTR fuel, will be commissioned shortly. All these augur well for the operation of FBTR at close to the

design temperatures with good availability factor in the near future. This will enable FBTR to take up meaningful irradiation of advanced core structural materials being developed by the Centre.

The major agenda of FBTR in the coming years will be to study the performance of metallic fuel. Currently, two designs of metallic fuel pins are under consideration: the sodium bonded pins and mechanically bonded pins. They will be irradiated to burn-up levels of 100 GWd/t and the final choice made based on their performance.

Towards developing the technology of large scale metallic fuel pin fabrication, a demonstration facility with a capacity of 600 pins per year is being put up. This facility is scheduled to be commissioned by 2014 and in subsequent years, metallic fuel SA will be inducted on a large scale surrounding the carbide driver fuel. Though these pins may not operate at the high design LHR characteristic of metallic fuel, they will definitely provide rich experience in the fabrication and irradiation of metallic fuel. The irradiated pins are proposed to be taken out at periodic intervals and reprocessed in a pyro-reprocessing facility being planned at IGCAR. The closing of fuel cycle through the pyro-route will call for changes in the fuel handling systems of FBTR and provide feedback for the design of the metal fuelled fast reactors. When these are accomplished, FBTR would have fulfilled all its intended missions. The regular production of $Sr^{89}$ for medical applications is also on the agenda as a societal application.

Currently, the $9^{th}$ ring of the core has been loaded with Thoria blankets. It is proposed to progressively load Thoria blankets in the $10^{th}$, $11^{th}$ & $12^{th}$ rings. They will remain in place till the end of life of the reactor, as originally envisaged. When discharged, they will provide data on the characteristics of $U^{233}$ derived from a fast reactor and help in developing the flow sheets for reprocessing and recovering $U^{233}$ from the breeder route. This will be a contribution of FBTR to the third stage of our nuclear programme.

**The woods are lovely, dark and deep,**
**But I have promises to keep,**
**And miles to go before I sleep,**
**And miles to go before I sleep.**
**- Robert Frost**

*(The favourite quote of Pandit Jawaharlal Nehru, the architect of modern India)*

# Acknowledgment

*We thank Dr.S.Banerjee, Chairman, Atomic Energy Commission and Dr.Baldev Raj, Director, IGCAR for their support in this endeavour. We also acknowledge the contributions of the following in preparing the book.*

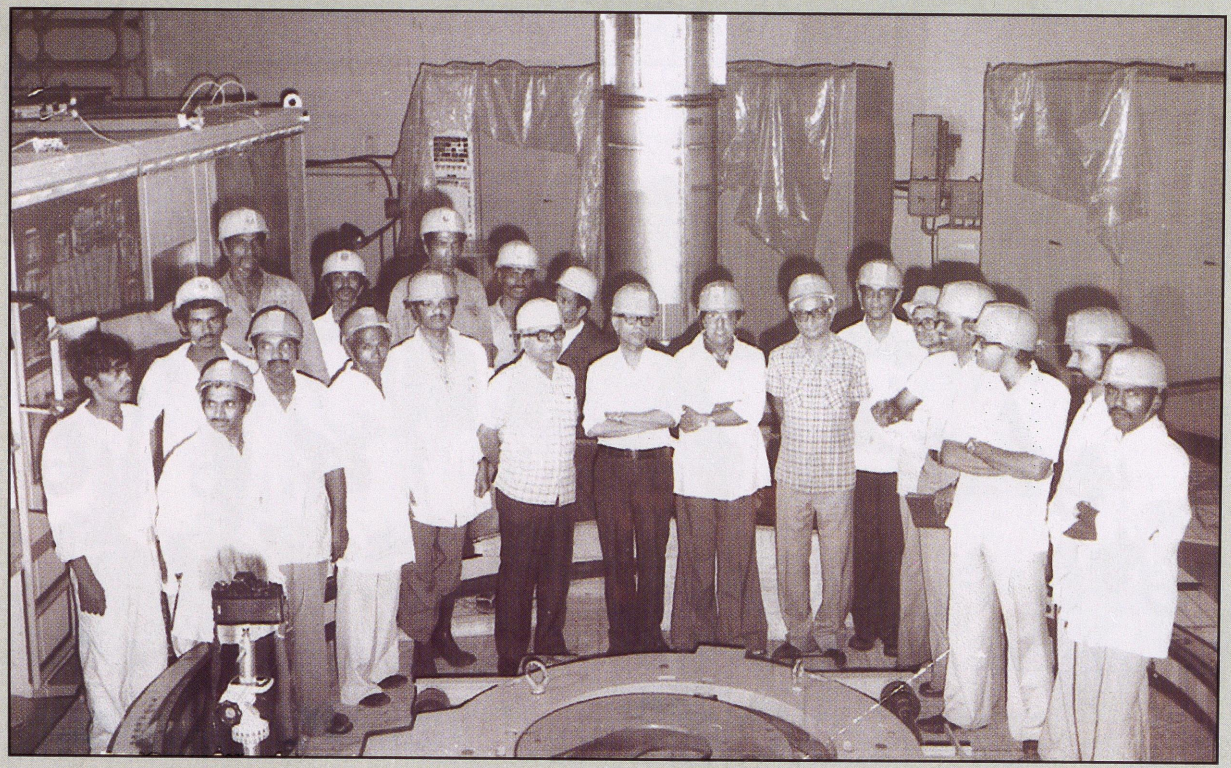
**Dr S.M. Lee**
**Shri R.D. Kale**
**Shri G. Vaidyanathan**
**Shri K.K. Rajan**
**Dr V. Meenakshisundaram**
**Shri R.R. Ramanarayanan**
**Dr M.T. Jose**
**Shri Arun Kumar (BARC)**
**Shri S.C. Chetal**
**Dr P.R. Vasudeva Rao**
**Dr P. Swaminathan**
**Dr C.S. Sundar**
**Shri R. Natarajan**
**Dr T. Jayakumar**
**Dr V. Ramanathan**
**Shri R.V. Subba Rao**
**Shri M. Murugesan**
**Shri K. Ganesan**
**Shri K. Varathan**

# *From our records*





*Veterans of Project and O&M Team*
*(On the occasion of 20 years celebration-2005)*



*Construction Team during erection of IHX (1982)*



*Senior Officers of the Project Team*



*Former Chairmen (Late) Dr. Raja Ramanna & Dr. M.R. Srinivasan in the Control Room after first criticality (1985)*



*Construction Team during erection of Grid Plate Assembly (1983)*



*Former Prime Minister (Late) Shri Rajiv Gandhi planting a sapling (1985)*



*Former Prime Minister Shri. Atal Bihari Vajpayee and Former Governor Ms. Fathima Beevi being briefed by (Late) Shri R. P. Kapoor, Reactor Superintendent, inside RCB (1998)*




*Dignitaries during 20 year celebration of FBTR*




*Dedication of FBTR to the Nation*

## Abbreviations

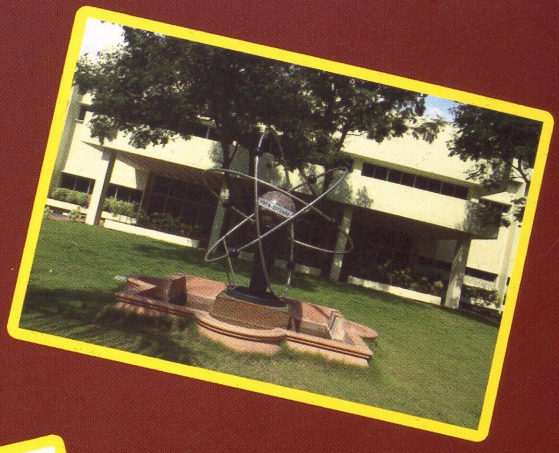
| | |
|---|---|
| **AB** | Active Building |
| **AEC** | Atomic Energy Commission |
| **AERB** | Atomic Energy Regulatory Board |
| **BARC** | Bhabha Atomic Research Centre |
| **BEP** | Best Efficiency Point |
| **BHEL** | Bharat Heavy Electricals Limited |
| **BOR-60** | Russian Fast Reactor |
| **BSC** | Biological Shield Cooling |
| **BTMS** | Blockpile Temperature Monitoring System |
| **C/M** | Carbon to Metal ratio |
| **CB** | Control Building |
| **CBP** | Condensate Booster Pump |
| **CCP** | Central Canal Plug |
| **CCPM** | Core Cover Plate Mechanism |
| **CDPS** | Central Data Processing System |
| **CEA** | French Atomic Energy Commission |
| **CEP** | Condensate Extraction Pump |
| **CPU** | Condensate Polishing Unit |
| **CR** | Control Rod |
| **CRD** | Clad Rupture Detection |
| **CRDM** | Control Rod Drive Mechanism |
| **CTG** | Capsule Transfer Gripper |
| **CWMF** | Centralised Waste Management Facility |
| **DAE** | Department of Atomic Energy |
| **DAE-SRC** | DAE-Safety Review Committee |
| **DBA** | Design Basis Accident |
| **DBGM** | Design Basis Ground Motion |
| **DCPL** | Development Consultant Pvt Limited |
| **DFR** | Dounrey Fast Reactor |

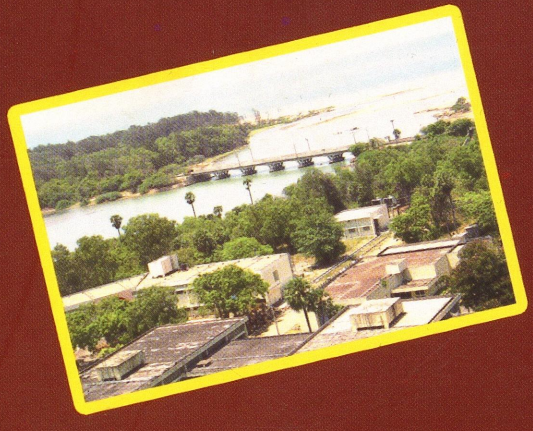
| | |
|---|---|
| **DFRP** | Demonstration FBR Fuel Reprocessing Plant |
| **DHR** | Decay Heat Removal |
| **DLP** | Deaerator Lift Pump |
| **DMD** | Displacement Measuring Device |
| **DND** | Delayed Neutron Detection |
| **dpa** | displacement per atom |
| **DSR** | Diverse Safety Rod |
| **EBR** | Experimental Breeder Reactor |
| **ECIL** | Electronics Corporation of India Limited |
| **EFPY** | Effective Full Power Years |
| **EM Coil** | Electro Magnetic Coil |
| **EM Pump** | Electro Magnetic Pump |
| **FBTR** | Fast Breeder Test Reactor |
| **FCCI** | Fuel Clad Chemical Interaction |
| **FCMI** | Fuel Clad Mechanical Interaction |
| **FFSA** | Fresh Fuel Sub Assembly |
| **FFTC** | Fresh Fuel Transfer Chamber |
| **FH** | Fuel Handling |
| **FIT** | Fine Impulse Test |
| **FRFCF** | Fast Reactor Fuel Cycle Facility |
| **FRTG** | Fast Reactor Technology Group |
| **GAM** | Gas Activity Monitor |
| **GWd/t** | Giga Watt days/tonne |
| **HAD** | Hydrogen-in-Argon Detection |
| **HEPA** | High Efficiency Particulate Activity |
| **HLT** | Helium Leak Testing |
| **IAEA** | International Atomic Energy Agency |
| **IFSA** | Irradiated Fuel Storage Area |
| **IGCAR** | Indira Gandhi Centre for Atomic Research |
| **IHX** | Intermediate Heat Exchanger |
| **IIT** | Indian Institute of Technology |
| **INES** | International Nuclear Event Scale |
| **LHR** | Linear Heat Rating |
| **LM** | Liquid Metal |
| **LMFBR** | Liquid Metal Fast Breeder Reactor |

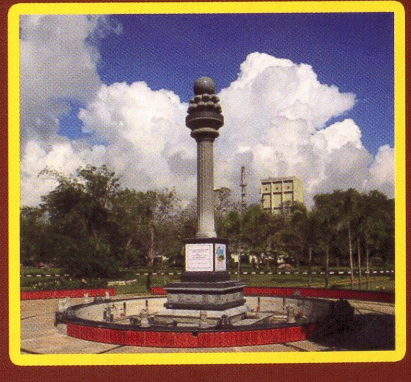
| | |
|---|---|
| **LMS** | Liquid Metal Seal |
| **LOCA** | Loss of Coolant Accident |
| **LOFA** | Loss of Flow Accident |
| **LOR** | Lowering of Rods |
| **LPFT** | Low Pressure Flash Tank |
| **LPH** | Low Pressure Heater |
| **LRP** | Large Rotatable Plug |
| **LSSS** | Limiting Safety System Setting |
| **MAPP** | Madras Atomic Power Project |
| **MAPS** | Madras Atomic Power Station |
| **MB** | Maintenance Building |
| **MBFP** | Main Boiler Feed Pump |
| **MI** | Mineral Insulated |
| **MK-I** | Mark-I |
| **MOX** | Mixed Oxide |
| **NFC** | Nuclear Fuel Complex |
| **NPSH** | Net Positive Suction Head |
| **NRV** | Non Return Valve |
| **O&M** | Operation & Maintenance |
| **O/M** | Oxygen to Metal |
| **ODC** | Over Dimensional Component |
| **OFT** | Over Flow Tank |
| **PAM** | Particulate Activity Monitor |
| **PC** | Personal Computer |
| **pcm** | Percent milli K (multiplication factor $x10^{-5}$) |
| **PFBR** | Prototype Fast Breeder Reactor |
| **PHEC** | Pre Heating and Emergency Cooling |
| **PHWR** | Pressurised Heavy Water Reactor |
| **PIE** | Post Irradiation Examination |
| **PWHT** | Post-Weld Heat Treatment |
| **QC&IW** | Quality Control & Inspection Wing |
| **R&D** | Research & Development |
| **RAAMS** | Radiation and Air Activity Monitoring System |
| **RBGM** | Review Basis Ground Motion |
| **RCB** | Reactor Containment Building |

| | |
|---|---|
| **RCL** | Radio Chemistry Laboratory |
| **REL** | Reactor Engineering Laboratory |
| **RMD** | Radiometallurgy Division |
| **RML** | Radiometallurgy Laboratory |
| **RRC** | Reactor Research Centre |
| **RTD** | Resistance Temperature Detector |
| **RV** | Reactor Vessel |
| **SA** | Subassembly |
| **SARCOP** | Safety Review Committee for Operating Plants |
| **SB** | Service Building |
| **SCC** | Stress Corrosion Cracking |
| **SED** | Safety Engineering Division |
| **SF** | Special Flask |
| **SG** | Steam Generator |
| **SGB** | Steam Generator Building |
| **SGLD** | Steam Generator Leak Detection |
| **SIP** | Sputter Ion Pump |
| **SRI** | Safety Research Institute |
| **SRP** | Small Rotatable Plug |
| **SS** | Stainless Steel |
| **TB** | Turbine Building |
| **TDC** | Trombay Digital Computer |
| **TG** | Turbo Generator |
| **TLM** | Technical Liaison Mission |
| **TNT** | Tri Nitro Toluene |
| **TOPA** | Transient Over Power Accident |
| **UPS** | Uninterrupted Power Supply |
| **WL** | Ward-Leonard |









Published by Scientific Information and Resource Division, IGCAR, Kalpakkam.