# Fast Breeder Test Reactor

Journey of FBTR - Reaching Further Heights

50 years
Journey of
FBTR

April 30, 2022

सत्यमेव जयते

Government of India
Department of Atomic Energy
Indira Gandhi Centre for Atomic Research
Kalpakkam - 603 102

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डॉ. बी. वेंकटरामन
प्रतिष्ठित वैज्ञानिक एवं निदेशक

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## Message

It gives immense pleasure to me for being part of this wonderful occasion of releasing the launch of second edition of FBTR book titled "Journey of FBTR- Reaching Further Heights" on the eve of founder's day. It indeed is a sequel to its earlier endeavor titled "An exciting journey through a new technology" published to mark the completion of 25 years of successful operation of FBTR. Started with a modest journey with 22 fuel subassemblies (SAs) core rated for 10 MWt, FBTR has reached its designed power of 40 MWt on 07.03.2022. This momentous occasion is etched in the annals of Indian fast reactor programme. It is commendable that this feat has been achieved with a novel core of indigenous design with MARK-1 fuel SAs and few poison subassemblies. Full kudos to FBTR team for this greatest milestone with special mention to the successive team leaders of FBTR and their dedicated team without which this momentous feat could not have been achieved.

During this wonderful journey, FBTR has carried out many mission based programmes including testing of metallic fuel pins of different composition, PFBR test fuel SA & TRISO coated surrogate fuel of compact high temperature reactor. Its journey never ends here & many more laurels are waiting on the wings.

With an estimated residual life of more than 6 EPY, FBTR can contribute to testing of metallic fuel SA & advanced structural materials for future fast reactor to gain valuable base line data. Being a fast breeder reactor with a flux level more than one order magnitude compared to thermal reactors and high neutron energy, FBTR could be a source of generating radioactive isotopes $Sr^{89}$ & $P^{32}$ for societal applications in future. I am sure that FBTR would continue to provide valuable inputs to the fast reactor program through irradiation of metallic fuels, structural materials and also human resource development.

I wish FBTR all the best for its future endeavors.

(B. Venkatraman)

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# Preface

The Fast Breeder Test Reactor is a 40MWt/ 13.6 MWe, loop type, sodium cooled fast reactor using a mixture of Plutonium carbide and uranium carbide as the fuel. Heat generated in the reactor is removed by two primary sodium loops, and transferred to the corresponding secondary sodium loops. Each secondary sodium loop is provided with two once-through steam generator modules. Steam from the four modules is fed to a turbine-generator. A 100% dump condenser is also provided to operate the reactor at full power even during non availability of Turbo Generator or Main condenser. FBTR attained first criticality on 18th October 1985 and completed thirty six years of successful operation on 18th October 2021. Thirty six years of successful and purposeful operation of Fast Breeder Test Reactor is a significant mega stone in the annals of fast breeder and closed fuel cycle technologies.

Fast Breeder Test Reactor (FBTR) continues as the flagship of Indira Gandhi Centre for Atomic Research (IGCAR) which is dedicated to fast reactor research in the second stage of the Indian nuclear power program. To mark its 25 years of successful operation, a book titled “An exciting journey through a new technology” was released in October 2010 which broadly covered the operation history of the plant, experience gained during commissioning, challenges encountered during the journey and major modifications/retrofitting implemented for enhancing safety and ease of operation. Since then, FBTR has crossed many more milestones including operation of the plant at it design power of 40 MWt. Covering important achievement and the challenges faced by FBTR, this book titled “Journey of FBTR - Reaching Further Heights” is released. It is really a pleasure to release this in the eve of founder’s day.

Being a test reactor, its main objective is to operate on mission mode irradiation campaigns. So far 29 irradiation campaigns have been successfully completed since its inception and 30th campaign is in progress. Even though plant is rated for 40 MWt, maximum power could be realized with MARK-I, Mark-II and MOX fuel subassemblies (SAs) respecting the minimum required shutdown margin (SDM) was only 32 MWt. With an objective to reach rated power, core was redesigned by adding four 50% enriched $B_4C$ poison SAs to maintain the SDM above the minimum required value (4200 pcm) with all Mark-I FSA core. After completing all mandatory reactor physics experiments, on 07.03.22 at 17.30 h, reactor power was raised to the target power of 40 MWt in the 30th irradiation campaign. This has been a historical moment and achievement for FBTR. All major works such as normalization of blanked tubes of Steam Generator and revamping of cooling tower have been completed prior to starting of the campaign.

FBTR continues to serve as the irradiation test for advanced fuels & structural materials for future FBRs. Also it contributes by producing radioactive isotope $Sr^{89}$ from $Yr^{89}$ for societal applications. As a prelude to gain experience on metallic fuel, metallic fuel pins of different compositions are being irradiated to generate base line performance data.

Besides its achievements, plant has faced several challenges also. There was an incident of tube leak in one of the modules of once through Steam Generator (SG). Identification of failed SG and its replacement have been of the first of kind experience for FBTR. The faulty module which remained interconnected in the water and sodium sides in west loop was identified successfully and was replaced within a record time of two months. After few months of operation, minor sodium leak occurred from one of the thermal baffles of the replaced SG module. This module was replaced with a redesigned module of different material of construction. There were instances of few fuel pin clad failures in the recent past in various campaigns. A robust procedure was used for failed fuel identification which has minimized core shuffling to a large extent.

Based on the operational experience feedback, maintenance issues & obsolescence, plant undertook upgradation of systems, components and structures to enhance its safety to the current standards. Further, post Fukushima, an extensive retrofitting programme has been carried out to protect the plant against the external events such as flood, Tsunami & Seismicity. As a part of seismic retrofitting programme, the adequacy of system to withstand Safe Shutdown Earthquake (SSE) for safe shut down, decay heat removal and containment integrity have been assessed and found to meet the regulatory requirements.

As part of relicensing, comprehensive Periodic Safety Review (PSR) was carried out after competing 20 years of operation. PSR covered ageing management of non-replacable components, equipment qualification, In Service Inspection (ISI), Operational experience feedback & compliance with current regulatory standards. Based on extensive multilayer review, license was obtained to operate the reactor till 2018 June. Subsequently, Application of Renewal of Authorization (ARA) was submitted & license has been extended up to 2023 June. Second comprehensive PSR has been commenced to get regulatory consent beyond 2023.

Irradiation induced changes in mechanical property of grid plate is one of the factors considered in estimating the remaining life of FBTR. Based on 10% residual ductility criteria of grid plate at the end of life, plant has a residual life of 6.32 Effective Full Power years (EFPY). The remnant life would be used for irradiation of advanced fuel and structural materials. Also to further enhance the residual life of reactor, it is planned to replace bottom axial stainless steel (SS) pins used as shielding material in the fuel SA with Tungsten Carbide (WC) to reduce the fluence seen by the grid plate.

K. V. Suresh Kumar

Director, RFG

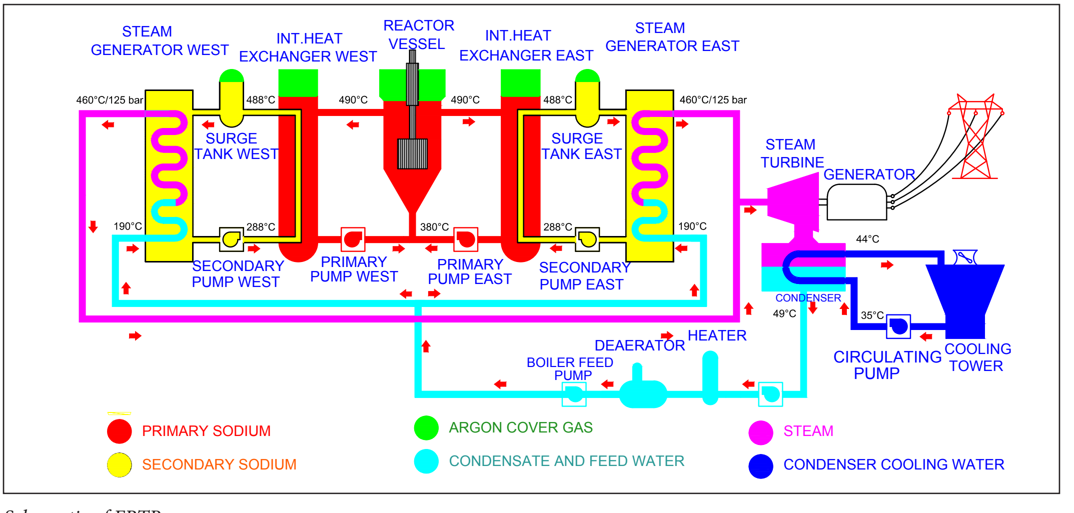
# Journey of FBTR - Reaching Further Heights

## 1. Introduction

The Fast Breeder Test Reactor (FBTR), the flagship reactor of the second stage of the Indian nuclear power program, has completed 29 irradiation campaigns and thirty-six years of safe and successful operation, since its inception. The 30th campaign is presently in progress and is successfully operating at its design capacity of 40 MWt.

FBTR located at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, is a 40 MWt loop type, sodium cooled fast reactor. Its main objective is to provide experience in fast reactor operation, large scale sodium handling and to serve as a test bed for irradiation of future fast reactor fuels & advanced structural materials.

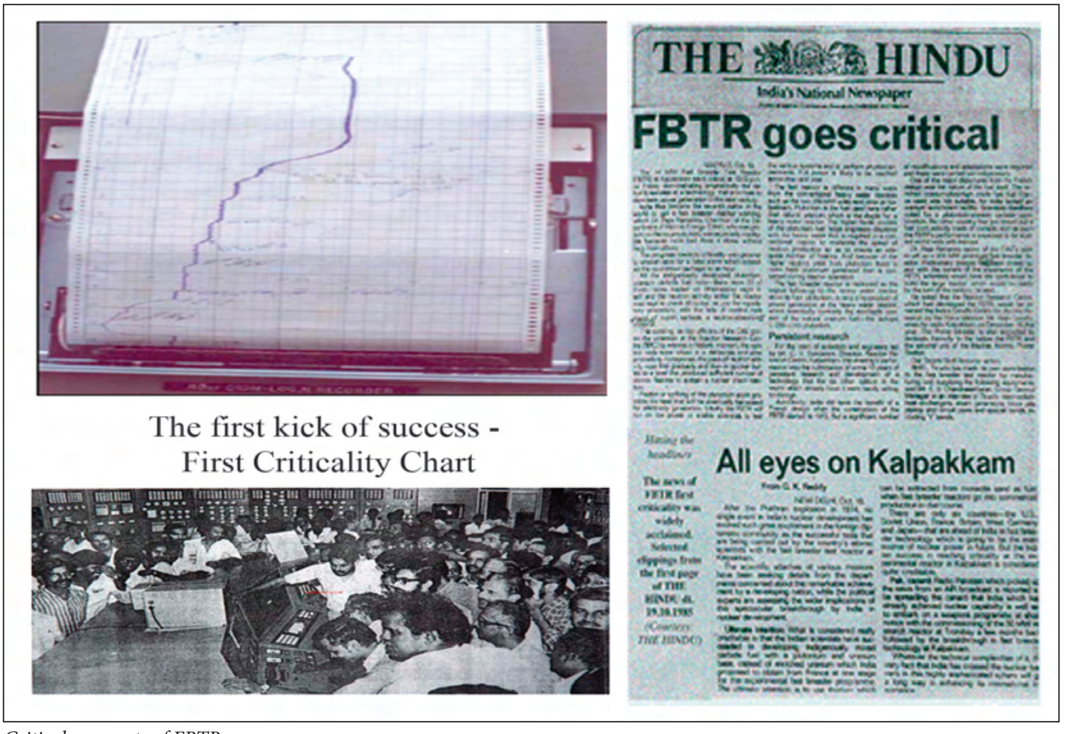
FBTR heat transport system consists of two primary sodium loops, two secondary sodium loops and a tertiary steam water circuit. Heat generated in the reactor core is transported to the tertiary circuit through the primary and secondary sodium circuits. There are two modules of Steam Generators (12.5 MWt capacity) in each loop and SG modules are enclosed in a common casing. The SGs are not insulated to facilitate decay heat removal by natural convection of air through the casing. A 100% steam dump facility is provided in the steam water circuit to operate the reactor at full power for experimental purposes even when turbine is not available.




*Schematic of FBTR*

## 2. Journey of FBTR

The reactor achieved criticality with 22 carbide fuel subassemblies (70% PuC +30%UC FSA) in October 1985. Commissioning of the plant was carried out in four stages such as the low power operation without SG (1985), operation with sodium on SG shell without valving-in water on the tube side (1986-90), power operation with water valved into SG (1993-1996) and operation with Turbo-Generator (1997 onwards) which culminated in the complete integration of all the systems of the reactor.

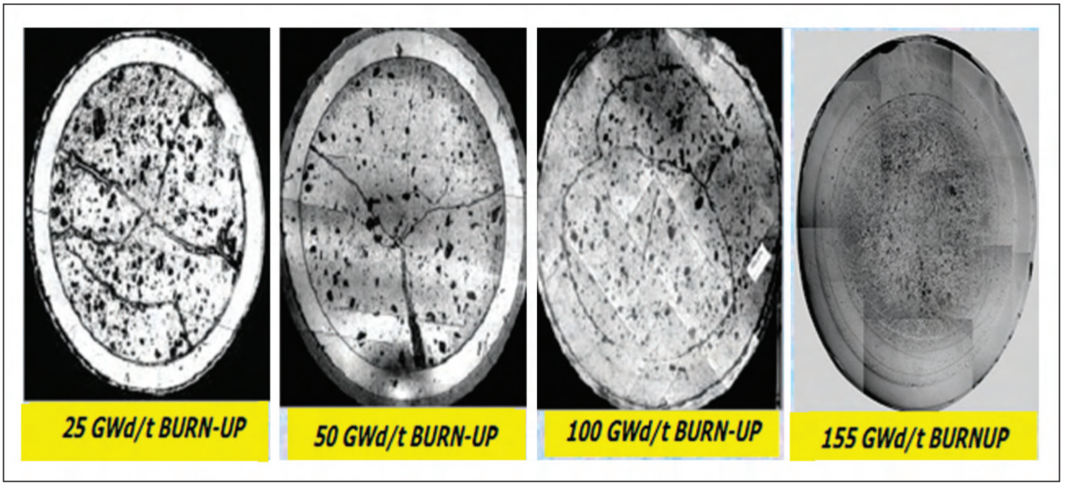



*Critical moments of FBTR*

Lack of in-pile experience with the carbide fuel limited the linear heat rating of the fuel to 250 W/cm initially and the target burnup was set at 25 GWd/t. The target burn-up has been progressively increased based on the post-irradiation examination (PIE) performed on the fuel Subassemblies (SA) discharged after attaining 25, 50, 100 and 155 GWd/t of burnup. The Linear Heat Rating (LHR) was increased from the initial value of 250 W/cm to 320 W/cm based on the excellent performance of the fuel. The current limit on burnup is 140 GWd/t for SAs operating above 320 W/cm and 155 GWd/t for the Mark-I SAs operating ≤320 W/cm. In the light of the excellent performance of the carbide fuel to endure a burnup of 155 GWd/t, one lead SA was irradiated up to 165 GWd/t. The core has been gradually expanded by the addition of Mark-I (70% PuC- 30% UC) along with Mark-II (55% PuC-45% UC) and MOX (44% $PuO_2$-56% $UO_2$) Fuel SAs (FSA) to compensate for the burnup reactivity loss and replacement for burnt Mark-I and Mark-II SAs. With the addition of fresh SAs, the reactor power has been progressively raised from 10.6 MWt to a maximum of 40 MWt. Based on further studies, PIE results and fuel availability, the peak LHR has been increased to 400 W/cm from 17$^{th}$ irradiation campaign onwards.

The original design values for the reactor inlet and outlet temperatures are 380°C and 515°C respectively. In order to achieve these temperatures, with the constraints on core size and reactor power, a major modification, i.e., blanking of three tubes out of the seven tubes in each SG module (Total: 4 nos.) was done in 2008. With this change, it was possible to conduct irradiation of fast reactor materials close to design temperatures even at lower reactor power varying from 22.5 to 32 MWt and study the performance of various systems.

Till 29$^{th}$ irradiation campaign, FBTR could be operated only for a maximum rated power of 32 MWt




*Post irradiation Examination at various burn ups*

due to restriction on shutdown margin (SDM). In 30th irradiation campaign, FBTR core was redesigned into 40 MWt core (all Mark-I SA core) by adding poison SAs in the second ring and additional MK-I fuel SAs. 40 MWt operation necessitated the normalisation of blanked SG tube modules and the same was done.

The 40 MWt core has 68 MK-I fuel SAs. Reactor power was raised to rated design power level of 40 MWt on 07.03.22 with TG synchronized to grid delivering about 10 MWe.

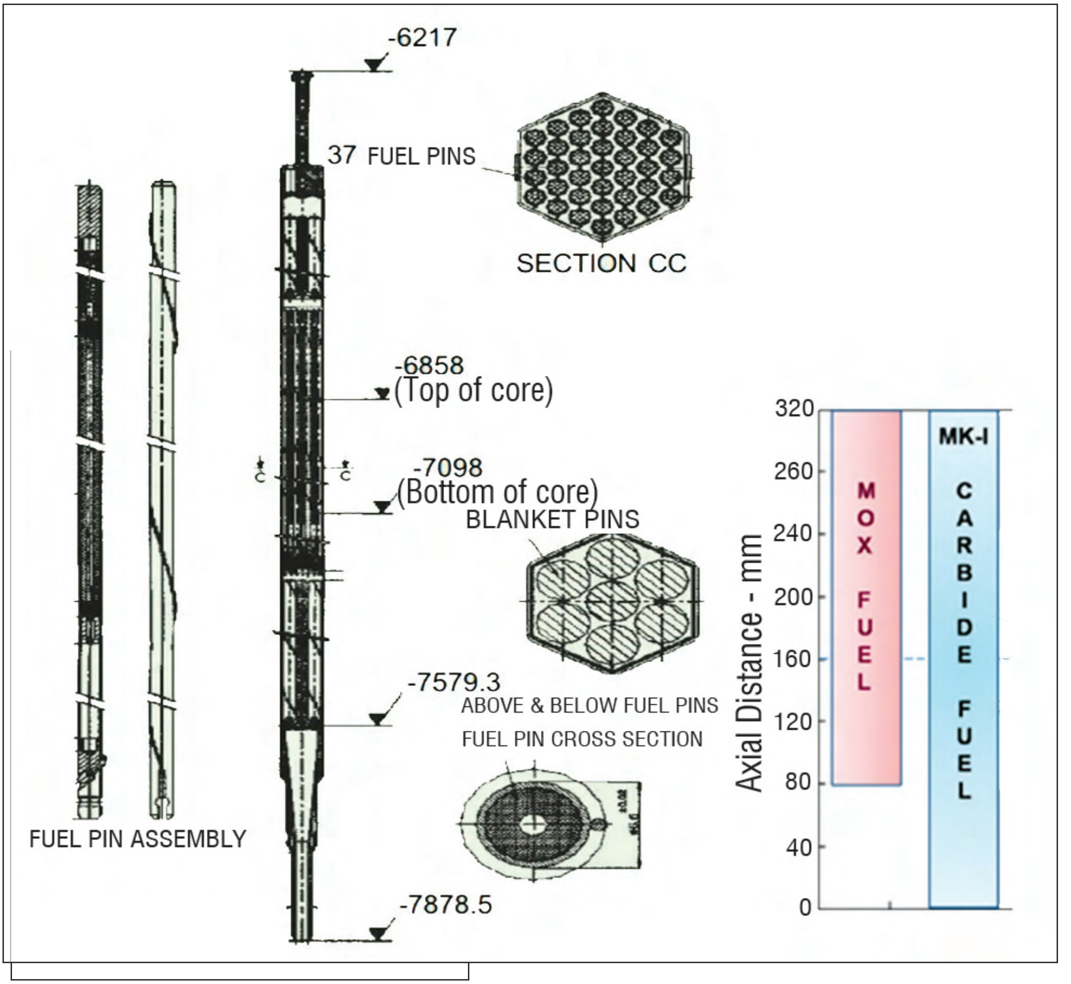
### 3. Irradiation of Structural and Fuel Materials

FBTR has been utilized for irradiation of Prototype Fast Breeder Reactor (PFBR) test SA up to 112 GWd/t, TRISO coated particle with $ZrO_2$ kernels and disc specimens of Nb-1Zr-0.1C for HTGR. Moreover, short term irradiation of the sphere-pac fuel pins and irradiation of ferro-boron shield material have been completed for future developments. The reactor was also utilized for studying the irradiation creep behaviour of Zr-Nb alloy which is being used in the Indian Pressurized Heavy Water Reactors. The feasibility of producing radioactive $Sr^{89}$ used in bone cancer therapy was demonstrated. Tungsten carbide (WC) which is a potential lower axial shield material in fuel subassemblies for reducing the fluence on the grid plate has been irradiated to the required fluence and the PIE results are encouraging.

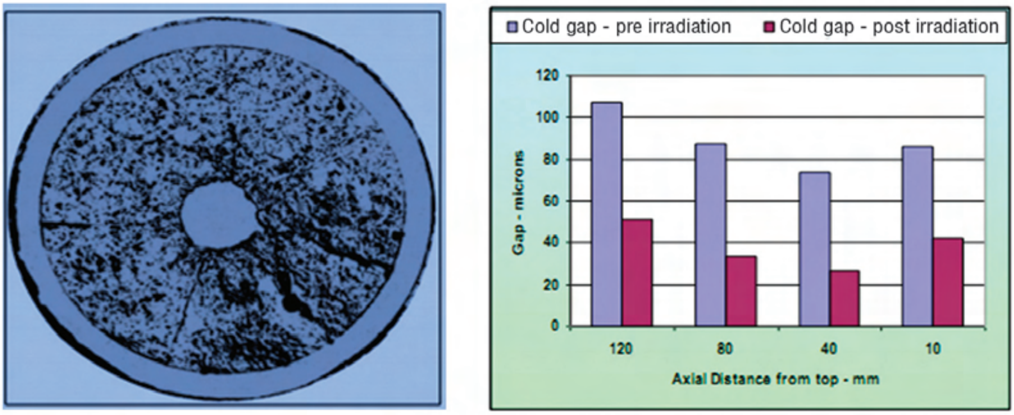
### 3.1 Irradiation of Prototype Fast Breeder Reactor Test Fuel Subassembly

FBTR reached a major milestone in October 2010 when the irradiation of the PFBR test fuel was completed. The irradiation of the MOX fuel simulating the PFBR fuel composition commenced in the year 2003 in the 11th irradiation campaign. PFBR is designed with a mixed oxide fuel of plutonium and natural uranium of two different compositions viz. 21% and 28 % of $PuO_2$ for the two enrichment zones. The fuel is designed to operate at a peak linear heat rating (LHR) of 450 W/cm and target burn-up of 100 GWd/t. The flux in FBTR core is $3x10^{15}$ n/cm$^2$/s, whereas the flux in PFBR is 8 x $10^{15}$ n/cm$^2$/s. To achieve the linear heat rating of 450 W/cm with the lower flux in FBTR, the PFBR test fuel was enriched with $^{233}U$, thus retaining the chemical composition as in PFBR fuel. Typical fuel composition of the pellets of the PFBR test subassembly is 29% $PuO_2$, 38% oxide of $^{233}U$ and balance 33% oxide of natural uranium. The composition was designed such that when the high-grade MK-I fuel in the first ring was operating at a peak linear heat rating of 320 W/cm, the test subassembly operated a peak linear heat rating of 450 W/cm at the core centre. The PFBR test subassembly had 37 pins of 6.6 mm OD and 5.7 mm ID, same as that of PFBR fuel pins. On either side of the fuel pin bundle, steel rods were used as reflectors. The fuel pellet had 5.52 mm OD with a central hole of φ1.6 mm. Active fuel column length was 240 mm as against 320 mm for FBTR carbide fuel. Top of the fissile column was in level with the top of the core (-6858 mm). The pin had two classes of pellets- class A pellets with a linear mass of 2.18 g/cm and class B pellets with a lower linear mass of 2.11 g/cm. Class B pellets were loaded at the top 1/3$^{rd}$ of the fissile column. The test subassembly generated 350 kWt at 450 W/cm. Accordingly, the subassembly was with a zone IV orifice. In addition, the steel reflector pin dimensions were slightly different from standard FBTR blanket pin dimensions in order to modify the hydraulic characteristics of the subassembly as required. The clad and wrapper are made of cold worked D9 alloy.

While it was possible to operate at the required linear heat rating in FBTR and reach the target burn-up, displacement per atom (dpa) and clad damage fraction (cdf) corresponding to the target burn-up in PFBR could not be simulated in FBTR fully due to the lower flux and lower sodium temperature. For the operating parameters of FBTR, the clad damage fraction that could be achieved was




*PFBR TEST SA*



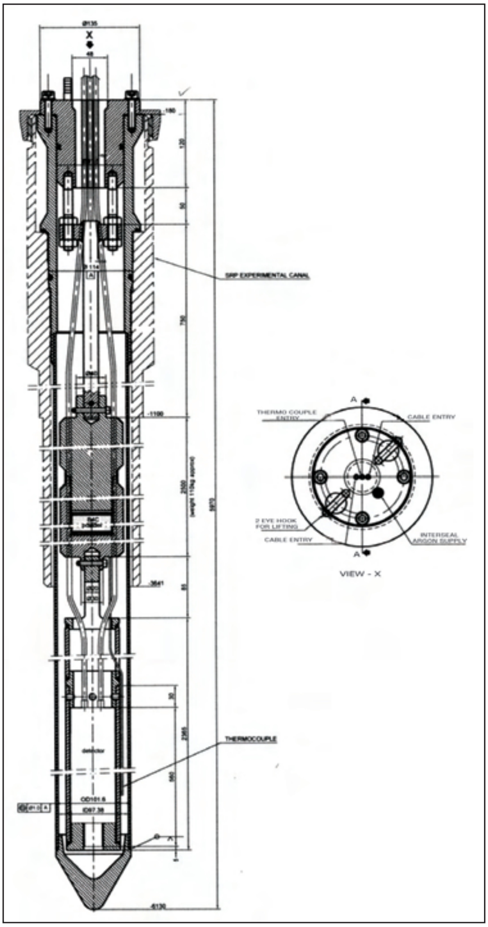

*Photomosiac of PFBR test SA*

about 0.12 as against the PFBR design clad damage fraction of 0.3. However, the irradiation helped in understanding the behaviour of the MOX fuel per-se. Since $^{233}$U will be used in the third stage of India's nuclear energy programme, the irradiation provided some useful and valuable insights into its behaviour as well. Initially, the subassembly was intended to be irradiated up to a peak burn-up of 100 GWd/t at a reactor inlet temperature of 380 $^{0}$C. Irradiation of this sub-assembly was carried out in six irradiation campaigns starting from 11$^{th}$ campaign. During this period, the core progressively evolved from 39 FSA core to 47 FSA core in the 16$^{th}$ campaign. During the 11$^{th}$ to 14$^{th}$ campaigns, the reactor was operated with inlet sodium temperature of 350 $^{0}$C only. In later campaigns, the reactor inlet temperature was raised to 380 $^{0}$C by blanking three out of seven tubes in all the four SG modules. Based on the lower operating inlet temperature during the earlier campaigns, the burn-up limit was raised to 112 GWd/t. The revised target burn-up was reached at the end of 16$^{th}$ irradiation campaign. After in-core cooling for about three months, when the decay heat reached 200 W, the subassembly was discharged and sent to hot cells for post-irradiation examination. After completion of PIE on this test FSA, feedback was given to the designers. PIE results validated the design of the fuel SA and its manufacturing process.

The principal material of choice for the reactor core components and coolant circuit in FBTR is stainless steel (SS 316). To develop alternate materials for structural components, various materials like SS 304LN, SS 316L, SS 316LN, D9, ferritic –martensitic steel and tungsten carbide (a potential axial shield material) have been irradiated so far. Irradiation of sodium bonded metallic fuel pins i.e. ternary fuel-1 (23% Pu-19% EU-6% Zr) ternary fuel-2 (19% Pu-U-6% Zr), binary fuel (14.8% EU-6% Zr) and natural uranium-6% Zr metallic fuel and testing of instruments and equipment at high temperatures which are important to India's fast reactor programme has also been carried out.

### 3.1.1 Testing of PFBR High Temperature Fission Chambers

The prototype of the High Temperature Fission Chambers (HTFC) required for PFBR developed




*Sketch of special fuel sub-assembly*

by BARC were tested in FBTR by installing it in the experimental canal (517 mm radially from core centre) at an elevation of 300 mm above the top of the subassembly.

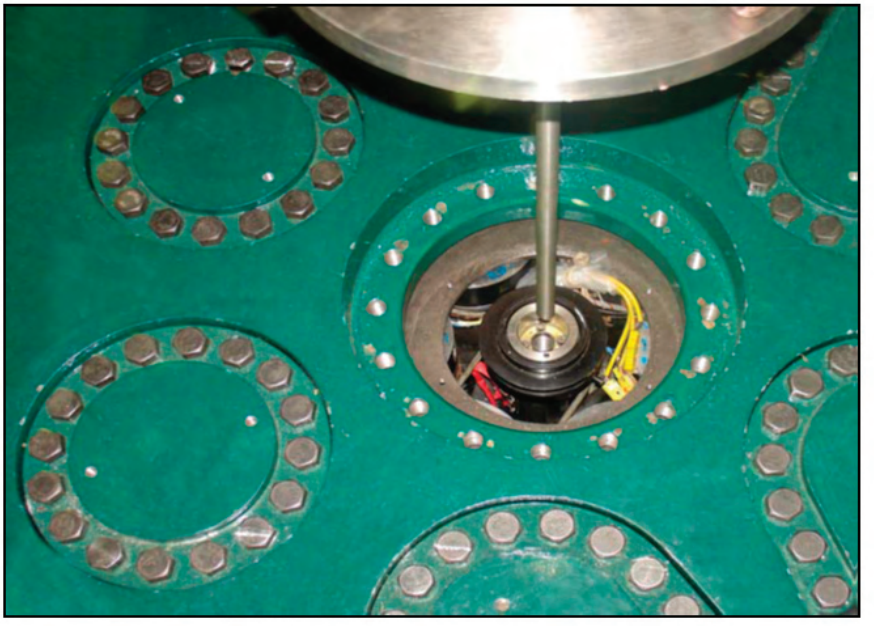
The detector made of Inconel 600, is designed to measure flux in the range of 0 to $10^9$ n/cm$^2$/s and can withstand gamma up to 5x$10^6$ R/h and temperature up to 600°C. It has two SS 316 sheathed, triaxial MI cables of 6 mm OD and 12 m length, and weighs 5.4 kg. The detector was introduced into an outer sheath assembly already installed in the experimental canal of reactor. Another detector was assembled in the detector cage and installed in the detector pit E5 in the biological shield. The signal cables were connected to the two wide range channels.

The readings of both the detectors were logged, when the power was raised to 16.2 MWt. The temperature at the in-core detector was around 470°C. The test results were shared with the developer.

**Gamma Measurement**

High Temperature Fission Chambers (HTFC), to be used in PFBR for monitoring the neutron flux at various stages, are being developed indigenously. As these detectors are exposed to high temperatures (around 570$^o$C) and high gamma fields of 5 x $10^5$R/h in PFBR, the performance of the detectors has to be tested. These detectors were tested in BARC at high temperature and high gamma field independently. The gamma field in FBTR has two major components, one due to sodium activation gammas and the other coming from the core. The gamma coming from the latter is composed of capture gammas and fission gammas. The sodium activation is mainly from $Na^{24}$ activity with a half life of 15 h and the activity saturates after a few half lives. However, core gammas are directly proportional to the reactor power. Hence, the total gamma activity in the detector has a component depending on the reactor power and reactor operation history.

As the calculated gamma flux has inherent error due to the approximations in the reactor modelling and errors and uncertainties in the data used, it was felt necessary to measure the gamma field and measure the gamma dose distribution in central canal location and extrapolate in the experimental canal using the calculation. A miniature high temperature gamma detector was used to measure



*Gamma detector being loaded into CCP*

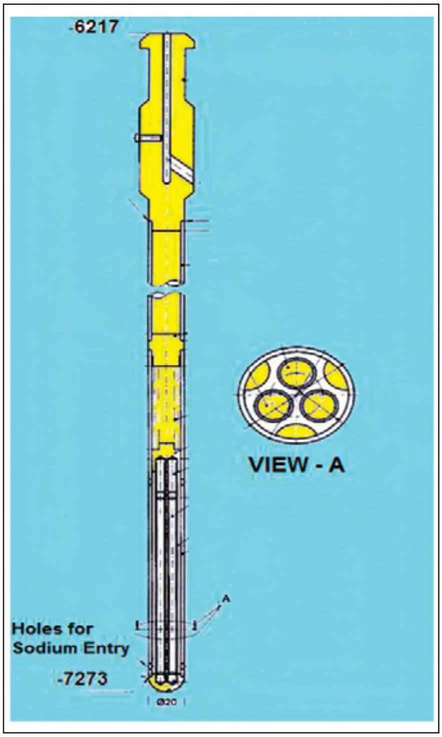
the gamma dose distribution. This detector was earlier calibrated in a standard gamma field. The efficiency of the detector was found to be $3.7 \times 10^{-14}$A/(R/h). A special plug was designed to load the detector in the central canal location.

The supplementary shield plug was taken out from the central canal plug (CCP) for inserting the gamma detector for gamma radiation field measurement inside reactor. A Ferro-boron shield box was fabricated and kept over the SRP anti explosion floor exactly over the central canal to prevent radiation streaming during reactor operation. After the gamma field measurement, the detector and shield box were removed, supplementary plug was installed in CCP and the pile was normalized.

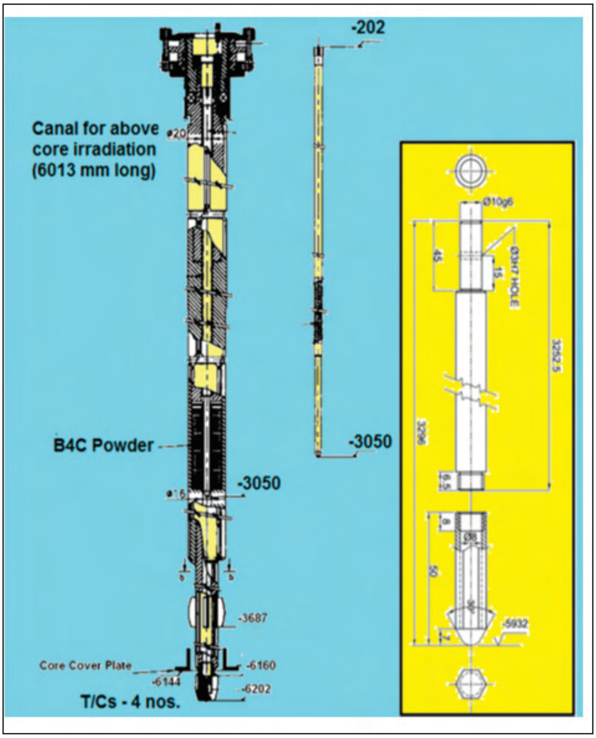
The HTFC detector holder was modified to have the detector located at about 1 m from the present level to reduce the effect of gamma on the detectors.

**3.2 Fast Flux Irradiation Facility**

A new facility has been added for fast flux irradiation and cross-section measurements above the core, using the cavity in the Central canal Plug. Position of the sample holder is above the sodium level. The cavity in the central canal plug (CCP) extends from top to the bottom which is just above the head of the central fuel subassembly. The cavity is closed using a small auxiliary shield plug to limit the dose at the top of the plug due to neutron and gamma streaming during reactor operation. Four thermocouples are provided in the CCP to monitor the temperature of the sodium at the outlet of the central subassembly. The auxiliary shield plug of the CCP is modified by attaching a titanium rod with a sample holder at its end. A titanium sample holder of 50 mm length and Φ12 mm is attached to the bottom of this rod. The bottom position of the sample holder is below sodium level and about



*Typical Irradiation capsule for testing materials*



*Fast Flux Irradiation Facility*

315 cm on top of the central fuel subassembly. Insertion and removal of samples would be from the top of the reactor. Titanium was chosen based on the properties of mechanical integrity under the irradiation conditions and the gamma dose seen due to the activation of the material and the impurities in the material.

After operation for 10 days at 400 kWt, the dose due to the titanium rod works out to be about 125 mR/h at a distance of 2 cm and reduces to < 1mR/h at 1 m distance.

Two test irradiations were carried out using this facility at 25 kWt power for 4 h. In both cases about 10/15 mg of gold, wrapped in high purity aluminium foil was used. The foils were assayed by high resolution gamma spectrometry and it was computed that the gold equivalent flux was $2.5 \times 10^9$ n/cm$^2$/s. This facility is proposed to be used for determination of reaction cross sections, and fast neutron activation analysis.

**3.3 FBTR- a Test Bed for Experiments**

Over the years, several safety related physics experiments have been conducted. These include measurement of temperature and power coefficients of reactivity, sodium void coefficient, reactor kinetics experiments, response of delayed neutron detection system to detect clad failure and flux mapping in sodium above core by foil activation, control rod drop-time measurements using Kalman filter technique, measurement of worth of fuel subassemblies & control rods, testing of neutron detectors and experiments to validate the Failed Fuel Detection System. Several engineering tests have been carried out to validate the results obtained using the codes meant for incident analysis like the off-site power failure, tripping of one pump in the primary, secondary or tertiary loops, natural convection tests to assess the heat removal capacity and the occurrence of events as per the sequence stipulated in the safety analysis. Further, primary pump coast down characteristics, take-over by the batteries and low speed running of the pumps were studied and found to satisfy the design intent. Other tests to study the heat removal capability of Biological Shield Cooling (BSC) &Preheating and Emergency core Cooling (PHEC) systems, RCB leak rate, SG instability and concrete temperature evolution during station black out were conducted and the results obtained thereof were validated.

**4.0 Challenges Faced**

Some of the challenges faced during the 36 years of operation include a major fuel handling incident, primary sodium leak, reactivity transients, leak in biological shield cooling coils and leak in steam generator both on tube side and shell side. The experience gained in facing these challenges has been documented and will be highly useful for the design and operation of future fast reactors.

**4.1 Steam Generator Replacement**

Steam generator (SG) of FBTR is a once-through shell and tube type counter flow heat exchanger. The sodium flows in the shell side and water/steam through the tube side. There are two steam generator (SG) modules each in east and west secondary sodium loops connected in parallel. Each module rated for 12.5 MWt consists of 8 mm thick shell of outer diameter 198 mm and seven numbers 4 mm thick tubes of diameter 33.7 mm. The total heat transfer length is 90.4 metre and the material of construction is 2.25Cr- 1Mo-Nb stabilized ferritic steel. The steam generator generates superheated steam at 125 kg/cm$^2$ and 480 $^o$C by transferring nuclear heat from primary sodium to the tertiary steam water system through secondary sodium with inlet sodium at 510 $^o$C.

As the sodium water reaction is exothermic with evolution of $H_2$, highly sensitive instrumentation has been provided to detect incipient leaks from the tubes to sodium side. The leaks are categorized as small leak, medium leak and large leak. Small leaks are detected by sputter ion pump (SIP) based instrumentation system and safety action taken on crossing the absolute threshold and rate of rise of $H_2$ concentration in sodium. For this, three sputter ion pumps are provided in each loop which initiates safety action with a 2/3 logic. Medium leak is detected by increase in pressure of the expansion tank and large leak by bursting of rupture discs provided at the inlet and outlet of steam generator headers. Safety action is initiated by the signals from these systems. In addition to these, electrochemical hydrogen meter and hydrogen in argon detection by thermal conductivity detectors (TCD) are also provided, which give indication about increase in $H_2$ concentration in sodium and argon cover gas respectively.

On 7$^{th}$ October 2016, during 25$^{th}$ irradiation campaign at 27.3 MWt power, reactor tripped on tube leak in the western steam generator. The triplicated highly sensitive sputter ion pump based steam generator leak detection system (SGLDS) detected the tube leak in the incipient stage itself. Signals from two detectors crossed the threshold initiating safety action automatically. After reactor trip, the steam generator modules in the affected west loop were put on "Safe Configuration" by isolating them from steam/water side, depressurizing the same and injecting nitrogen to the tube side to keep it inerted. Sodium from the loop was drained subsequently.

Analysis of the cover gas in the expansion tank showed the concentration of $H_2$ as 5% and the plugging temperature of dumped sodium in the west loop was found to be 112 °C, as against the normal value of <105 °C. As any increase in hydrazine content in feed water and any oil leak from sodium pump also could cause increase in $H_2$ concentration, the Content of hydrazine in the feed water was measured and found to be normal and the level of pump oil was found to be steady. From these observations, it was been concluded that there was a genuine water/steam leak from one of the west loop steam generator modules into sodium. Conservative estimates based on the $H_2$ accumulation and the sodium plugging temperature in the west loop has indicated that the magnitude of leak is about 0.9 g/s. As the two steam generator modules (SGna 600A &SGna 600B) remain interconnected at sodium side and steam/water side in the loop and there are no isolation valves, identification of the leaky module was a big challenge. This was compounded by the fact that the quantum of leak was very minor, which occurred at high pressure (125 bar) and temperature (460 °C) and also that these conditions could not be recreated again for identifying the leak. Hence, a novel gas tracing technique was employed to identify the leaky module. Helium and argon gases at 40 bar pressure were admitted into SGna 600A and SGna 600B respectively. The shell side of the modules was sampled for the presence of helium. As no helium could be detected, the gases in the two modules were reversed. With helium in SGna 600B and argon in SGna 600A, presence of helium was detected in the shell side of SGna 600B. At the end of six hours, helium concentration was found to be 1042 ppm indicating that SGna 600B is leaking and the order of leak being very minute.

The failure of steam generator has been a first-of-the kind experience for FBTR and the replacement of the leaky module called for elaborate activities viz. cutting of water/steam/sodium headers, sodium cleaning/safe disposal and facilitation of operations including maintaining the system in inert atmosphere during the interventions, erection of massive scaffoldings inside and outside the steam generator casing, handling of structures like carrying beam, removal of hot beams, supporting steam generator modules to facilitate removal of the common support beams, modification of spare steam generator module to introduce welded orifice assemblies, requalification of the preserved spare steam generator module by helium leak testing of shell welds and tube side, removal of leaky module from the steam generator casing, introduction of spare module including positioning and alignment with common water header, steam header and sodium inlet/outlet headers, welding the joints, post-weld heat treatment of weld joints and qualification by liquid penetrant inspection, radiography, helium leak testing and finally by hydro testing of the tube side.

As FBTR is under the regulatory control of AERB, the incident, its consequences and the restoration plans were notified to AERB and approval was obtained. The requalification of the spare module fabricated more than thirty years ago was completed.

A special task force including personnel from operations, maintenance, technical, quality assurance and industrial safety sections was constituted to execute the task. The task force made elaborate and meticulous planning, split major activities into smaller segments, entrusted each of them with small groups, integrated the split activities, did mock-up of all major activities, qualified the crane for handling stresses by load tests.

The most difficult part of the entire programme was the handling of the slender steam generator module of 200 mm diameter with overall dimensions (15 metre (L) x 6 metre (H)) and a weight of six tonnes inside steam generator casing, with narrow working space and absence of access by crane. Lifting of the module and movement of the module had to be done manually with utmost care and precision. The removal of the leaky module and the introduction of spare module were done by moving very slowly (inch by inch) and each operation lasted for about 8 to 9 hours.

Highest industrial safety standards were followed during the execution of work which spanned over more than 10 weeks and there was no accident or man-hour loss. Timely help from other agencies like Madras Regional Purchase Unit and Quality Assurance Division expedited the schedule. The entire procedure could be completed in a record time of two months against the original plan of four months. The systems were normalized and after observing the operating parameters of the steam generator, the reactor was raised to the target power of 27.3 MWt.

Completion of the critical SG replacement in the shortest possible time and subsequent resumption of reactor operation received the appreciation from Director, IGCSC & Chairman, AEC.

## Extracts from the MOM of IGCSC (Feb'2017)

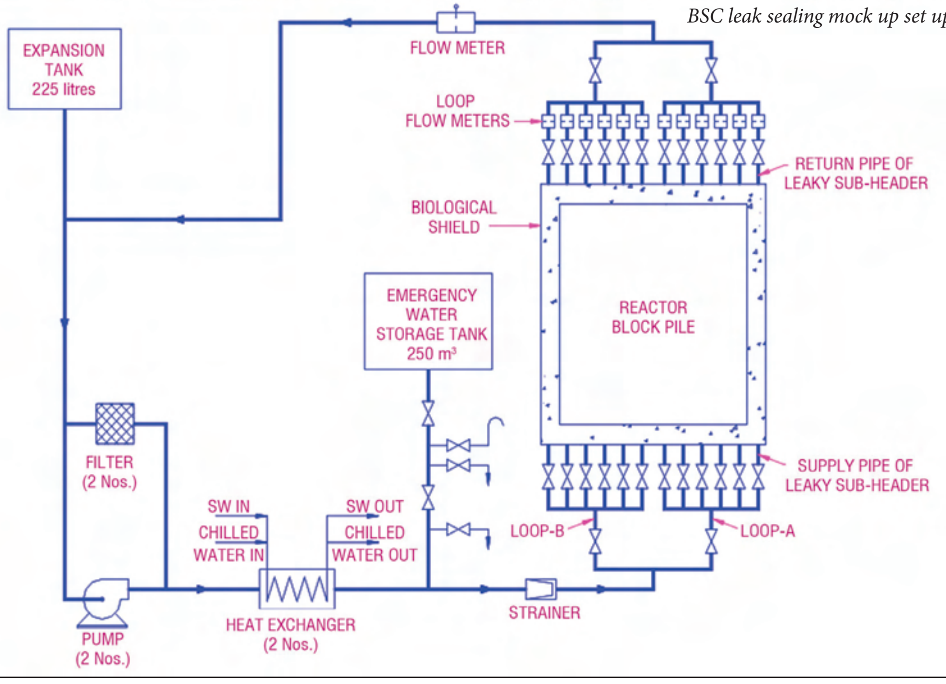
Group Director, RFG, informed the members that FBTR Steam Generator has been replaced within a short span of two months. Chairman informed the members that Secretary, DAE, conveyed his appreciation for completing the task in such a short span of time as it was expected to take much more time for replacement, IGCSC placed on record its appreciation for completing the task much ahead of anticipated schedule. Chairman said that with the replacement of steam generator FBTR should run on full power.

*SG replacement*

## 4.4 Water Leak from Cooling Coils Inside Biological Shield Concrete

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 $60^0$ sector in the southwest portion of the biological shield 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 (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 by chemical sealing 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. However, after one year, leaks of small magnitude started appearing in some of the coils. During October, 2005, the leak rate 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 tubes 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

*BSC leak sealing mock up set up*




*BSC circuit*

availability and our confidence in arresting such leaks in future. This development work received Group achievement award from DAE.

**4.5 Experience with Handling Sodium**

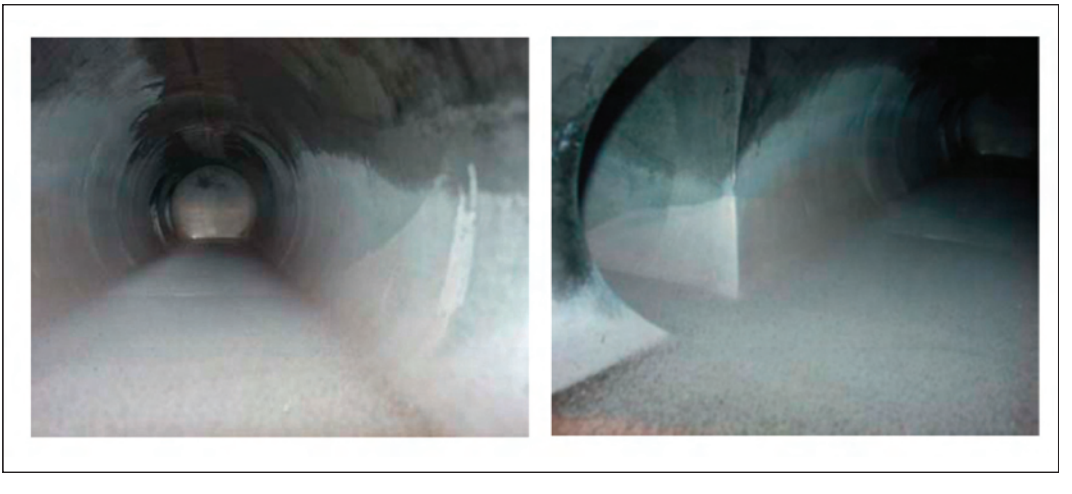
Sodium systems have been operating for the past thirty six years and their performance had been excellent except a few leak incidents in sodium bellows seal valves and one major leak of 75 kg from primary purification system into its inerted metallic cabin. The sodium purity has been maintained consistently well for the past three and a half decades by the cold-trapping of oxides (impurity level < 0.6 ppm). The four sodium pumps have logged trouble free, cumulative operation of 9,61,152 h. There have been no incidents of oil leak from the pump seals into the sodium circuit so far. The steam generators which operate with high pressure steam / water on the tube side and sodium on the shell side have operated for more than 44,425 h. There was only one incident of minor sodium-water reaction and one minor incident of sodium leak from the thermal baffle of SG. The primary cover gas purity has been maintained within safe limits.

### 4.6 Replacement of Rupture Disc in SG

Rupture Disc Assemblies are provided in the sodium inlet/ outlet headers of steam generators and in expansion tank cover gas space in the secondary sodium system. These safety devices protect the system and components from over pressurization from any sodium-water reaction in the SG. The existing rupture disc assemblies are of knife edge type developed in-house through an elaborate fabrication and testing programme. Thanks to good fabrication standards, there has been no major water leak from SG and rupture discs were not called to play their role. However, as a good operation practice, it was required by safety bodies to take out one set of assemblies and test them for any degradation. Since spare rupture discs were not immediately available, it was decided to replace the rupture disc assembly in the west loop by state-of-the-art, scored type rupture discs, designed for rupture at 8.2 $kg/cm^2$. These were developed indigenously and qualified after extensive testing.

The expansion joint in the cyclone separator side of the rupture disc was dismantled. Special handling fixture was made for handling the rupture disc assembly. The existing rupture disc assembly was cut and removed. The sodium inlet & outlet headers were found partially filled with sodium as there is no provision for draining the sodium fully from the headers. The headers were completely cleaned of sodium (≈30 kg) with special tools. Special seal plugs were used for sealing the opening to prevent air ingress during the various operations. The new assemblies were welded. The weld joints of all the rupture disc assemblies were qualified by Gamma radiography and argon sniffing.

After replacing the rupture discs, the system was filled with sodium. The plugging temperature was only 120 °C, giving testimony to the extent of sodium cleaning and care taken to avoid entry of air & moisture into the system during the work. The removed rupture discs were then cleaned to remove sodium deposits, by immersing them in hot thermofluid oil. Both the assemblies were then tested at room temperature to find out their rupturing pressure using a separate setup. The rupturing pressure was found to be within the design bursting pressure range indicating no degradation over the years.



*SG headers with frozen sodium before removal by scooping as part of RD replacement*

*Scooping of frozen sodium from the SG inlet header*

Secondary West loop was in drained state for 87 days for the expansion tank & SG rupture discs replacement.

**5.0 Plant improvements**

**5.1 Online Sodium Purity Monitoring System**

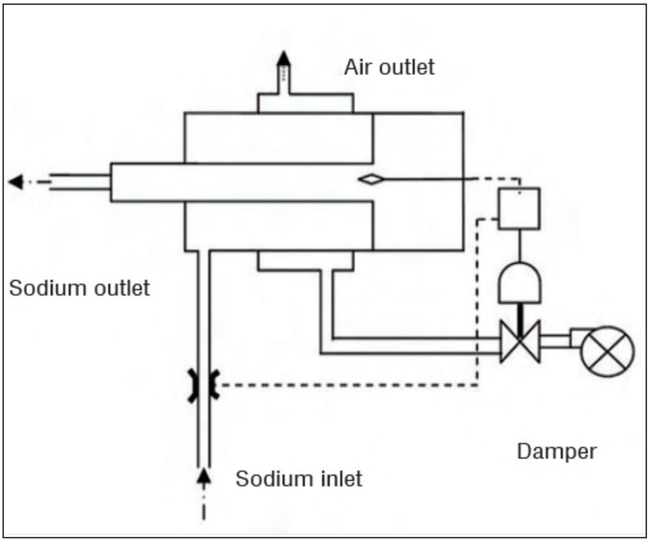
The process of accurate estimation of the impurities present in the sodium has been automated using a closed loop controller.

In the manual process of measuring sodium impurity using plugging indicator, the operator has to watch the temperature and adjust the damper position i.e. cooling air flow rate accordingly. When the sodium temperature is high, operator has to switch ON the blower and adjust the damper so that air will be blown to bring the temperature to the desired value. Once it reaches the desired value (20% flow reduction), the damper has to be closed and the blower shall be switched off.

The process has been automated using standalone dedicated PC based system with distributed data acquisition and control module and software based control algorithm.

Data acquisition and control system has been used for acquiring field input signals like temperature and flow. Relay output module is used for ON/OFF control of blower. The damper position is controlled by Analog Output (AO) module. Software based PI control algorithm is operated in the Industrial PC based system.

The system controls the cooling rate of sodium by opening and closing the damper in the air flow path. Faster cooling rate is maintained up to change over temperature and slower cooling rate below the changeover temperature. When the sodium flow drops to about 80% of the normal flow, the




*Plugging indicator*

temperature at which the flow drops is measured as Plugging Temperature. Then the damper is closed and the blower switched off. Due to this, the temperature will increase. Once the temperature reaches initial threshold value, plugging run is repeated i.e. blower is switched ON and the damper is opened for air flow. This is known as cyclic mode.

Human Machine Interface (HMI) is executed through the front end Graphical User Interface (GUI) as shown in the following Figure.

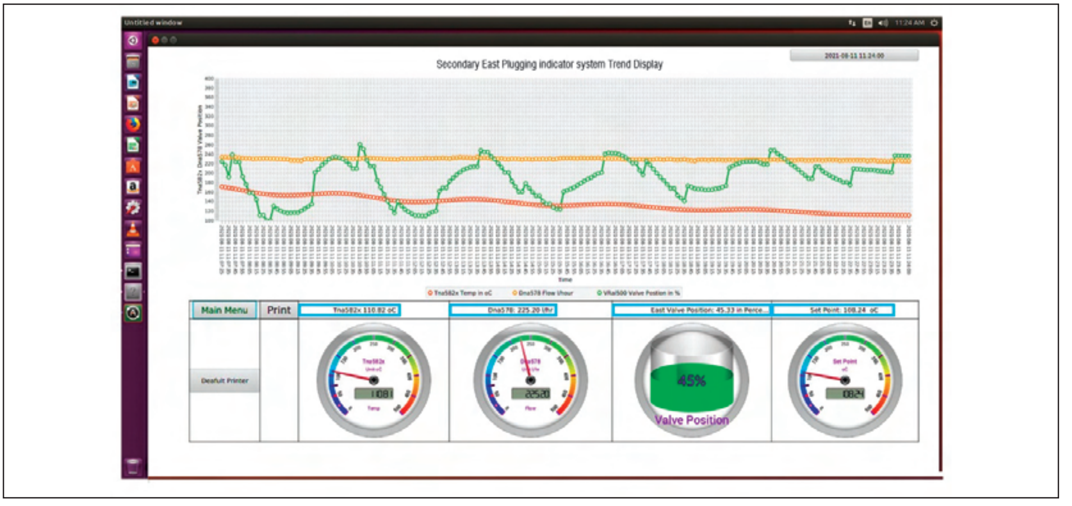
**GUI has The Following Important Features**

i. The instantaneous values of all the analog signals, status of digital input contacts, relay status and damper position is displayed and refreshed at every one second interval.

ii. The history of plugging temperature is stored with time & date for review.

Using the aforesaid methodology, the manual process of measuring the purity of sodium in terms of plugging temperature in primary and secondary circuits of FBTR has been automated. The online


*Graphical User Interface*

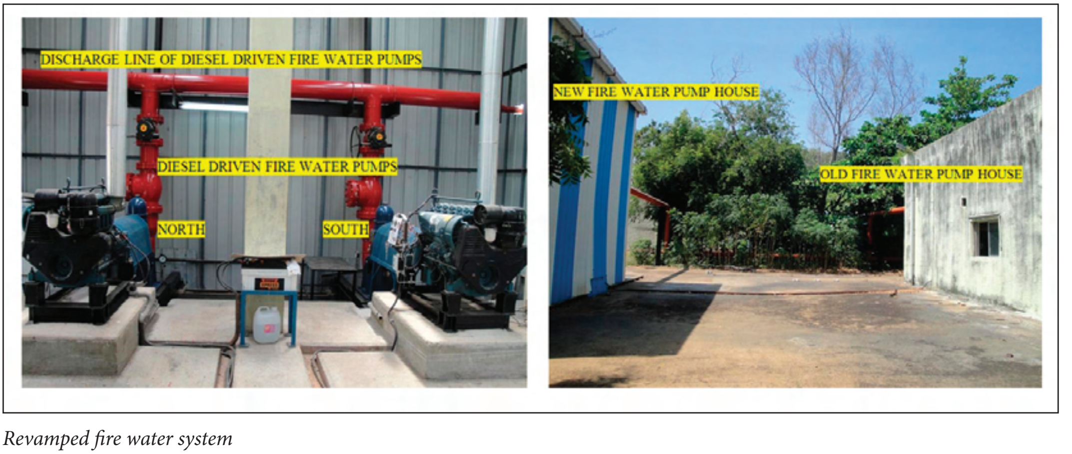



*Online Trend*

sodium purity monitoring system using PI control has been designed, developed & commissioned in FBTR. The performance of the system has been found to be satisfactory. The advantage of this system is that the data acquisition system (DAS) module can be used for distributed environment using ethernet interface.

**5.2 Improvements in Fire Water System:**

Earlier, the fire water requirement of FBTR in the old fire water system was met with two 50% capacity negative suction pumps, one motor driven and the other diesel driven. The capacity of these pumps was 90m$^3$/h at 85 mlc. Whenever any one of the pumps was not available, the RDL fire water system stood as standby source with communication through an isolation valve and interconnecting pipe line. The fire water ring main pressure was maintained at 1kg/cm$^2$ by overriding the domestic water tank. The ring main piping was mostly underground with CI pipe with lead caulked joints, buried at 1.5 m depth. The outside hydrants and internal hose points were connected to the ring main through




*Revamped fire water system*

*Motor Driven water system*

individual isolation valves. The underground header formed a single ring around FBTR-RML complex without any isolation valve. Corrosion, ageing, frequent pressurization & depressurization induced fatigue failure, joint leaks, pipe line cracks etc. were taking a toll on the system resulting in non-availability of the system for prolonged periods. Localization of the leak in the buried portion posed a major challenge. In order to reduce fatigue cycles, a Jockey pump with an accumulator was incorporated in the system for keeping the line pressurized all the time near the discharge pressure of the pump. Starting/stopping of the jockey pump became very frequent due to unidentifiable leaks in the piping system. The foot valves of the pumps were found cracking due to high pressure as suction line of the pumps was getting pressurized through the bypass communication provided across the pump NRVs.

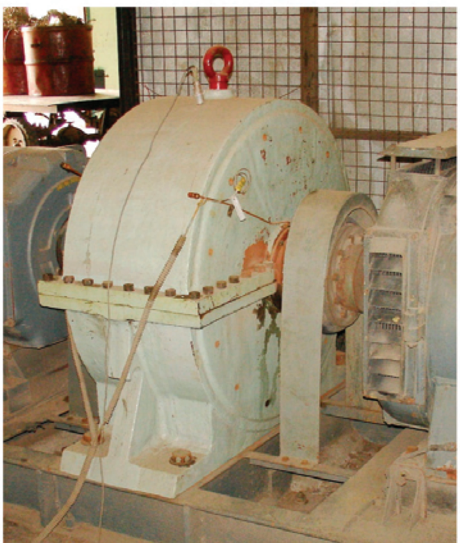
In order to improve the availability of the fire water system, a decision was taken in 2006 to totally revamp the system with submersible pumps and fully welded overhead piping system. The new system has three 50% capacity submersible vertical turbine pumps out of which two are diesel engine driven and one motor is driven. The system has been designed as per IS 1648 and AERB standard specifications. The Carbon steel piping system has been made above ground with isolation valves for segment wise isolation and a major portion of the piping has been routed through the terrace of FBTR-RML complex. The piping at road crossings is laid underground through RCC trenches. A new pump house has been constructed to house both diesel driven pumps. The motor driven pump and two jockey pumps with the accumulators are located in the old pump house.

The discharge lines from all the three pumps join together and form a 250 NB common discharge header. Two 200 NB pipe branches tapped from this common header run in opposite direction and encircle the FBTR –RML complex to form the ring main. Five isolation valves are provided, one each in two branch lines, one in the interconnecting line and two on the terrace (one in the Service building and another in the Control building). With this arrangement, the ring main forms three segments. For any maintenance involving a particular segment, other two segments will be available. From the ring main, five 150 NB lines are branching out for providing supply to fire hydrants and hose points in Turbine building, Active building & Maintenance building and to provide supply to Diesel Generators in service building and Mulsifyre system for transformers. There are 15 fire hydrants in and around FBTR-RML complex and all are provided with landing valves. Venting provision with 50 NB ball valves are provided at seven locations out of which six are in the terrace. Two drain valves are also provided to facilitate draining of the piping segments. An interconnection with two isolation valves in series has been provided with IGCAR Zone II fire water system which acts as a standby for FBTR fire water system.

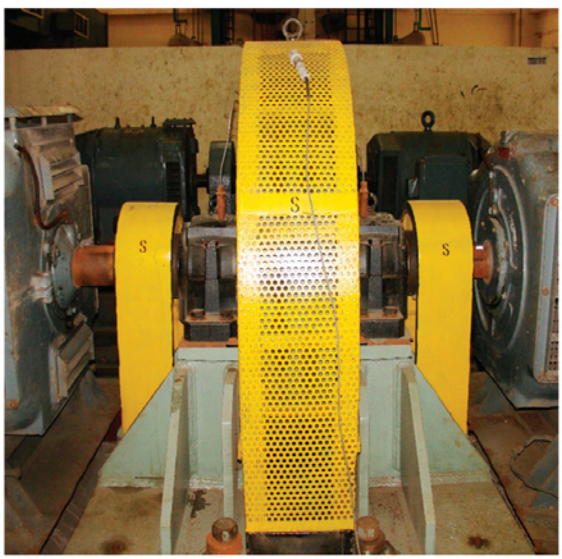
**5.3. Ward Leonard Flywheel Housing Modification**

During the last several years, the flywheel assemblies of Ward-Loenard drives (4 nos) were experiencing high vibration as well as high bearing temperature. Damage to bearings was very frequent. Many times, due to deficiency in these drives, reactor had to be shut down for rectification works. The corrective actions like strengthening the base frames, correction of out of concentricity of bores, changing the bearings with different type etc. were of not much help. A complete relook at the present design of the flywheel assembly revealed several inherent deficiencies. The tower like design was found to be not stiff in both radial and axial direction. Due to this, any small disturbance / minor defect was getting amplified causing high vibration leading to bearing failures and consequent shutting down of the equipment.

In order to address the above, it has been decided to have a welded steel construction flywheel housing with Plummer blocks which can withstand overloads and shock loads. This design gives lot of flexibility in tackling the vibration issues since it gives good rigidity and strength.



*Old flywheel assembly*



*Modified flywheel assembly*

The requirement of increased stiffness in both radial and axial direction to reduce the vibration level, ease of replacement of bearing & its housing, a fully ventilated flywheel cover to facilitate more air circulation and prevention of vibration transfer from drive end to non-drive end were considered during design.

The existing design has a common base frame for induction motor, flywheel assembly and generator. Any change in length of the flywheel housing requires complete replacement of base frame and relocation of the equipment. Hence, it was decided to keep the overall dimension as same as the old housing. In the new design, the housing (drive end and non-drive end) was separated by using independent Plummer block housings to reduce the cross talk. The higher width of the Plummer block housing posed a big challenge for maintaining the overall dimensions of the housing. In the new design, the tyre type coupling was replaced with pin and bush type coupling to prevent axial excitation due to presence of tyre coupling on the drive end and non-drive end. The welded steel construction flywheel housing with Plummer block was designed in-house at FBTR considering the above factors and fabricated.

During fabrication of newly designed welded housing, as the plate thickness was more, preheating was done to prevent distortion and crack due to faster cooling of weld metal. Ribbon type heaters of adequate rating were used for preheating. The base plate and pedestal surfaces were machined to achieve uniform level. The flywheel rotor assembly was kept over the Plummer block housing and top housing was assembled. The fabrication was done after ensuring the design clearances.

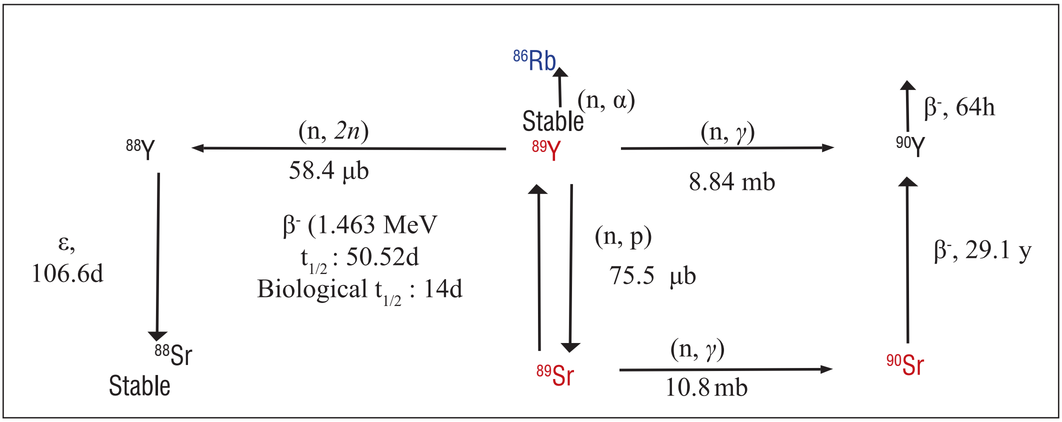
The existing flywheel assembly was replaced with the new design flywheel assembly. It was aligned with induction motor and dc generator. The initial trial run was smooth with low vibration and bearing temperature. After satisfactory trials without load, the drive with flywheel assembly was loaded by starting the pump. In the loaded condition also, the performance was found excellent with very low vibration levels and low bearing temperatures (max. 47 °C).

Based on the successful running of prototype design, all remaining Ward-Loenard flywheel housings were replaced with modified design housing. During the 30$^{th}$ campaign at 40 MWt, the performance was found excellent with very low vibration levels (max. 3mm/sec) and low bearing temperatures (max. 50 °C). The Ward-Leonard flywheel drives were running continuously for the past 2 years without any downtime validating the successful design development and fabrication by FBTR.

## 6.0 Societal Application of FBTR

### 6.1 Production of Strontium- 89

The most common pain syndrome encountered in cancer patients is metastatic bone pain. It is seen in up to 70% of patients with prostrate and breast cancer, and up to 30% of patients with lung, bladder and thyroid cancers. In addition to pain, common complications include skeletal fractures, hypocalcaemia and spinal cord or nerve root compression, all of which affect mobility and sleep, greatly reducing the patient's quality of life. Management of bone pain includes analgesia,

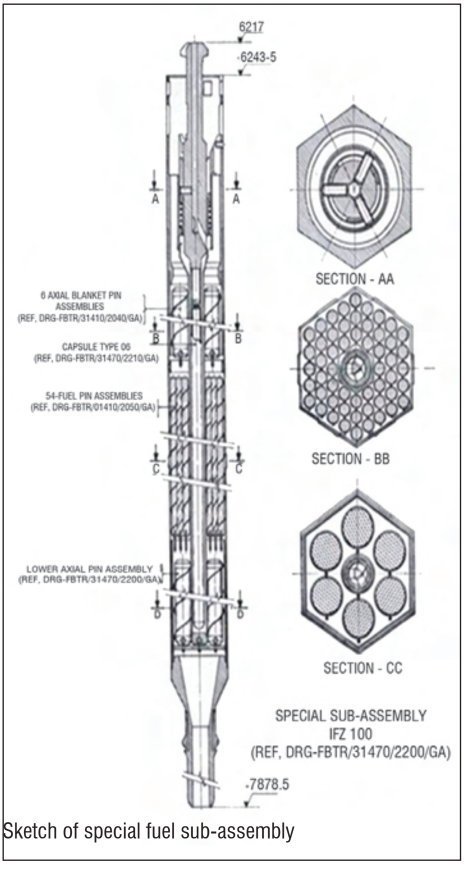



*Production of $^{89}Sr$*

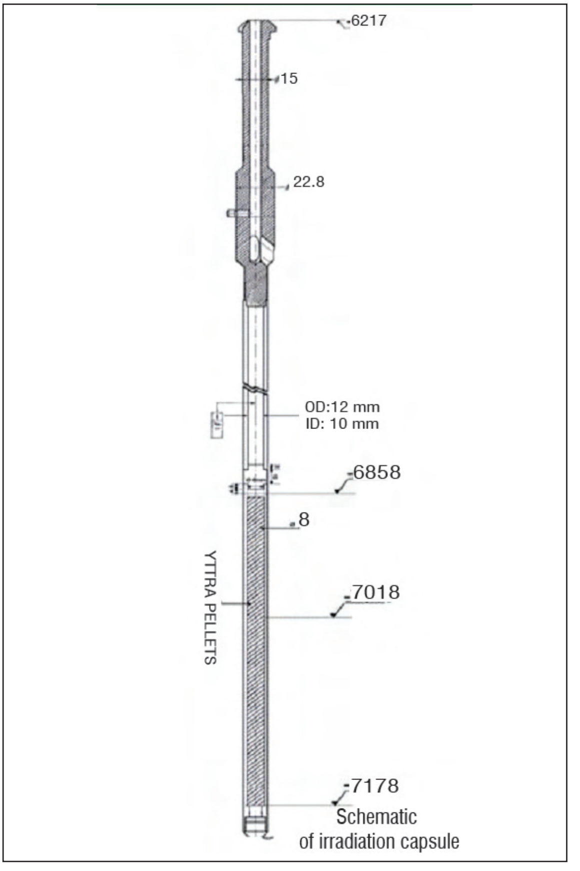
radiotherapy, radio frequency (RF) ablation, hormones, chemotherapy and surgery. Radiotherapy uses bone seeking isotopes like phosphorous ($^{32}P$, $^{33}P$), strontium ($^{89}Sr$), etc.

Strontium is an element that behaves biologically like calcium. Strontium-89 chloride localizes selectively in bone, especially bone cells that are rapidly dividing (such as areas with bone metastases that are causing the pain). It remains in the bone for several weeks providing pain relief – mean duration of relief is about six months, but can last up to fourteen months. The radiation emitted is absorbed almost completely within this area, maximizing the efficacy of the treatment. The usual therapeutic dose is 148 MBq (4 mCi). The compound is currently marketed under the brand name “Metastron” in USA and Canada and costs about USD 3000 per dose.

*Capsule containing Yttria pellets*




*Special SA containing Yttria capsule*




*Capsule containing Yttria pellets*

Preliminary experiments have been carried out in Fast Breeder Test Reactor to determine the feasibility of producing $^{89}Sr$ and are described below.

$^{89}Sr$ is mainly produced using the (n,γ) reaction on enriched strontium ($^{88}Sr$>99.9%) targets in thermal reactors. Enriched targets are necessary to avoid the formation of $^{85}Sr$ – an undesirable impurity in the present context – and to increase the yield of $^{89}Sr$. This radionuclide is also produced in fast reactors using the (n,p) reaction on $^{89}Y$. One of the major advantages of using the (n,p) reaction is that the element produced is different from the starting element enabling to chemically separate the desired radionuclide with very high specific activities. In fast reactors, specific activities of about 19 kCi $^{89}Sr$ per gram of strontium have been achieved vis-a-vis about 10 Ci/g strontium in thermal reactors. However, the penalty paid lies in the more complex procedures required for the introduction and removal of samples from the sodium covered fast reactor pile, formation of undesirable radionuclide due to other reactions {e.g. $^{88}Y$ due to the $^{89}Y$(n,2n) reaction} and complex chemical separation procedures.

Natural yttrium – as yttria, $Y_2O_3$ – was used as the starting material for the production of $^{89}Sr$. Yttrium has only one stable isotope $^{89}Y$, and the various nuclear reactions take place when this nuclide is irradiated in a fast flux. $^{89}Y$ undergoes the (n,p) reaction to form the nucleus of interest, $^{89}Sr$. The

$^{89}Sr$ formed also under goes neutron capture to form $^{90}Sr$ and also decays by negatron decay to the starting nucleus, $^{89}Y$.

The yttria pellets were obtained by compacting the powder either with or without a binder like zinc Oxide. The pellets were sintered at 1600 °C for five hours and then loaded into the irradiation capsule in an inert atmosphere glove box and sealed by welding. The length of the pellet column was 300 mm and the column was arranged to be in line with the fuel column of the fuel pins in the core. The Outer diameter of the irradiation capsule was 12 mm and the inner diameter 10 mm. The yttria pellets had a diameter of about 8 mm to facilitate easy loading and removal from the irradiation capsule.

| Radio nuclide yields in FBTR core centre and BOR-60 reactor – 30 days irradiation campaign | | |
|---|---|---|
| S.NO | Irradiation Position | Radionuclide yield (Ci/g) |
| 1 | $^{89}Sr$ yield in FBTR | 0.0113 |
| 2 | $^{89}Sr$ yield in BOR-60 | 0.01 |
| 3 | $^{88}Y$ yield in FBTR | 0.0046 |
| 4 | $^{88}Y$ yield in BOR-60 | 0.003 |
| 5 | $^{90}Sr$ yield in FBTR | 1.98E-09 |

The irradiation capsule is then locked in a special fuel sub- assembly, IFZ100 and loaded into the reactor. The irradiation capsule can be inserted into the tube sheath and locked. A fine hole is provided at the bottom of the tube sheath for draining the sodium when the capsule is taken out after irradiation. There will be a trickle flow of sodium around the capsule (i.e. in the annular space between the irradiation capsule and the tube sheath) but for computing the temperature of the pellets, this flow is neglected and it is assumed that the sodium in the annular space is stagnant.

**7.0 Life Assessment of FBTR**

In general, reactor life is governed by replaceable and non- replaceable components, whereas the ageing factors are based on corrosion, creep-fatigue interaction, loss of ductility, etc. All components were design checked for 2000 cycles of operation with an inlet temperature of 380˚C and outlet of 600˚C. Various up-gradation activities for improving the safety and plant life extension were carried out over recent years.

In FBTR, regular surveillance and preventive maintenance programmes are in place for condition monitoring and checking performance of various SSCs. Performance of each system is assessed separately and reviewed annually. Components which are accessible and replaceable are either over-hauled or replaced based on the results of surveillance and/or preventive maintenance. The Technical Support team keeps a tab on the technology changes and the SSCs facing obsolescence are upgraded systematically. The SSCs are also upgraded / modified to increase the work safety. No particular life is allotted to the SSCs in conventional systems as they are easily replaceable as and when required.

By Periodic Safety Review (PSR), residual life of plant is assessed systematically and license for reactor operation is obtained from the Atomic Energy Regulatory Board (AERB), India. Application for Renewal of License (ARL) was submitted to AERB in Jan'18 and based on the review, the license to operate FBTR reactor was extended till June 2023.

For SSCs which are non-replaceable or replaceable with great difficulty, life assessment is performed periodically to ascertain the remaining life. The health of these SSCs is maintained and monitored with the help of the life management practices followed. The main life governing factors for SSCs in primary sodium system are creep-fatigue damage, thermal cycling and irradiation damage.

The current Life extension practices followed in FBTR consist of:

1. Measures to mitigate & control the ageing mechanisms by Good operation practices.
2. Monitoring the failure of any equipment at the incipient stage by on-line monitoring of the functional integrity of component.
3. Monitoring & trending the ageing effects by in-service inspection.

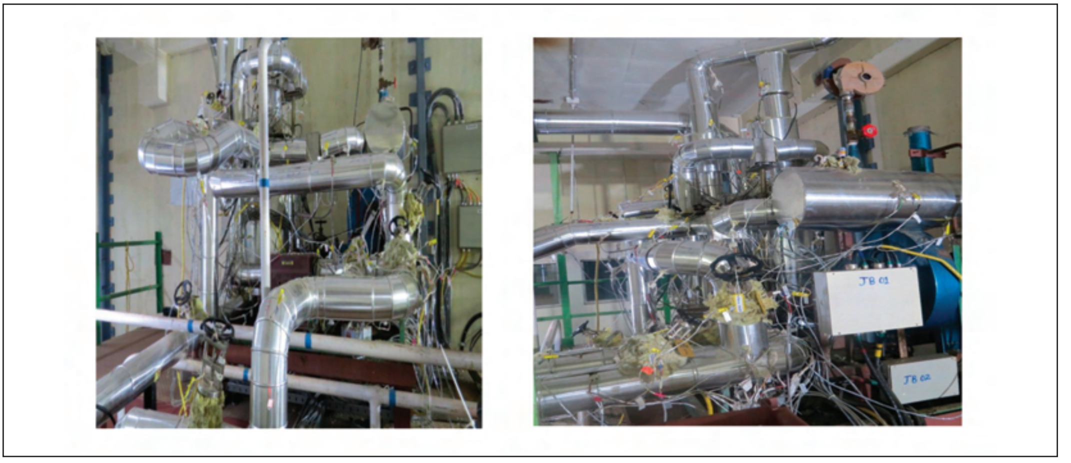
Regarding the non-replaceable components, residual life assessment has been carried out based on the operational history vis-à-vis the design limits for each component. The major limiting factor found in the life assessment studies has been the fluence on the grid plate which supports the core (based on the 10% residual ductility criterion for the guide plate) and the thermal cycling of the cold junction of the Clad Rupture Detection (CRD) circuit's downstream pipeline. It is planned to introduce Tungsten carbide as bottom axial shield in the fuel subassemblies to reduce the fluence on the grid plate by 30% & thereby extending the life of reactor by 30% of the remaining life.

The life of nuclear components is being monitored by periodic surveillance. Visual inspection of the reactor vessel and thermal shields is done every two years using periscope and projector. All the accessible surfaces have been found to be normal. Sodium deposits have been seen in the cooler regions at the top, but these are normal for sodium systems.

Life assessment for all the important components in the heat transport circuit has been done and it was found that the creep-fatigue damage is insignificant as the reactor has operated at lesser power than designed for many years.

The total number of high temperature operation is ~ 48000 hours as against the design life of 100000 hours. Hence creep damage to the high temperature components is well within the limits. So far, the significant thermal cycles are only 177 at a maximum differential temperature of 91°C and hot leg temperature of 482°C. Life based on cumulative creep-fatigue damage has also been established in the design and found to meet the above creep life and fatigue cycles for all the components.

Irradiation induced changes in the mechanical properties of grid plate material is one of the factors considered for estimating the remaining life of FBTR. The main factor responsible for limiting the life of FBTR is the radiation damage to the grid plate which supports the core. The fast flux at the grid plate has been measured using Np foils. The residual life of the grid plate is governed by the neutron fluence or dpa corresponding to 10% uniform elongation. This limit will be reached in 6.32 Effective Full Power Years (EFPY). With a plant availability factor of 50%, the reactor can be operated up to 2032 or beyond. To make a practical estimate, irradiation of SS316 material of FBTR grid plate



*Flooding sodium purification circuit*

quality was carried out during the 25th and 26th irradiation campaigns for 95 effective full power days (EFPDs) to attain 6.75 dpa at the grid plate location where the temperature is ~400˚C. Considering the higher rate of loss of ductility of grid plate during lower (340ºC) operating temperature of FBTR till 0.78 dpa, the limiting residual ductility of 10% uniform elongation is attained at 6.3 dpa for FBTR grid plate. This information confirms that FBTR can be operated even beyond 2032.

**8.0 Modification works**

**8.1 Flooding Sodium Purification System (FSPS)**

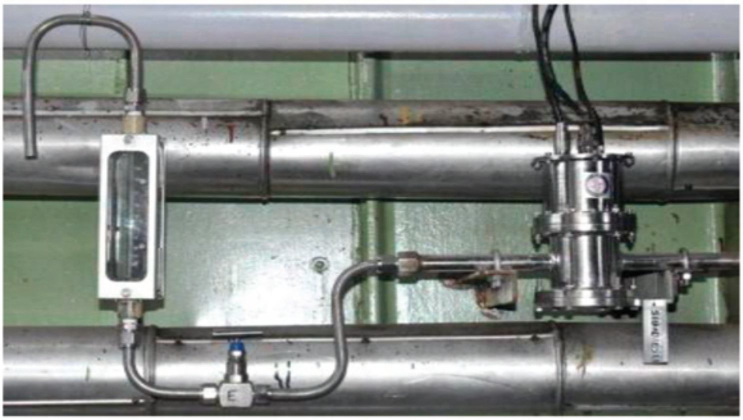
The sodium inventory in Sodium Flooding system is 65.5 $m^3$ out of which 64 $m^3$ is in the two manual flooding tanks and 1.5 $m^3$ in auto Flooding tank. The sodium after the micro filter purification was stored in tanks and was not purified so far. The sodium in the two manual flooding tanks is maintained at 150±20ºC and is stagnant for more than 30 years. The impurity level of sodium in any of the flooding tanks is not known as there was no provision in the system for taking a sample or plugging run. Hence in order to purify the sodium in the two manual flooding tanks, a mobile purification circuit is commissioned with plugging indicator.

After commissioning the purification system, plugging temperature was measured for the sodium in the manual flooding tanks and was found around 140° C. Then, entire inventory of sodium in both manual flooding tanks was purified till the plugging temperature came below 105 °C.

**8.2 Global Sodium Leak Detection System (G-SID) in Secondary Sodium System**

Design provisions exist to contain sodium leak and leak detectors are provided to detect the leak. For detecting sodium leak, wire type detectors are wound over sodium pipelines, spark plug type detectors are used for bellows sealed valves and double enveloped pipes, Mutual inductance (MI) probes are used to detect failure of nickel tube in Steam Generator Leak Detection System (SGLDS) and Sodium Ionization Detector (SID) is used for detection of sodium leak in SG casing. To provide diverse detection of sodium leak at the potentially susceptible leaky points, Global Sodium Leak Detection System

(GSLDS) was installed and commissioned in secondary sodium system. The basic intent of GSLDS is to detect sodium leak from remotely operated (motorized/pneumatic/hydraulic) bellows seal valves in the secondary sodium system.



*Global SID System*

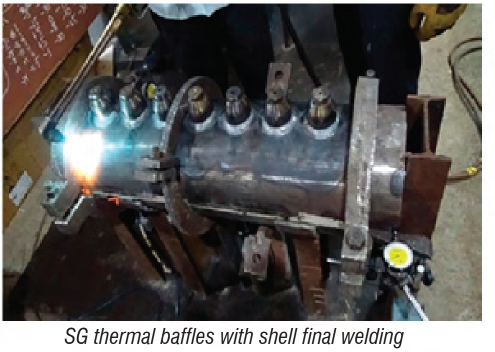
**8.3 Repair and Restoration of Steam Generator Module having Defective Thermal Baffle**

Steam Generator (SG) of FBTR is a once through shell and tube type counter flow heat exchanger. The sodium flows in the shell side and water/steam in the tube side. Since the SGs do not have tube sheets at both ends, thermal baffles are used at the entry and exit of tubes in the sodium inlet and outlet headers. There was a sodium leak from one of the thermal baffles. The leak was found to

*SG sodium inlet header shell prefabrication completed*

*Assembling of stiffener arrangements for SG baffle with shell welding*

*SG thermal baffles with shell root welding completed*



*SG thermal baffles with shell final welding and heating completed*

*Repair and restoration of SG model having defective thermal baffle* be from the Heat Affected Zone (HAZ) of the weld joint between the tube and the baffle. In-situ metallography of the affected portion indicated subsurface flaw in the material which has caused the failure. Considering the critical function of the thermal baffles which encounter hot sodium at 515 °C, it was decided to replace all the thermal baffles with fresh ones. The defective steam generator was removed from the system and kept on a support structure in the Steam Generator building. After scooping out the left over sodium from the sodium inlet and outlet headers, the headers were thoroughly cleaned with alcohol. The shell side was kept under inert conditions with argon after sufficient flushing. As replacement of the baffles in the SG sodium inlet header alone was not practically possible, it was decided to replace the entire sodium inlet header with prefabricated components. The procedure for baffle replacement and Quality Assurance Plan (QAP) were prepared and approval was obtained for execution. The raw material (2.25Cr-1Mo) for sodium inlet header replacement was qualified by chemical analysis, mechanical testing and ultrasonic testing (UT) of the plates. New thermal baffles (7 No.) made of 9Cr-1Mo were machined for assembling in the steam generator. Procedure qualification and welder qualification for 9Cr1Mo with 2.25Cr-1Mo dissimilar welding was carried out successfully. The water tubes and the thermal baffles were cut for removing the header. The longitudinal weld joint and circular weld joint of the sodium inlet header were cut and the header was removed as two halves. For fabricating the new sodium inlet header, the required 10 mm plates were machined out from 16 mm thick plates. The plate was formed for the manufacture of 2 halves of sodium inlet header and heat treatment (normalizing and tempering) was carried out for the header halves. The top half header was drilled for assembling the thermal baffles and qualified by Liquid Penetrant Inspection (LPI). Suitable clamping was made to prevent distortion during welding of thermal baffles to the header half. After ensuring all dimensions, sequential welding of the thermal baffles with header was carried out. The inlet header halves were welded for longitudinal and circular seam to integrate with the steam generator. Suitable measures like clamping were carried out to prevent distortion. The welding of the tube to thermal baffle was made keeping the uniform clearance between them. All the weld joints were successfully qualified by LPI and UT. The restored SG is preserved and kept as a spare.

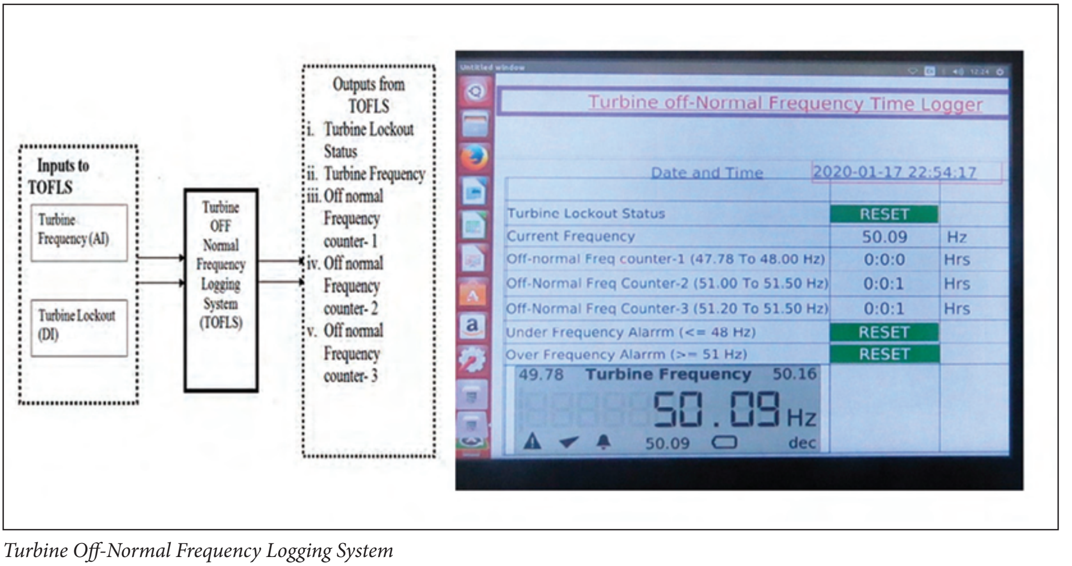
### 8.4 Inspection Windows for SG Casing

In order to visually inspect SG casing whenever a leak detector is actuated without opening the top man hole cover or trap doors, three high temperature quartz glass (rated for 1000 °C) viewing windows were installed on SG casing at suitable locations.

*Viewing window of SG casing*

## 8.5 In-house Development of Off-Normal Frequency Logging System for FBTR TG Set

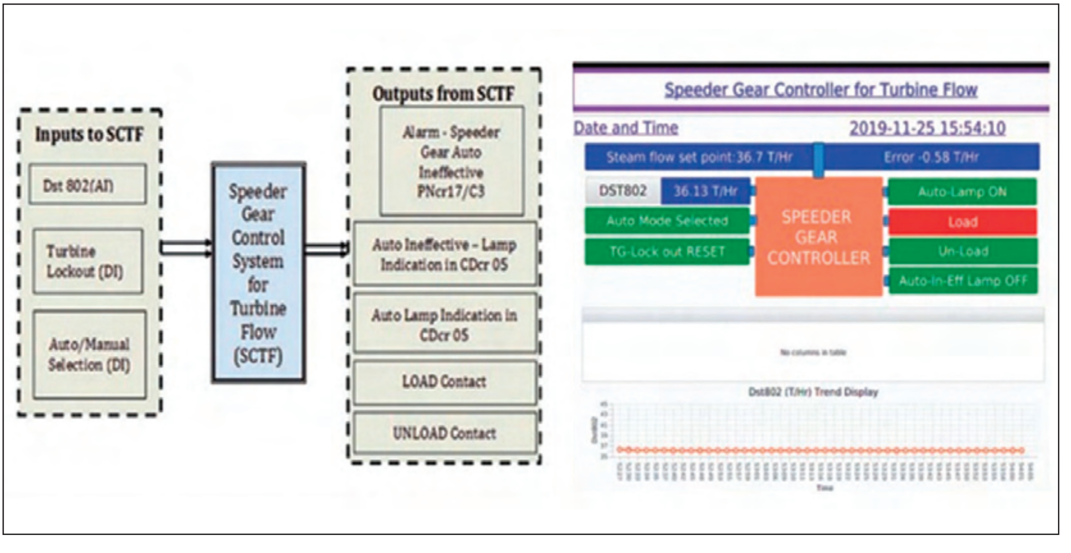
The safe operation frequency range of FBTR Turbo-Generator (TG) set is 47.78 Hz to 51.5 Hz. Low frequency operation stresses turbine blade, causes blade resonance, shortens blade life and even makes blade break. Over frequency will lead to high vibration and stress accompanied by reduction of fatigue life leading to eventual failure. So there is a need to monitor and log the cumulative off-normal frequency operation to decide on turbine internal inspection periodicity. For this purpose, Turbine Off-normal Frequency Logging System (TOFLS) has been developed in-house. The TOFLS is a Smart Touch panel PC based computer system along with a Multi-Function USB DAQ module. The block diagram of TOFLS system is shown in the following figure. Frequency signal of 0 to 10 V DC corresponding to 45 to 55 Hz is generated from alternator Potential Transformer and connected to TOFLS. The TOFLS scans the frequency and Turbine lockout condition at every second. The system initiates logging time of off-normal frequency only when the turbine lockout is RESET and the turbine is rolling. Application software is developed using “C” in Ubuntu environment. The GUI software is developed using JAVAFX. Linux Kernel 3.19.0 with Ubuntu 14.04.3 LTS is used in the Smart Touch panel PC System. The USB Multi-function DAQ module has both analog and digital input channels. This module is connected to the Smart Touch panel PC through USB and it is self powered. The Turbine frequency signal is connected to the USB DAQ module through a unity gain Isolation amplifier. Turbine lockout signal is extended from relay room to Control Desk CDcr 05 and then connected to the DAQ module kept in PNcr 17. The industrial touch panel system is energized with 230 V UPS supply. The TOFLS scanning module scans the TG set frequency signal at one second interval and checks whether the current frequency lies in the Range-I, Range-II or Range-III provided the Turbine lockout is in RESET condition. The current turbine frequency is compared with the ranges of frequency settings. If the frequency lies in Range-I, Range-II or Range-III, then the respective frequency counter will be incremented. The accumulated time for all the




*Turbine Off-Normal Frequency Logging System* three frequency ranges is stored and displayed. The current frequency of TG set, Turbine Lockout status, off-normal frequency counter for Range –I, II and III are displayed in GUI and updated every1 second interval. The TOFLS application software writes the scanned input signal status into a Mysql database at every second and the GUI software reads the scanned information from the Mysql server and displays it in the GUI screen. The system was commissioned successfully and was validated when TG was synchronized during 28th and 29th irradiation campaigns.

### 8.6 Installation and Commissioning of Speeder Gear Auto Control System for FBTR Turbine (SCTF)

In FBTR, nuclear heat from the reactor core is transferred through primary and secondary sodium systems to steam generator to generate superheated steam at 125 bar and 480 °C. The superheated steam is fed to the Turbo Generator (TG) to generate electric power. The TG is synchronized to the grid to export the generated power. Earlier, the steam flow to the TG was being controlled manually using the speeder gear motor. Whenever the grid frequency varies, steam flow to the TG needs to be varied manually to maintain the power. Hence, steam flow to the turbine has to be adjusted manually by operating the speeder gear frequently. This is found to be an additional load on the operator. In order to avoid manual operation and to ensure minimum steam flow through the bypass steam circuit for maintaining the required auxiliary header pressure and maximum steam flow to the turbine, it was proposed to have a Speeder gear auto Control for Turbine Flow (SCTF) for the turbine. In thermal reactors, drum type boilers are used and the Boiler Pressure Control (BPC) System controls the speeder gear of turbine for positioning turbine governor valves to enable turbine follow-reactor type control. In FBTR, Steam Generator (SG) is of once through type and in the absence of a steam drum/ separator, this scheme could not be implemented. Hence, the new scheme is installed. The block diagram of SCTF system is shown in the following figure. SCTF is a smart touch




*SCTF GUI*

panel PC based computer system running on Linux based Ubuntu OS with multi function USB DAQ module connected to it. SCTF System is developed in "C" language in Ubuntu environment. The SCTF system will function only on auto mode. The auto mode is effective only when "Auto/Manual" selector switch provided in CDcr 05 is selected on "Auto" and turbine lockout condition is in RESET condition. SCTF system scans the field inputs at every 10 second interval. Once the system is put on auto mode, the steam flow to the turbine at that instant is considered as the set point and SCTF system will maintain the flow within ±0.5 T/h from the set point. At every scan, the system scans the steam flow and compares it with the set point and error is computed. When the error exceeds +0.5 T/h, "Un-Load" output will be ON for 0.37 s and will remain OFF for next 9.63 s. In the next scan, if the error still persists beyond +0.5 T/h, "Un-Load" output will be again turned ON for 0.37 s and OFF for 9.63 s. This pattern will continue until the error reaches within + 0.5 T/h. This "Un-Load" output will actually reduce the steam flow to the turbine. Similarly, when error is in the negative side, loading of the turbine takes place. At any point of time, if the error exceeds ±10% of set point or if "Turbine Lock out" is SET, the auto operation is made ineffective and an alarm "Speeder Gear Auto Operation Ineffective" is generated and "Auto ineffective" lamp will be ON. At this time, operator has to take the speeder gear control on manual mode and control the turbine flow manually. The system will accept new set point when the mode is again changed from manual to auto. A mimic as shown in Figure has been developed to display the status of various parameters to the operator. The system has been commissioned successfully and was validated

### 8.7 Retrofitting a State-of-the-art Automatic Voltage Regulation System for Turbo-Generator of FBTR

FBTR has a turbo generator (TG) rated for 16.4 MWe. The TG operates at 3000 rpm and is synchronized to grid through 6.6 kV bus when the reactor is in operation. The existing Automatic Voltage Regulator (AVR) of the TG has become aged and obsolete. For continued safe operation of TG set synchronized to grid, it was decided to replace the same with a dual-channel digital control system. The new AVR has two identical and redundant control units. It is a fast-responding, continuous-acting digital AVR with silicon-controlled rectifier (SCR) based full control power stages. The AVR ensures stable operation of the alternator synchronized to the grid for the entire load range. It also ensures very fast recovery of generator voltage following step-changes in load demand. Various protective features are incorporated in the controller to ensure safe operation of the generator. The replacement work was taken up in December 2017 and after installation and commissioning, the AVR was put into service in March 2018. The AVR 'VXB22D' is a complete solid-state device consisting of a microprocessor based controller, a firing circuit and a three phase full-wave full-controlled thyristorized bridge unit. It monitors the alternator's terminal voltage through potential transformer and compares it with the reference set point. Control action is taken based on the voltage difference. The controller output is fed to the firing circuit which controls the firing angle of the thyristors to vary the field current of DC exciter so that alternator terminal voltage is maintained constant. The exciter gets its excitation voltage from any one of the bridge circuits which is in service in auto/manual mode. The

*AVR Controls in Control Room*

armature of the exciter is connected to the alternator field. The field circuit breaker connects the excitation voltage from the output of AVR to the exciter field and also exciter armature output to the alternator field. The AVR controller (DECS 300) has three modes of operation, Automatic Voltage Regulation (AVR), Power Factor Regulation (PFR) and Field Current Regulation (FCR). The dual-channel AVR has a main controller and a standby controller. In case of failure of main controller, the stand-by controller takes over and the transfer is bumpless. The controller has diagnostic features like data logging and event recording and in-built protection facilities. The control unit has a menu-driven HMI with PC-controlled parameter adjustment. Real-time metering, watchdog and loss-of-sensing detection are the main features of the controller. Several improvements were made during pre-commissioning. To avoid turbine trip during generator no-load condition, AVR lockout logic was extended to field breaker trip. Generator breaker status was also incorporated in the AVR trip logic. Power factor mode status and excitation transformer isolator status were incorporated with pre-start check logic to avoid manual errors. After the modifications, commissioning checks on the generator at no-load was carried out. Alternator open circuit characteristics (OCC) were plotted in AVR and FCR modes of the controller. AVR response was checked with varying turbine speeds and found to be satisfactory. Controller tuning was carried out in AVR and FCR modes and signals were analyzed. Local and remote mode operations were checked. During commissioning, alternator was started in FCR/AVR modes, voltage was built up to 6.5 kV at a rate of 1.08 kV/ minute, then matched with grid voltage and synchronized to grid in FCR/AVR modes. And the performance of the controller was evaluated. The step response was checked for each mode of operation. Power Factor mode was chosen and tuned properly. The house-load operation of the FBTR alternator was tested with new automatic voltage regulation system. Turbo-alternator was able to take over the FBTR house load successfully without any disturbance. The load throw-off test was also carried out with alternator load of 6 MWe and 10MWe in the 26th and 30th irradiation campaigns respectively. The performance was found to be excellent. The incorporation of AVR has ensured that the desired PF can be maintained at any power without any manual intervention. The new AVR controller is in service since 26th irradiation campaign.

**8.8 Revamping of Control System of Emergency Power Supply System**

The emergency power supply system of FBTR has two diesel generators of 1 MVA capacity, commissioned in 1982. They are in service as class III power supply source since then, feeding critical loads including the sodium pumps during class IV power supply failure. In the past few years, there were a few deficiencies observed during surveillance checks in the start/stop logic of the diesel generator sets. Also, there were spurious trips due to malfunctioning of the timers and protective relays. The timers, contactors and protective relays used in the system became obsolete and spares were not available. Hence, it was decided to replace the obsolete control panels with new panels with the latest components of improved design without modifying any of the existing logics. This replacement work was taken up in February, 2017 after obtaining due clearances from AERB, both for replacement as well as for planned violation of few clauses of FBTR technical specifications during installation and commissioning of the state-of-the-art system. During revamping, several improvements were carried out. Earlier, the Motor Control Centre (MCC) for auxiliaries and the

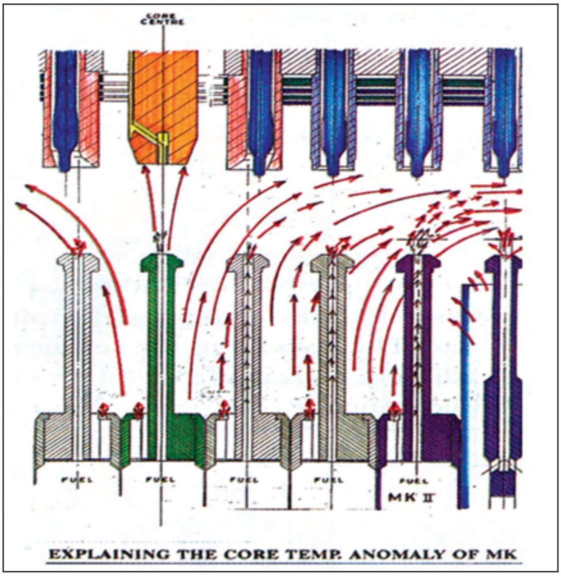
*Starting air circuit DG*

*Control panels of DG*

control panel were mounted side-by-side and also both the diesel generators had their panels mounted on the same cable trench without physical separation. The new panel was designed to have separate MCC and control panel for each diesel generator. The MCCs were mounted on a separate trench on east side and west side for DG#I and DG#II respectively. The control panels were mounted on the same trench with physical separation. This has avoided any common mode failure that may arise from having a common panel. In addition, the old DOLD-make relays were replaced with state-of-the-art numerical relays, which offer several advantages like improved protection, better fault diagnosis, customization of trip settings etc. Speed relays were replaced with non-contact type magnetic pick-ups with features like online testing, ease-of calibration, ability to change the threshold values etc. As per IGCAR Safety Committee recommendation, experience certificate of using numerical relays was obtained from MAPS. The modification was done on one diesel generator at a time. Since the diesel generator under modification was not available for nearly three weeks, availability of additional backup supply was ensured with the 140 kVA mobile diesel generator sets. The works on DG#I were started in March, 2017. After replacement, third party quality assurance checks were done on the wiring and all the alarms and trip settings were tested by simulation of fault conditions. No-load testing of diesel generators was conducted every day for one week and load testing was done once with connected bus load for 5 h. During no-load testing of DG#I, engine tripped on over-speed. Calibration of magnetic pick-up unit was done with variable frequency drive and found to be satisfactory. On further investigation, it was identified that speed sensing was done on an idle gear and not that mounted on the shaft. Hence, an escalation of all the set values by 1.19 times was done to suit the number of teeth in the idle gear to reflect the actual speed of the engine. Further testing was successful. Speed raise/lower, voltage raise/lower and start/stop operation from remote/local were checked. The engine developed the required voltage and frequency within the specified time limit. Load testing was carried out and the diesel generators were put into service. After ensuring satisfactory performance of DG#I, revamping of DG#II was taken up and completed in June, 2017. Both the diesel generators are in service since then.

### 8.9 Eddy Current Flow Meter in FBTR

The core cover plate mechanism (CCPM) has two parts namely a fixed plate housing the thermo wells and a mobile plate housing the sleeves which direct the sodium from the subassemblies for temperature measurement. The mobile plate can be parked in three positions namely 15 mm (distance from SA head) for normal operation, 0 mm for accurate temperature measurement for thermal balance calculation and experimental purpose and 75 mm during fuel handling operation. The mobile plate got stuck in the fuel handling position in 1996. Several investigations were carried out to find out the cause of the seizure but in vain. The seizure location is at the top between the command tube and the fixed outer sheath. The main effect of CCPM in fuel handling position was that it resulted in a plenum hydraulics wherein the hot sodium from the inner rings glided over the cooler sodium from outer rings of the core. Thus, the sub-assemblies in the outer ring (third ring onwards) were reading much higher temperatures than

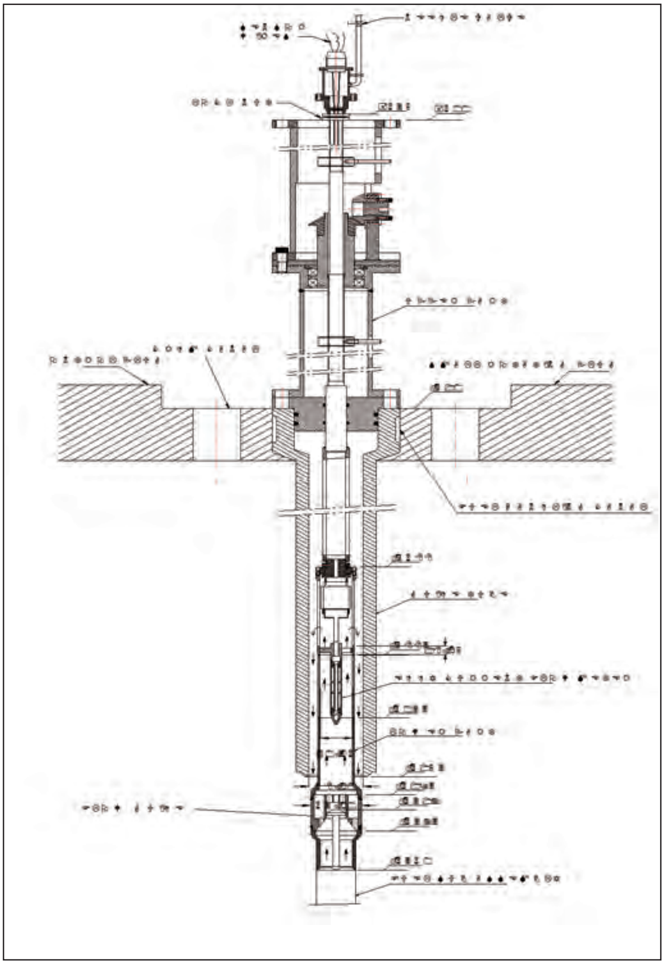



*Schematic of mixing of sodium streams when CCPM at 80 mm*

the actual temperatures. Even with higher readings for outer ring sub-assemblies, theoretical studies indicated that the detectable plugging levels were lower than allowable plugging levels and there was no safety concern. However, the core temperature supervision was supplemented with periodic flow measurement of fuel subassemblies that were showing temperature anomaly during fuel handling state of the reactor. An eddy current flow meter was developed, commissioned and made operational for this purpose with required modifications.

Flow measurement in FBTR was done using a flow measuring device (FMD). It consists of a flow guide fitted with the flow sensor, which can sit exactly over the fuel SA to enable sodium coming out of fuel S/A can pass through the sensor.

Electronic system consists of a constant current drive unit and signal-processing unit to give optimum output. The driver supplies a constant ac current of sinusoidal waveform to the primary coil at a constant frequency. In the signal processor, the signal outputs from the two secondary coils are processed to get the flow output. The two signals are independently amplified, filtered through pass filters, converted in to dc signals Va& Vb. Temperature compensation is also provided.



*Flow Measuring Device*

## 9.0 Safety Upgradation

*Fault tolerant Real time computer systems of FBTR*

FBTR undertook up gradation of systems, components & structures to enhance the safety level based on the operational feedback, maintenance issues and obsolescence. Further, post Fukushima, an extensive retrofitting programme was taken to protect the plant against external events such as flood, Tsunami and seismicity. As per the up gradation programme, several major components have been replaced. These include the Neutronic channels, UPS, computers of the Central Data Processing System, main boiler feed pumps, five control rod drive mechanisms, deaerator lift pumps, reheaters of the steam water system, station batteries, DM plant, Nitrogen plant, starting air system & control panels of the emergency diesel generators, entire fire water system including pumps, isolation dampers of the reactor containment building, chargers of battery banks of primary sodium pump drive systems & AVR of TG. Due to obsolescence, 6.6 kV MOCBs were replaced with VCB and 415 V electro-mechanical relays were replaced with numerical relays.

### 9.1 Ageing Management of Load Centre Transformers

To cater to the requirements of various emergency and non-emergency loads at 415 V level, there are 2 emergency and 2 non-emergency buses (BTsb 100, BTsb 200, BTtb 300 and BTab400). All these are sectionalized buses with a bus coupler in between except for BTsb 400. Each section is fed by an independent 1 MVA 6600 V/415 V oil cooled load centre transformer. There are eight such load centre transformers. These transformers are in continuous service for more than 45 years. As

*Before and after overhauling transformer TMtb07* the life of the transformer depends mainly on the life of the insulating material, it was decided to either replace or renew these transformers. Accordingly, 5 new transformers of rating 1.25 MVA 6600 V/415V were procured. The replacement of transformers was taken up one at a time. Loads on the bus whose incomer transformer was to be replaced, have been shifted to other section so that none of the loads were affected.

**9.2 Overhauling of 10 MVA Main transformers**

There are 2 nos. 10 MVA 33 kV/6.6 kV (TMtb 001/TMtb 002) oil cooled transformers with On Load

*Transformer after overhauling*

*TMtb01 Transformer Installation Completed @ site*


*Transformer being transported to the overhauling site*

Tap Changer (OLTC) which are in continuous service catering the power supply requirements of the plant. Considering their age and long service, it was decided to carryout major overhauling of these transformers in a phased manner to maintain their availability for continued plant operation. Accordingly, transformer TMtb 001 was serviced at a transformer servicing facility at Chennai.

**9.3 Flood Protection & Post Fukushima Upgradations**

Recent studies estimated the revised flood levels as 12.01 meters (RL) under cyclonic condition

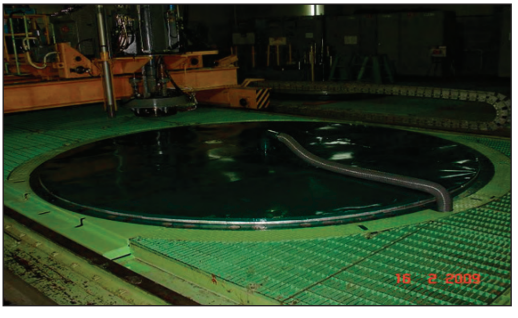

*Post Fukushima Retrofits*

combined with heavy precipitation & high tide with return period of 1000 years. Accordingly, entry points of the plant have been raised to 12.01 meters (RL) from the existing level of 11.5 meters. The flood level is estimated to be 12.896 meters (RL) taking into the consideration under worst case of Tsunami along with high tide with return period of 10000 years. For protecting the plant against this, easily installable FRP shutters of height 1 m will be provided at the entry points whenever Tsunami warning is received. As part of Fukushima retrofits, solar lamps were installed in and around FBTR.

In the aftermath of Fukushima accident, a study of post decay heat removal was carried out and various modifications / qualifications of the systems required to do their intended functions in detail were studied and the modification works were completed. Recent studies on extended black-out scenario indicate that the natural convection is sufficient to remove decay heat even if the steam generator trap doors are not opened. Also, even after one year of blackout, there is no risk of freezing of sodium in the primary capacities.

**10.0 Radiological Safety**

FBTR has an enviable record of minimum dose levels to working personnel. The general radiation levels in all the accessible locations in the reactor containment building is not more than 0.5 mR/h even when the reactor is operating at 40 MWt power. The average individual exposure is less than 0.03 mSv/y as against the permissible exposure of 20 mSv/y. Significantly, during the past 36 years of operation, there has not been any abnormal radioactivity release, personnel or area contamination.



*Cocoon over the reactor pile*

**11.0 Transition from 32 MWt to 40 MWt (Full Power) Core for FBTR and Addressing Regulatory Requirements**

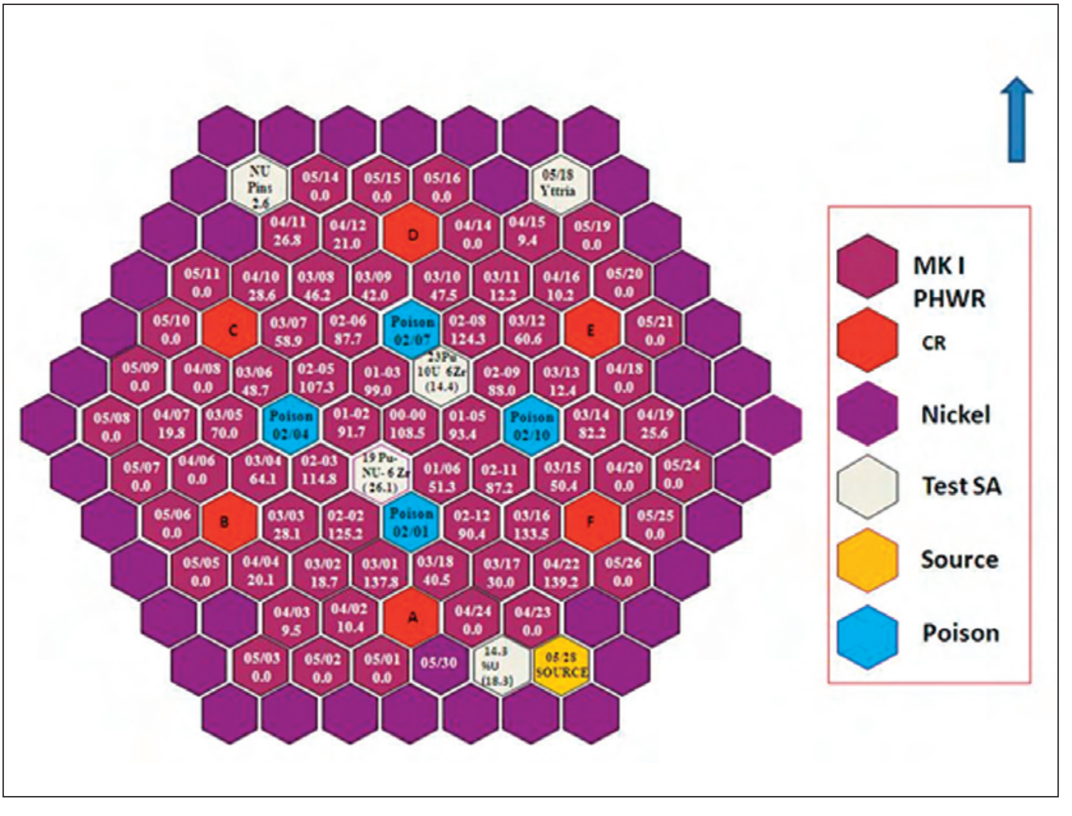
**11.1 Introduction**

The core of FBTR was originally designed with 65 SAs of MOX with 30%$PuO_2$ - 70%$UO_2$ with 85% enriched Uranium. Due to high cost and political reasons, enriched Uranium could not be procured from France. The search for alternate fuels ended up in mixed carbide fuel. This fuel has excellent compatibility with sodium, high thermal conductivity, good irradiation behaviour and good breeding potential. The reactor was made critical with 22 FSAs of mixed carbide fuel of MK-1 type which is of 70% PuC - 30% UC composition and the initial target power was 10.5 MWt. The reactor power was progressively increased by raising the operating Linear Heat Rate (LHR) based on encouraging results of Post Irradiation Examination (PIE) of FSAs discharged at burn ups of 25, 50 and 100 GWd/t and by adding more FSAs and a maximum power of 32 MWt could be achieved. During this time, all variants of carbide fuel viz. Mark I (HG), Mark I (PHWR), Mark II and mixed oxide fuels have been introduced into the core and valuable irradiation data obtained.

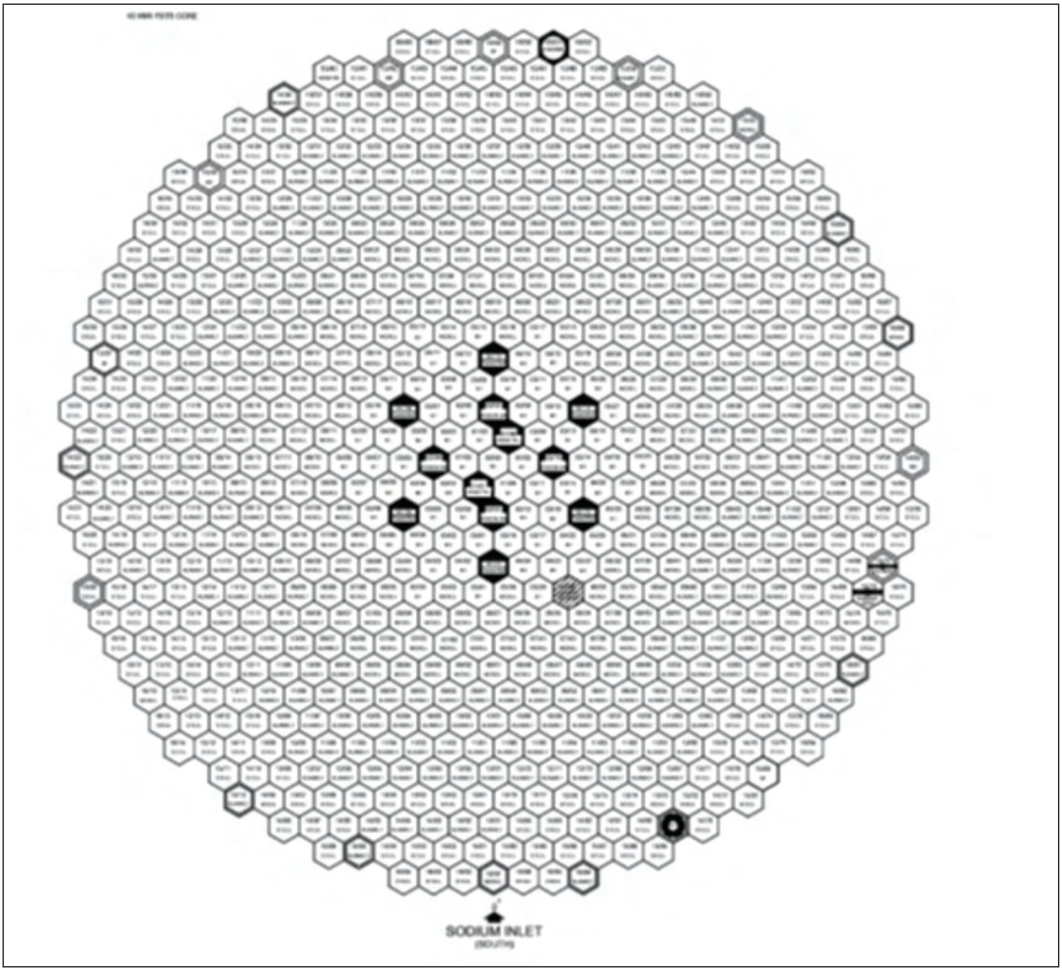
All the engineering systems in FBTR are designed for 40 MWt operations. Hence a feasibility study was carried out to raise the power of FBTR to its design target power of 40 MWt by adding Mark 1 subassemblies with the peak power rated subassembly operating at 400 W/cm. Theoretical studies have indicated that the maximum power attainable is limited to 32 MWt with 56 Mark I subassemblies after replacing all the Mark II and MOX subassemblies. Any further increase in the number of subassemblies reduces the shutdown margin. To ensure a minimum shutdown margin of 4200 pcm, which is the technical specification requirement, four poison subassemblies were added in the 2$^{nd}$ ring.

**The Salient Features of 40 MWt Core are as Follows:-**

- All fuel subassemblies are of Mark-I type
- Central location is reserved for experimental fuel SA (IFZ 100, with central seven pins removed and having 54 fuel pins) for irradiation of structural material
- Two locations in the first ring are reserved for experimental SAs (ISZ type SA) for irradiation of metal fuel pins
- Four equi-angular locations in the second ring have been chosen to load the B4C poison (50% enriched B10) subassemblies




*40 MWt core details*




*Nominal Core of FBTR showing 745 positions*

The safety report of FBTR had to be modified as the target power reaches 40 MWt. The chapters of the original safety report affected by core design changes were revised and issued as safety report for 40 MWt core. The chapters revised are Chapter 3 on Reactor Physics, Chapter 4 on Shielding, Chapter 5 on Core Engineering and Chapter 17 on Event Analysis.

**11.2 Event Analysis**

Detailed event analyses for the 40 MWt core of FBTR comprising of 70 nos. of MK-I fuel subassemblies was carried out using the plant dynamics code DYNAM. In all the events, except 'off-site power failure', 'station black out' and 'one control rod withdrawal', the clad hotspot temperature is limited below 800 $^{0}$C and fuel hotspot temperature is limited below melting point even without safety actions. Moreover, safety of the plant is demonstrated under off site power failure, station blackout and one control rod withdrawal events with reactor trip considered based on the second appearing parameter. During loss of feed water flow in one loop event, no parameter is available to initiate automatic trip of the reactor. However, there is no concern on core safety during this event even without any safety actions. Nevertheless, alarms are available to alert the operator once this event occurs. Hence, an additional LOR parameter based on "reactor vessel sodium inlet temperature high" to prevent continued operation of the reactor in case of loss of feed water flow to any one loop of steam Generator was introduced for 40 MWt operation after getting AERB approval.

**11.3 Accident Analysis**

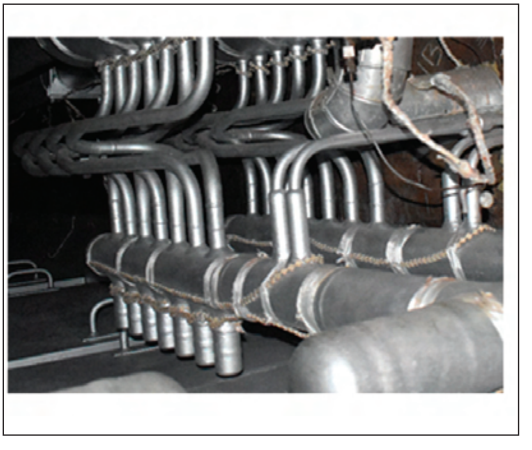
Hypothetical core disruptive accident analysis has been carried out for 40 MWt FBTR core with 70 Mark-I FSA. In case of ULOFA, the reactor power falls below the decay heat level and giving credit to decay heat removal system and the transient does not lead to core disruptive accident. For conservative reactivity addition rate of 50 $/s input during disassembly phase, the mechanical energy release under UTOPA is 6.9 kJ.  Under ULOCA, for conservative reactivity addition rate of 50 $/s, the mechanical energy release is 0.1 MJ when switch over is made to disassembly phase when peak fuel temperature touches boiling point; while it is 11.94 MJ when switch over is made earlier when core becomes prompt critical.  In later case, fuel in core is vaporized as much as 97%. Hence, for this core, 12 MJ is fixed as the highest HCDA mechanical energy release value.  The reactor vessel can take up to 9 TNT (39 MJ) of mechanical energy release without failure.

Following system changes were carried out for 40 MWt operation

**11.4 SG Tube Normalisation**

Steam Generator (SG) of FBTR is a once through shell & tube type counter flow heat exchanger which generates superheated steam at 125 kg/cm$^2$& 480$^o$C. There are two SG Modules in each east & west secondary sodium loops. Out of the total 4 modules, three are made of 21/4Cr-1Mo-Nb Stabilized ferritic steel and the other one is made of 9Cr-1Mo-Nb Stabilized ferritic steel. In the SG module, sodium flows through shell side and water/steam flows through the tube side. In order to operate the reactor near design temperature at lower reactor power levels,  3 out of 7 tubes in each SG module were blanked in 2008. Reactor was operated with partially blanked SG tubes till 29$^{th}$ irradiation campaign.

*Tubes of SG in blanked condition*



*Normalized SG tubes*

From 30th irradiation campaign onwards, it was planned to operate the reactor at design power level. This necessitated normalization of blanked tubes of SG modules.

Before starting the normalization work, welding procedure sequence and welder performance qualification were carried out for meeting the FBTR specifications. As FBTR Steam Generator modules were fabricated with different materials for Steam Header and Water Header of individual modules, separate procedure and performance qualifications were carried out.

After successfully completing the qualification requirement, mock-up trials were made for the weld configurations simulating the site constraints to train the working personnel as the entire welding work had to be performed in the highly congested SG cabin. Difficulties were faced in the internal grinding of SG tubes. To overcome this, a special purpose ID grinding tool was fabricated for SG tubes. For normalization, 12 blanked portions in each SG modules were to be cut and normalized.

After gaining sufficient confidence from the mock-up trials, caps of blanked tubes were cut suitably, spool tube piece was introduced between the open ends and welded. GTAW process with argon gas purging was used for welding. Every day, before commencing the actual job, trial pieces were welded to establish the consistent performance of welders. The tube internals were visually inspected with videoscope before fit-up. Weld edges were also visually inspected to avoid any abnormalities before fit-up. LPT was carried out on weld edges to detect any surface defects. Spool material was tested for its conformability to the joining tube material before making the weld joint with X-ray florescence spectroscopy.

After successful completion of all the welding works, Post Weld Heat Treatment (PWHT), Liquid Penetrant Test (LPT), Radiographic Test (RT) and Helium Leak Test (HLT) were carried out for all the weld joints. Finally, SG was subjected to a hydro test at 135 bar and SG was normalized.

**11.5 Revamping of 50 MWt Main Cooling Tower**

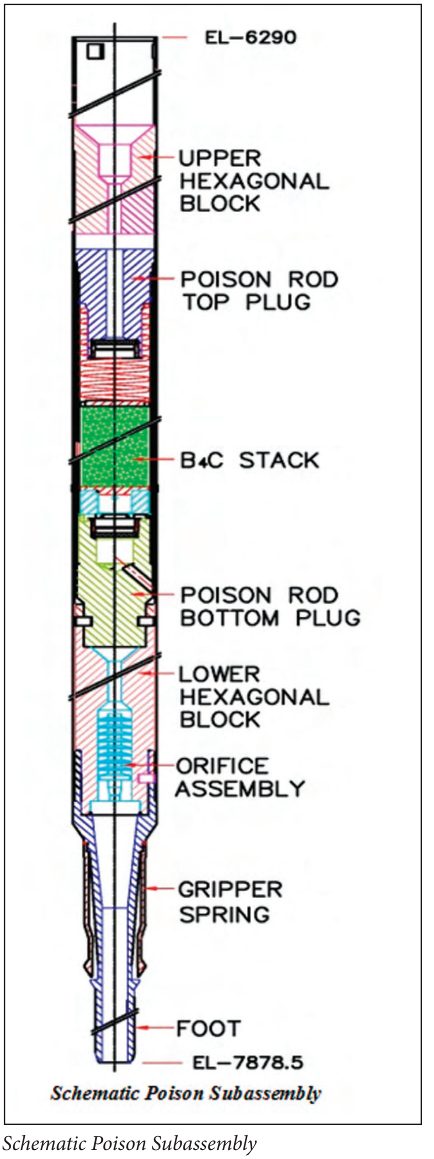
Prior to raising FBTR power to design power level, the cooling tower was completely dismantled and refurbished. The existing cooling tower splash box & cover assembly on water distribution system,

*FBTR cooling tower*

PVC 'V' type splash bars, Splash bars support grid system, PVC drift eliminators and GI corrugated louver sheets were replaced with fresh materials. Cooling Tower was put into service. Structural stability of all the internal materials was checked to be satisfactory.

### 11.6 Poison Subassemblies in Core

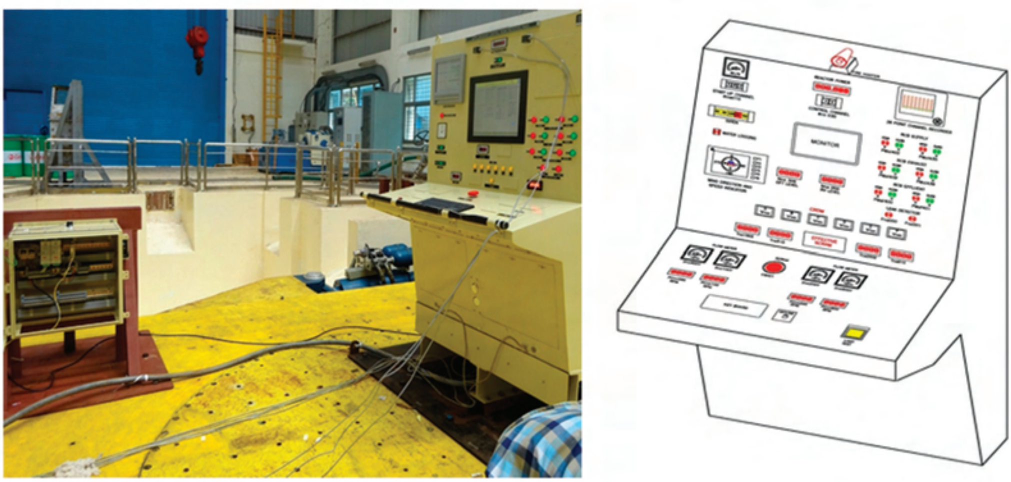
Four equi-angular locations in the second ring have been chosen to load the $B_4C$ (enriched to 50% in B-10) subassemblies based on the results obtained for various locations and enrichments. The arrangement of $B_4C$ column in these subassemblies is very similar to that in the control rods. The diameter of the $B_4C$ column is 38 mm, same as that in the control rod. The required column height of $B_4C$ is 375 mm. The top of the $B_4C$ column is aligned with the top of the fuel column whose length is 320 mm. The remaining 55 mm of $B_4C$ column will be below bottom elevation of the fuel column. This additional length serves two purposes. (1) It increases the magnitude of the negative reactivity introduced by these $B_4C$ subassemblies, thereby increasing the shutdown margin. (2) The design of $B_4C$ poison SAs are like any other SAs and are supported on the grid plate. However, for study purpose, a hypothetical event of ejection of a $B_4C$ subassembly from its location was considered. Due to availability of 55 mm of $B_4C$ column below the core, the reactivity change in the core is minimized, essentially by ensuring that the fuel column is covered even in the ejected condition. It may be noted that due to physical location of CCPM at 70 mm above the top of SAs, the maximum lift possible is only 70 mm. Based on the calculations, the reactivity addition due to ejection of one $B_4C$ subassembly, with CCPM in position, is 30 pcm.




*Schematic Poison Subassembly*

### 12.0 Periodic Safety Review of FBTR

Initial safety review of carried out well before the creation of the AERB in 1987. SEWG (Safety Evaluation Working Group) had carried out detailed review of the design and safety analysis of FBTR, covered in the preliminary safety analysis report titled 'FBTR hazard evaluation report'. This document was revised addressing the comments of SEWG review and the 'FBTR safety report' was submitted to the DAE-SRC in 1983. The safety analysis chapter was further revised based on core design changes due to use of small carbide core and rated power of 10.5 MWt in 1984. The technical specifications document was prepared based on those of RAPS and MAPS and was reviewed by RRC-CWMF SC and SEWG and approved by DAE-SRC in 1984. DAE-SRC accorded clearance for commissioning and criticality after ensuring availability of important station documents like safety report, technical specifications, system commissioning documents, procedures for physics and engineering experiments, system manuals and trained man power. However, no formal plant operation license was issued and campaign wise clearances were being obtained. First criticality of FBTR was achieved on October 18, 1985 with a small carbide core and nominal power of 10.5 MWt. Initially FBTR was operated up to 1 MWt without the steam generator, and later gradually raised to higher power in stages with the commissioning of steam generator, steam water system and the turbo alternator and adding more fuel SAs in the core. With the constitution of AERB in place of DAE-SRC, regulatory procedures were enforced in more formal ways requiring stage wise consent for the new projects, licensing, periodic regulatory inspection and relicensing of operating plants based on periodic safety reviews as per the established safety guidelines and procedures. In addition, periodical review of the safety status of the reactor through committees established at three levels was continued. AERB grants authorization for operation for five years normally based on Periodic Safety Review (PSR) and relicensing within the period of operating license of a nuclear power plant. Renewal of authorisation after this period is based on a comprehensive safety review considering cumulative effects of plant ageing, irradiation damage, results of in-service inspection, system modifications, operating experience, status and performance of safety systems and safety support systems, revisions to applicable safety standards, environmental impact etc. Preparation of the first Periodic Safety Review(PSR)document of FBTR was initiated in 2003 based on the periodic safety



*Successful Seismic testing of SCP panel and Junction box in Shake table & Supplementary control panel layout* review document of MAPS. The first draft of periodic safety review document (except a few chapters) was prepared and submitted to Operating Plants Safety Division (OPSD) of AERB for review in 2005. After review of the draft by the OPSD, AERB in 2006 and incorporating the comments, updating the plant parameters and other details, the document was resubmitted. The periodic safety review document was reviewed chapter wise by IGCAR-SC. This was an iterative process and led to 10 drafts till final approval in 2012 by SARCOP. One of the major exercises required was residual life assessment of the plant equipment as a part of ageing management for components and equipment, which cannot be replaced like block pile components, reactor vessel, grid plate etc. This was done by analysis to estimate the material damage due to neutron dose. Since FBTR has operated below creep range, enough residual life is available for the components whose life is governed by creep-fatigue damage and the life of FBTR is found to be essentially governed by the neutron dose on the grid plate. Stainless steel samples were irradiated to low damage levels and the loss of ductility was assessed. Equipment like obsolete switch gears, protective instrumentation and plant computers were replaced with state of the art systems. Another major work was the seismic re-evaluation of FBTR to revised seismic level. This was carried out as a joint research project between IGCAR and the AERB-Safety Research Institute by probabilistic seismic hazard analysis method. This involved ten main tasks including preparation of a criteria document, arriving at the Review Basis Ground Motion (RBGM) parameter based on analysis of site specific data and existing faults around Kalpakkam, safety analysis for arriving at the seismic structures, systems and components required to be qualified, detailed plant walk-down to identify seismic vulnerabilities due to interactions and easy fixes to overcome the same, analysis of as-built system data to assess adequacy of supports and anchorage, structural analysis for qualifying the various safety related seismic structures, systems and components, testing of components on shake table and finally assessment of seismic capacity of the structures, systems, components and adequacy of available seismic margin. Entire task was carried out by the team from 2006 to 2010. By December 2009, all the reports were submitted and reviewed by the subcommittees constituted by SARCOP and recommendations for retrofits and easy fixes were identified. Yet another major work was equipment qualification, qualification of the plant structures, systems and components for ensuring safe shutdown, decay heat removal and containment, to ensure capability of systems for mitigation of the consequences events like extended station black out, loss of coolant accident etc. Qualification of instrumentation from sensor to the final control element for monitoring temperature, radiation and pressure as part of post accident management in case of beyond design basis accidents taking into account the harsh ambient condition and high radiation in RCB. Design compliance to present practices was studied and major non compliances like non availability of back up control room, were identified and retrofits were engineered. The periodic safety document was reviewed by SARCOP in June 2012. The Fukushima accident added a new dimension to the exercise. Analyses of beyond design basis accidents had to be carried out and retrofit for increased design basis flood levels had to be worked out. Periodic safety review was a massive and challenging exercise. There are no parallel reactors from which the

*Supplementary control Room*

methodology for periodic safety review could be adopted. The plant is of vintage design and built at a time when the tools, techniques for analysis, design used were evolving and were more qualitative and manually done as compared to current practice of computer aided tools and codes. As archival methods for data retrieval were not adequate and also since most of the personnel associated with design/ construction have since retired, data/document retrieval was daunting. Safety analysis had to be carried out afresh for the current core with MK I, MK II and MOX fuels. The same team of FBTR catering to normal operation and maintenance had to carry out periodic safety review. Insistence on formats made on the actual document preparation was more time consuming. Being a new type of reactor under AERB regulation, the review process also got prolonged. Though it took a long time of about nine years, it was carried out meaningfully and the final periodic safety review document consisting of over 400 pages was completed and was reviewed by SARCOP in June 2013. The strengths and weaknesses of the plant were assessed by a revisit to the safety concerns/ issues and operation experience and suitably addressed.  FBTR was formally relicensed till June

2018. Subsequently in 2017, Application for Renewal of Authorization was submitted to Regulatory authorities and Relicensing was granted till June 2023.

During Periodic Safety review of FBTR, AERB recommended installing a Supplementary Control Panel (SCP) to monitor vital plant parameters in case Main Control Room becomes inaccessible due to any reason. Accordingly, seismically qualified control panel was designed as per the relevant AERB code. The SCP of FBTR is located in the turbine building operating floor. After seismic qualification tests, the SCP was installed in the new SCP cabin. The SCP is identical to the existing one in the main control room of FBTR so that the similar interface is provided to the operator for control action. With this, operation of FBTR under certain postulated design-basis events like fire in the relay room/ Main Control Room rendering the MCR uninhabitable has been made safe.

**13.0 Seismic Retrofits in FBTR**

FBTR site is in an area considered free of seismic events (zone number 1 as per IS 875). Nevertheless, considering the importance of the nuclear reactor installation and as a provision for future, seismic coefficients of 0.1 (g) horizontal and 0.05 (g) vertical had been considered in the design of safety related structures and buildings and 0.05 (g) horizontal and 0.025 (g) vertical for auxiliary buildings and structures. However, in view of the higher seismic values used for PFBR, it was required to carry out seismic re-evaluation of FBTR. Seismic re-evaluation involved identification of the seismic structures, systems & components for fulfilling the three safety functions-viz. safe shut down, decay heat removal and containment of radioactivity. Structural analysis was carried out for the buildings namely, reactor containment building, control building, service building, steam generator building and structures like cooling tower, diesel day tank and systems like primary and secondary sodium. 'Experience based method' was followed for battery stands, electrical distribution and instrumentation panels. 'Rule of the box method' was applied for components within the I&C panels. Components like electronic timer circuits, functional circuits of the safety critical computer system which are in the form of PCBs, cable penetration assembly etc. were qualified by testing using the shake table. In order that the functioning of the seismic structures, systems and components located in the neighborhood of non-safety related equipment is not hampered due to interferences or interactions during a seismic event from improperly supported equipment, such interfering equipment were identified by detailed plant walk down. A team of seismic experts knowledgeable and experienced in carrying out plant walk down visited various areas of FBTR and checked for interference to SSC, adequacy of anchorage of SSC, supports to various pipelines, tanks and other equipment. The team also identified URM walls (brick walls or unreinforced masonry walls) close to SSC affecting safety of SSCs. After evaluation of the existing anchorage/supports or based on the available documented experience data, the easy fixes and minor retrofits were identified. The team also identified areas housing redundant safety equipment and systems, prone to common cause failures during seismic event due to lack of segregation and also ease of access to critical equipment required for egress and also for safety related manual actions. The major retrofits required to ensure seismic qualification of FBTR are:-

i. Replacement of the original unanchored wooden battery stands by seismically qualified metallic stands anchored to the floor

ii. Seismic anchoring of gas cylinder banks in the plant

iii. Structural reinforcement of unreinforced masonry walls in service building (ground and first floor) and steam generator building

iv. Reinforcement of steel structure supporting equipment at -2.81 m elevation in reactor containment building

v. Complete segregation of the redundant battery banks of control power supply systems by extending the partition wall and provision of fire proof seismic doors and

vi. Improvements in access to the steam generator trap doors and manually operated valves of pre-heating and emergency cooling (PHEC) system.

The plant walk down also identified panels, structures and equipment which need improvement by means of easy fixes. This involves about 400 such modifications in the plant. Important ones are: -

i. Support arrangement for field and armature battery banks of primary and secondary Ward Leonard mounted within the panels

ii. Provision of adequately sized anchors for all safety related I&C panels, electrical distribution cabinets and MCCs and also the equipment and panels which are in their vicinity

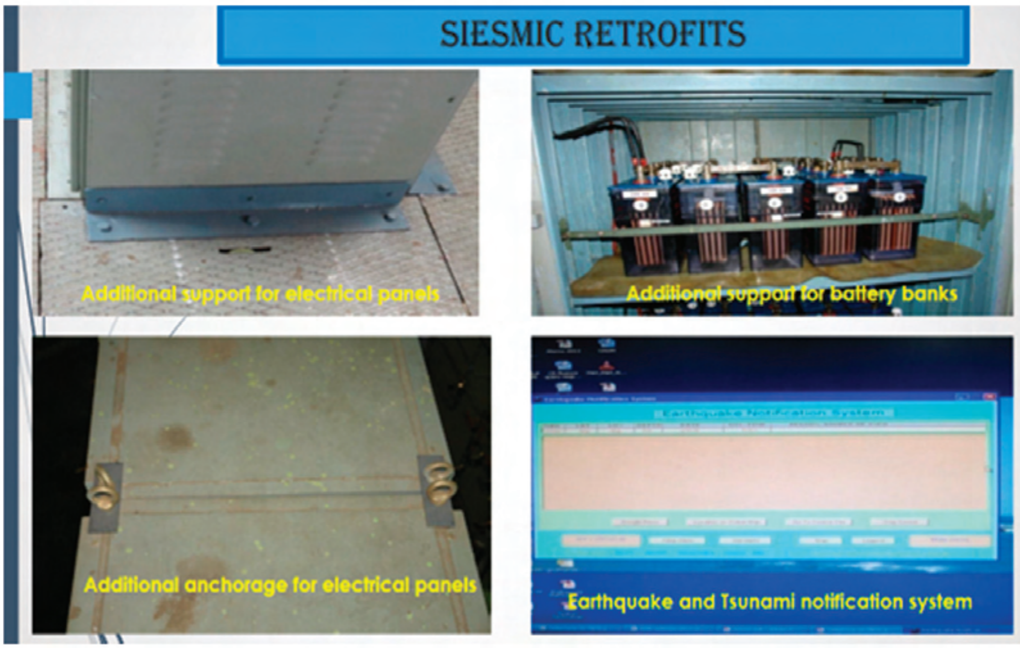
iii. Bolting of adjacent panels in RCB, SB, SGB, TB, Electronics & CDPS rooms and relay room to avoid seismic interactions

iv. Seismic locking of fuel handling flasks and cranes

v. Anchoring of furniture, lead brick partitions between redundant exhaust filter banks, lead bricks used for complementary shielding in reactor containment building

vi. Additional supports for the cable trays located near the SSC vii. Decongestion of cable trays in RCB -2.81 m elevation

vii. Provision of clamps for relays, improving the supports of the panel mounted recorders and meters Metallic battery stands




*Seismic sensor located at RCB*

viii. Clamping of WL system batteries located in the panels

ix. Clamping of isolation transformers of control power supply system

Among the major modifications, battery stands have been replaced and all the cylinder banks have been secured with adequate anchorage. Pneumatic mechanism has been provided for operation of the trap doors of SG casing. Battery banks of Ward Leonard drives have been provided with clamps to avoid dislocation during seismic event. Anchoring of MCC panels and bolting of adjacent panels have been completed for RCB, SB, relay room and SGB. Manually operated PHEC valves are replaced by pneumatic valves operable from control room. URM wall reinforcement was a major work which will be taken up during 2014. In its existing condition during the seismic reevaluation, FBTR was qualifying for a peak ground acceleration (PGA) of 0.09 (g) and with the retrofits & minor fixes, it was qualified for a PGA of 0.22 as per the Review Basis Ground Motion (RBGM).

As a part of seismic retrofitting programme, the adequacy of the systems to withstand SSE for safe shutdown, decay heat removal and containment integrity have been assessed. In particular, plant buildings, anchoring of electrical & instrumentation panels and sodium tanks and other capacities were verified and wooden battery stands of UPS and control power supply were replaced with

*Seismic recording instrumentation in Relay room*

seismically qualified metallic stands. It is planned to install two more seismically qualified DG sets and emergency switch gears in the new seismically qualified Flood Safe Service Building (SFSB). Seismic instrumentation to measure seismic activity in safety structures as well as free-field close to the reactor has been commissioned


*Seismic sensor located at RCB*

**14.0 Journey of FBTR Continues**

FBTR has reached its full 40 MWt design power level for the first time in the year 2022. For raising the reactor power to its rated full power, necessary core changes were made in six fuel handling steps after obtaining safety clearance from AERB. Present core consists of 68 MK-I fuel subassemblies and four poison SAs in the 2$^{nd}$ ring. Following are the present major missions of FBTR: -

> Continue irradiation of the sodium bonded metallic fuel pins

- Ternary fuel-1 (23% Pu-19% EU-6% Zr) at location 0104
- Ternary fuel-2 (19% Pu-U-6% Zr) at location 0101
- Binary fuel (14.8% EU-6% Zr) at location 0530
- Natural U-6% Zr metallic fuel at location 0513

> Continue long term irradiation of structural materials

> Irradiation of Yttria for production of Strontium($Sr^{89}$).

After completing necessary core changes, 30$^{th}$ irradiation campaign was initiated on 22.02.22. At 25 kWt, all low power physics experiments such as Control rod calibration, calibration of high range gamma monitor, Isothermal temperature coefficient and calibration of Steam Generator Leak Detection Channels (SGLDS) channels were completed. On 04.03.22, reactor power raising was initiated. While raising reactor power, power co-efficient was measured for each MWt and ensured to be negative for ensuring the inherent safety of reactor as per design intent. On 05/03/2022, TG was synchronized to grid at 04.15 h. On 07/03/2022, reactor power was gradually raised and reached to 40 MWt at 17:30 h achieving the design power of FBTR for the first time in the history of FBTR.

At present, FBTR is operating at 40 MWt with TG connected to grid delivering 10 MWe. Electrical energy generated so far in this campaign is 6 MU. It is planned continue operation of FBTR at this power level in the forthcoming campaigns also for irradiation of advanced fuel & structural materials and for production of medical isotopes.

The current residual life of FBTR has been estimated as ≈6.3 EFPY and it will continue to be the work–horse for its subsequent missions for the development of the next generation of fast reactors. It is planned to extend the life of reactor by replacing the bottom axial stainless steel (SS) pins used as shielding material in the fuel SA with Tungsten Carbide (WC) to reduce the fluence seen by the grid plate. By this reduction of fluence on the grid plate, life of FBTR will be extended by another ≈2 EFPY.

The major agenda of FBTR in the coming years will be to study the performance of metallic fuel and the metallic fuel will be irradiated to the burn up levels of 100 GWd/t.

Following irradiation programs are planned in the near future depending on the availability of fuel:-

- To irradiate a test SA having 37 nos. of sodium bonded metallic fuel pins of natural uranium and zirconium alloy
- To irradiate a test SA having 37 nos. of sodium bonded metallic fuel pins of 23%Pu-Nat.U-6%Zr alloy
- Long term irradiation of advanced structural materials like Ferritic Steels (T-91), IFAC-1 (D9I), Yttria stabilized Oxide dispersion strengthened alloys (ODS) in IFZ 100
- Installation of Cesium & Manganese traps in the reactor core for trapping these elements in sodium

Presently 274 Thoria blanket SAs are loaded in the outer rings of the core and they will remain in place till the end of the life of reactor as originally envisaged. When discharged, it will provide data on the characteristics of $U^{233}$ derived from a fast reactor and help in developing flow sheets for reprocessing and recovering $U^{233}$ and provide fuel supplement for development of third generation reactors. This will be a contribution of FBTR to the third stage of our nuclear programme.