# Operating experience of FBTR

SURESH KUMAR.K.V, THANGAMANI.M, MANIMARAN.N, BABU.A,

Indira Gandhi Centre for Atomic Research

Kalpakkam, India

Email address of main author: [kvsuresh@igcar.gov.in](mailto:kvsuresh@igcar.gov.in)

**Abstract:**

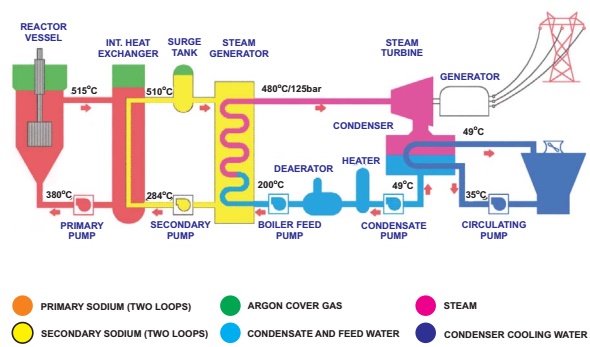
The Fast Breeder Test Reactor (FBTR) is a sodium cooled, loop type fast reactor commissioned in the year 1985. It is a research reactor with an evolving core with a unique Pu/U mixed carbide fuel. The reactor has continued to provide valuable operating experience. FBTR has been operated at various power levels up to 40 MWt /10 MWe over the last 35 years or more, realizing its objectives viz; mastering sodium cooled fast reactor technology, testing of advanced metallic fuels & future reactor structural materials and manpower training & development. Based on the excellent performance of the Mark-I fuel, its burn up limit has been progressively increased after post Irradiation Examination at various burn-up. The maximum burn up attained so far without clad failure is 165 GWd/t. FBTR has completed 29 irradiation campaigns so far and 30th irradiation campaign @ 40 MWt is in progress. Primary sodium temperature nearer to rated design value with less reactor power was achieved by operating the Steam Generator with 3 out of 7 tubes blanked. This paper describes the various safety up gradations carried out, operating experience with sodium pumps, experience with failed fuel localization, life assessment & extension studies of FBTR, replacement of liquid metal seals in block pile, replacement of Steam Generator modules, one with tube leak and other with shell side sodium leak, modification done in the reactor protection circuit to avoid SCRAM during LOR.

**Key Words**: FBTR, sodium cooled Fast reactor, Carbide fuel

## **INTRODUCTION**

The Fast Breeder Test Reactor (FBTR) 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 & materials. FBTR was built on the lines of the French Rapsodie-Fortissimo reactor with modifications to make it a generating plant.

FBTR heat transport system consists of two primary sodium loops, two secondary sodium loops and a tertiary steam water circuit (Fig: 1). 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 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.



*FIG.1. Schematic of Heat Transport system of FBTR*

TABLE 1. MAIN PARAMETERS OF FBTR

|  |  |
| --- | --- |
| Parameters | Values |
| Reactor Power | 40 MWt / 10.0 MWe |
| No. of Fuel SAs | 68 MK-I . |
| Fuel Pin diameter | 5.1 mm |
| No. of pins / SA | 61 |
| Maximum LHR | 400 W/cm |
| Peak Neutron flux | 3.76\*1015 n/cm2/s |
| No. of Control Rods | 6 |
| Control Rod Material | B4C (90% enriched in B10) |
| Speed | 1 mm/s |
| Control Rod worth | 9744 pcm |
| Reactor inlet sodium temperature | 380oC |
| Reactor outlet sodium temperature | 490oC |
| Primary Sodium Flow | 938 m3/h |
| Feed water temperature | 190oC |
| Steam Temperature | 460 oC |
| Feed water flow | 53.8 m3/h |
| Steam Pressure | 120 kg/cm2 |
| Sodium Inventory | 150 t |
| Steam Generator | Once through type, 4 tubes in Shell, triple S shape |
| Turbo Generator(TG) | 16 stages, condensing type, 16.4 MWe air cooled. |

FBTR started its operation with a small carbide core rated for 10.6 MWt. The carbide fuel has been performing extremely well and the reactor is operating at its rated power of 40 MWt.

## **operation experience**

The reactor attained sustained chain reaction for the first time in October 1985 with a small core of 23 Mark I subassemblies rated for 10.6 MWt at a peak LHR of 250 W/cm. Reactor power was progressively increased by enlarging the core and TG was synchronized to grid in July 1997. The present core has 68 fuel subassemblies (FSAs) rated for 40 MWt at a peak LHR of 400 W/cm. The reactor has so far clocked 60195.25 h and 1611 EFPD of operation and the Mark-I driver fuel has achieved a peak burn up of 165 GWd/t without clad failure.

As the mixed carbide fuel of this composition was used as driver fuel for FBTR, the Linear Heat Rating (LHR) for the fuel was restricted to 250 W/cm and the target burn-up was set at 25 GWd/t initially. The target burn-up has been progressively increased based on Post-Irradiation Examinations (PIE) at 25, 50, 100 & 155 GWd/t. In the light of the excellent performance of the carbide fuel based on Post-irradiation examinations, which endured a burn-up of 155 GWd/t without any clad failure (one lead subassembly up to 165 GWd/t), the core has been gradually expanded further by the addition of Mark-I (70% PuC- 30% UC) along with Mark-II (55% PuC-45% UC) and MOX (44% PuO2-56% UO2) fuel and reactor power was raised progressively. A major modification i.e. isolation of 3 tubes out of 7 tubes in each steam generator module was done to raise the sodium temperature close to design values reactor inlet/outlet temp (380ºC /515ºC) with constraints on core size and reactor power so as to conduct meaningful irradiation of materials and to study the performance of the systems at design temperatures at the end of 14th irradiation campaign.

FBTR has so far completed 29 irradiation campaigns in the last 35 years of operation. It is currently operated on a mission mode for the irradiation of metallic fuels & structural materials. PFBR test subassembly irradiation up to 112 GWd/t based upon Cumulative Damage Fraction (CDF) value, TRISO coated particle irradiation with ZrO2 kernels and disc specimens of Nb-1Zr-0.1C for High Temperature Gas Cooled Reactor (HTGR), short term irradiation of the sphere-pac fuel pins, production of Sr89 by irradiating Yttria - a medical isotope used for bone cancer therapy, irradiation of Ferro-boron shield material & other structural materials were completed. The reactor was also utilized for studying the irradiation creep behaviour of Zr-Nb for Pressurized Heavy Water Reactors. 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.

Over the years, several safety related physics experiments have been conducted. These include, temperature and power coefficient measurements, sodium void coefficient measurements, reactor kinetics experiments, response of delayed neutron detection system to detect clad failure, flux mapping in sodium above core using foils, drop time measurements using Kalman filter technique, subassemblies and control rod worth measurements, neutron detectors testing and experiments to validate the Failed Fuel Detection System. Several engineering tests also were carried out to validate the codes used in incident analysis like off-site power failure and tripping of one pump each in the primary, secondary or tertiary loops. Natural convection tests were carried out and the sequence of events confirmed to be as per safety logic. Primary pump coast down characteristics, take over by the batteries and low speed running of the pumps, were studied and found to be as per the design intent. Other tests like heat removal capability by Biological Shield Cooling (BSC) & Pre-Heating and Emergency Cooling (PHEC), Reactor Containment Building (RCB) leak test, steam generator instability test, biological shield concrete temperature evolution test were checked and validated.

The challenges faced during the 35 years of operation include a major fuel handling incident, primary sodium leak, reactivity transients, leak in biological shield cooling coils, replacement of Steam Generator modules subsequent to tube leak in one & sodium leak from the other and replacement of rupture disc assemblies in the secondary sodium system.

Sodium systems have been operating for the past 35 years or more and their performance has been excellent except for a few leak incidents from bellows sealed sodium valves. The sodium purity has been maintained consistently well all these years by cold-trapping of oxides (impurity level < 0.6 ppm) and cover gas purity is also well maintained. The four sodium pumps have logged trouble free, cumulative operation of more than ≈ 950,000 h. There were no incidents of oil leak from the pump seals to the sodium circuit so far.

Two major modifications were carried out - one on the Steam Generator Leak Detection System (SGLDS) and the other on the steam-water circuit. These modifications have helped in improving the campaign availability from less than 50% initially to more than 90%. Also several up gradation and modifications were done in the plant to improve the system reliability, availability, safety and operational convenience. All these modifications have gone a long way in improving the campaign availability factors to more than 90%.

The thermal energy generated taking into the cumulative value since first criticality is 783.65 GWh and the turbo generator has generated cumulatively ≈ 90 Million Units of electricity.

The general radiation level in all the accessible locations in reactor containment building during the operation of the reactor was very less and within the acceptable limits. During the last 35 years, total radioactivity released to the environment is very low and there has been no significant event of any abnormal radioactivity release, personnel or area contamination testifying the fact that Fast Reactors are radiologically safe to man and the environment.

## **Safety upgradation**

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 upgradation programme, several major components have been replaced. These include the Neutronic channels, Uninterrupted Power Supply (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 & Automatic Voltage Regulator (AVR) of TG. Due to obsolescence, 6.6 kV Minimum Oil Circuit Breakers (MOCBs) were replaced with Vacuum Circuit Breaker (VCB) and 415 V electro-mechanical relays were replaced with numerical relays.

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 seismically qualified metallic stands. It is planned to install two more seismically qualified Diesel Generator (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

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

During Periodic Safety review of FBTR, AERB has 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 operating different equipment. With this, operation of FBTR under certain postulated design-basis events like fire in the relay room rendering the MCR uninhabitable has been made safe and smooth.

## **life assessment & extension studies**

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 licence 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.

The plant life is governed by replaceable and non-replaceable components and the ageing factor is based on corrosion studies, 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 520˚C.

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:

* **Measures to mitigate & control the ageing mechanisms by Good operation practices.** These are meant to mitigate the ageing mechanisms and ensure that the design life of each component is met.
* **Monitoring the failure of any equipment at the incipient stage by on-line monitoring of the functional integrity of component.** Various important systems are monitored online to ascertain their leak-tightness and functional integrity. For example, Interseal argon flow is continuously monitored. In the event of any increase, it is investigated. Leaks so far have been due to failure of bellows in CRDM or passing of O-rings in CRDM or thermocouple passages on top, and not due to opening up of any thermowell in Core Cover Plate Mechanism.
* **Monitoring & trending the ageing effects by in-service inspection.** The in-service inspection programme is designed to monitor the ageing and degradation of various components, and is based on the feasibility of carrying out the inspection. In-service inspection (ISI) programme ensures the continuing integrity/operability of safety related equipment by periodic monitoring. The programme takes into account typical service environment and special design features incorporated to safeguard failure of various barriers against release of radioactivity.
* 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 life limiting mechanisms of heat transport systems of FBTR are creep and fatigue. Since the reactor has operated at reduced power and temperature until 2008, the creep effect is insignificant. The total number of significant thermal cycles seen by the reactor components as of now is 177 as against the design cycle of 2000 for most of the components. 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%.
* 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. 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.

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 SS316 grid plate material irradiated in an earlier campaign in FBTR at 340°C up to 2.5 dpa indicated a residual ductility of above 20%. Towards assessing the strength and ductility changes of grid plate material subjected to low neutron doses up to 6.75 dpa and facilitating life extension of FBTR, accelerated irradiation experiment was conducted in FBTR. Tensile test and disc specimens fabricated from archival FBTR grid plate stainless steel of grade 316 were irradiated in an experimental capsule to a neutron dose of 2.30 to 6.75 dpa at 380°C during the 25th and 26th irradiation campaigns. Tensile testing of grid plate grade SS316 samples indicated progressive increase in yield and ultimate tensile strength (UTS) and a corresponding reduction in elongation with increasing dpa. The uniform elongation of FBTR grid plate SS316 specimens at 6.58 and 6.75 dpa, tested at 380oC was in the range of 12-15% and total elongation 16-19%. Factoring the higher rate of loss of ductility of grid plate during lower (340oC) operating temperature of FBTR up to 14th irradiation campaign, FBTR can be operated up to about 6.3 dpa based on the design limit of 10% ductility. Accumulated neutron dose seen by the guide plate of grid plate assembly is 2.35 dpa at the end of 29th irradiation campaign.

## **High speed operation of primary centrifugal sodium pumps**

During shut down state of reactor when pumps are operating ≤ 500 rpm, the entire primary cover gas circuit is in communication and the isolation valves of both IHXs & pumps are in open condition. In order to maintain sufficient level above IHX window at higher sodium flows for reactor operation thereby avoiding cover gas entrainment, both primary sodium pumps are being started keeping the IHX cover gas communication valves closed. When the level in the pump reaches ≈ 60%, the cover gas communication valves of both pumps are isolated. This is done to have sufficient level in the pump capacities at higher speed of operation. Closure of cover gas communication valves also ensures the following:-

In the event of tripping of one of the pumps, the drop in free sodium level (de-flooding) of the operating pump and increase in sodium level (flooding) of the tripped pump is within the permissible range 3.2% to 97.5% as given in the technical specifications.

After reaching the rated pump speed required for the irradiation campaign at 180ºC sodium temperature, IHX is under vacuum (≈ -50 mb) and the sodium level in IHX is ≈ 80%. In addition to seat passing of cover gas communication valves, while raising the sodium temperature with cover gas spaces of both the IHXs isolated, vacuum in IHX cover gas space reduces due to increased vapour pressure. Hence, IHX level decreases gradually with decrease in IHX vacuum. With progressive decrease in cover gas vacuum, sodium level may fall below the minimum stipulated level of 42.1%. Hence, maintenance of required vacuum in IHX is essential for maintaining the required differential head and thereby the sodium level. In order to maintain the required vacuum in IHX, a vacuum pulling set up is integrated to the existing IHX vent line. This setup is put into service when IHX level drops below 60% (to avoid argon entrainment) and stopped if level reaches above 80%.

## **experience with failed fuel localization**

During the reactor operation in the last 35 years, 2 wet ruptures (in 2011 & 2020) and one gas leaker in 2018 took place. FBTR is designed with Clad Rupture Detection (CRD) system in cover gas and Delayed Neutron Detection (DND) system in sodium to detect the failure of fuel clad. But provision for locating the failed subassembly from the core does not exist. The detection system responded as per design intent in all three failures. Methodology adopted for identifying the failed sub assembly is as follows:-

* Identification of suspected fuel SAs based on observed DND contrast ratio, activity ratio of Kr85/ Kr88 at the time of reactor SCRAM
* Shifting of suspected fuel SA to storage location and check for the evolution of DND counts by operating the reactor at 10% of the target power. If DND counts are same as the background counts corresponding to the core with healthy FSAs, the fuel SA shifted to storage location is failed one.
* Shifting of suspected fuel SA from east location to west & vice versa and checking for reversal of DND contrast ratio by operating the reactor at 10% of the target power
* Carry out flux tilting experiment at 10% of the target power to observe for change in evolution of DND counts for identifying the suspected FSAs
* After identification, the suspected fuel SA is moved to storage location and operate the reactor at high power to ensure that DND counts are close to back ground values

## **Replacement of liquid metal seals in block pile**

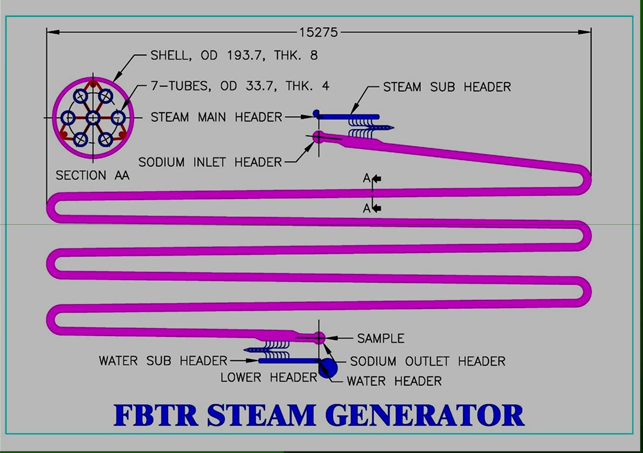
The top of the Reactor Vessel (RV) of FBTR has a Large Rotating Plug (LRP) and a Small Rotating Plug (SRP) to facilitate rotation of the plugs for carrying out fuel loading/discharging operation at 745 positions in the reactor through a single canal. The primary leak tightness between SRP and LRP with RV is achieved through liquid metal seals (Eutectic alloy, Cerrotru: 58% Bismuth and 42% Tin, Melting point: 139oC) and the secondary leak tightness is achieved by inflatable seals. During fuel handling, the LM seals are melted to facilitate plug rotation and frozen during reactor operation. A 30 mm thick layer of high viscous silicone oil is filled over the interseal side of LM seals to prevent oxidation when molten.

Over the years, gradual degradation of LM seals was observed resulting in loss of leak tightness and increased activity in Reactor Containment Building (RCB) due to increased cover gas leak. The silicone oil cover in the inter-seal region was found getting deteriorated and turning black. The cover gas leak rate was also found to be increasing with time. The silicone oil of SRP LM seal was found completely solidified probably due to enhanced polymerization under radiation environment. The polymerized oil along with the oxidation products of the cerrotru alloy has formed a 30 mm thick black deposit of rubbery nature over the LM seal initially and it has become hard and powdery now. Also, due to oxidation, Tin and Bismuth started separating from the alloy and Bismuth was found settling at the bottom. The chemical analysis of the existing LM seal material done indicated that the composition of the alloy has changed drastically (SRP LM seal- Bismuth: 48.2%, Tin: 50.5%, LRP LM seal: Bismuth: 52.9% Tin: 45.4%) from its original composition (Bi: 58% Tin: 42%). This resulted in loss of leak tightness of the seal.

The replacement work has been done in stages with partial quantities as complete removal of seal at one go will expose the reactor to atmosphere for several days. The retrieval and refilling set up was developed in-house. After extensive mock-up trials and improvements, the set up was deployed for retrieval and filling in reactor. In the first phase, 50 kg of LM seal was retrieved from LRP and SRP and refilling was done to demonstrate its effectiveness of the set up. Then, 50% LM seal was retrieved and fresh seal material was refilled in both LM assemblies. Again, 50% LM seal was retrieved and fresh material was filled.

## Tube leak in the Steam Generator

In 2016, during 25th irradiation campaign at 27.3 MWt power, reactor tripped on west Steam Generator (SG) leak. The triplicated highly sensitive Sputter Ion Pump (SIP) based Steam Generator Leak Detection system (SGLDS) detected the tube leak. 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 the 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 Expansion tank cover gas showed hydrogen concentration of 5% and the Plugging temperature of dumped sodium in the west loop was found to be 112°C 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 H2 conc., Hydrazine content in feed water was measured to be normal and the pump oil level was found to be steady. Conservative estimates based on the H2 accumulation and the sodium plugging temperature in the west loop indicated that the magnitude of leak was ≈ 0.9 g/s.

As the two SG modules (SGna 600A & SGna 600B) remained 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, occurred at high pressure (125 b) and temperature (460oC) and these conditions could not be recreated again for identifying the leak. Hence, a novel technique (Gas tracing) was employed to identify the leaky module. Gases Helium and Argon at 40 b pressure was admitted into SGna 600A and SGna 600B respectively. The shell side of the modules was sampled for the presence of Helium. As no He content could be detected, the gases in the two modules were reversed. With He in SGna 600B & Ar in SGna 600A, presence of He was detected in the shell side of SGna 600B. At the end of 6 h, He concentration was found to be 1042 ppm indicating that SGna 600B was leaking and the order of leak was very minute.

Replacement of the leaky module called for elaborate activities viz. cutting of water/ steam headers, sodium headers, sodium cleaning & safe disposal, maintaining the entire system in inert atmosphere during the interventions, erection of massive scaffoldings inside and outside the SG casing, handling of structures like handling structure & carrying beam, removal of hot beams, supporting of all SG modules to facilitate removal of the common support beams, modification of spare SG module to introduce welded orifice assemblies, Requalification of the preserved spare SG module by Helium Leak Testing of shell welds and tube side, removal of leaky module from the SG casing, Introduction of spare module and positioning & 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 plan were notified to AERB and due approval was obtained for the restoration plan. The requalification of the spare module fabricated more than 30 years back was successfully done. A special taskforce with personnel from operation, maintenance, technical, Quality assurance and Industrial safety sections was constituted to execute the task. The entire work could be completed in a record time of two and a half months against the original schedule of 4 months. The systems were normalized and after observing the SG operating parameters, the reactor power was raised to the target power of 27.3 MWt in that campaign.

## **replacement of leaky Steam Generator module due to sodium leak from the thermal baffle joint in the outlet header**

During 25th irradiation campaign, SGna 600B developed a tube leak as described in § 8. The defective module was replaced with a qualified 2.25Cr 1Mo spare module. This module suffered shell side leak from one of the thermal baffles to tube joint (of Tube G which was in blanked condition). The first leak was during reactor operation and the defect was rectified after establishing the pattern of defect propagation as it was found to have originated from a material defect which went unnoticed during testing of the module during fabrication. The second leak occurred during the qualification of the module after its repair at high temperature. Quantity of sodium leaked was found to be few grams on the above occasions. As the same module leaked twice, it was decided to replace the same with spare 9Cr 1Mo module. Replacement of the leaky module called for elaborate activities viz. cutting of water/ steam headers, sodium headers, sodium cleaning & safe disposal, maintaining the entire system in inert atmosphere during the interventions, erection of massive scaffoldings inside and outside the SG casing, handling of structures like handling structure & carrying beam, removal of hot beams, supporting of all SG modules to facilitate removal of the common support beams, requalification of spare SG by carrying out hydro test (1.3 times the design pressure i.e. 210 bars) at tube side and pneumatic test (at 1.1 time design pressure i.e. 56 bars) with helium at shell side, cutting and blanking of three tubes at both water and steam header, further requalification of SG module by Helium leak testing of shell welds and tube side, removal of leaky module from the SG casing, introduction of spare module and positioning & 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 (LPI), Radiography, Helium Leak Testing, hydro testing of the tube side & finally performance testing of SG at sodium temperature of 400oC for about 15 days, cooling to 180oC for 2 days and again raising the temperature to 400oC for 3 days.

Subsequent to the above leak event from SG casing, sodium leak detection system was augmented by triplication of the Sodium Ionisation Detection (SID) system. Sodium Ionization detection (SID) system is provided to detect minute sodium leak from steam generators shell side or sodium header into SG casing by sampling air from the casing. Surveillance on checking the response of SID system of SG casing is done once in two years by burning very small amount of sodium.

Global SID System was also introduced to find out sodium leak from potential sources sodium leak viz. Motorized valves etc. in the secondary sodium system.

## **modification done On the reactor protection circuit to avoid SCRAM during LOWERING OF CONTROL RODS (LOR)**

# Whenever LOR was initiated due to any parameter, invariably it was getting converted into a reactor SCRAM on high ϴi (deviation between actual and expected temperature rise across a fuel subassembly). This conversion of every LOR to SCRAM caused unnecessary thermal shock on the core components defeating the very purpose of providing LOR. As per the detailed investigation, the cause of LOR getting converted to ϴi SCRAM was due to varying response time of the core FSA thermocouples. During LOR, the FSAs with maximum response time continue to read higher temperatures; whereas the core average temperature depends on the average response time of all thermocouples. Due to this, the temperature seen by some of the thermocouples crosses the expected temperature by 10 oC leading to reactor SCRAM. As the number of FSAs increases, the target power increases and the power at which the LOR getting converted to SCRAM also increases. Inhibition of ϴi SCRAM subsequent to LOR has been done. In this scheme, ϴi SCRAM from CDPS is inhibited under the following conditions:-

1. Effective LOR is ordered (to ensure that LOR is taking place).
2. Reactivity is negative (above the alarm limit of 5 pcm), in all the three reactivity channels (This ensures that power is coming down due to LOR action).

# The modification ensures that the SCRAM on ϴi is inhibited only after ensuring that LOR is effectively taking place and that the reactor power is reducing, as indicated by reactivity channels which will be showing negative reactivity.

## **Conclusion**

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 three and a half decades of successful operation, since its inception. Towards obtaining regulatory clearance for raising the power of FBTR to its name-plate capacity of 40 MWt, design of core, reactor physics and shielding studies, safety analysis and plant dynamic studies have been carried out. The preparatory activities and major modifications are in advanced stage of completion for raising the reactor power to 40 MWt.

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