Operating Experiences of FBTR

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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 experience to improve its performance figure year after year. FBTR has been operated at different power level up to 27.3MWt/5.8MWe over the last thirty years, realizing its objectives viz; mastering sodium cooled fast reactor technology and testing future reactor materials. Based on the performance of the Mark-1 fuel, its burn up limit has been increased in steps and attained maximum burn up level of 165GWd/t. FBTR has completed 25th irradiation campaigns so far. Primary sodium temperature nearer to rated design value with reduced reactor power level was achieved by operating the Steam Generator with three out of seven tubes blanked. This paper describes the operating experiences of FBTR starting from commissioning and successful operation and various problems encountered in different systems / areas during the last thirty years viz; Spurious reactor trips due to noise pickup, Experience with failed fuel localization, Fuel handling incident, Spurious trips from Steam generator Leak detection System, Leaks from Biological concrete shield cooling system, Reactivity transients, Problems with Control Rod drive Mechanism & Core cover plate Mechanism, Dropping of orifices from once- through Steam Generator, Detection and management of sodium leak incidents.
Key Words: FBTR, sodium cooled Fast reactor, Carbide fuel,

1. Introduction

The Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, is a 40MWt 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 located 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 so as to operate the reactor at full power for experimental purposes even when turbine is not available

2. Reactor operation experience

The reactor attained its first criticality in Oct 1985 with Mark I core consisting of 22 fuel subassemblies of 70% PuC + 30% UC fuel. Subsequently, low power physics and engineering experiments up to 1 MWt were completed in 1992. After completion of commissioning of SG and its leak detection system, reactor power was raised to 10.6 MWt in Dec.'93 for the first time. Before resorting to steady power operation, a series of safety related engineering tests were conducted in 1992-93, simulating the various incidents postulated in the safety analysis. These included natural convection tests in the primary and secondary with reactor operating at low power simulating the decay heat. The results of these tests are very useful in gaining confidence in the capability of the decay heat removal systems and analyzing various transients in PFBR systems. After completing high power engineering and physics tests, reactor operation at high power was continued.

Commissioning of Turbo-Generator (TG) and its auxiliaries were completed and TG was synchronized to the grid in July '97.

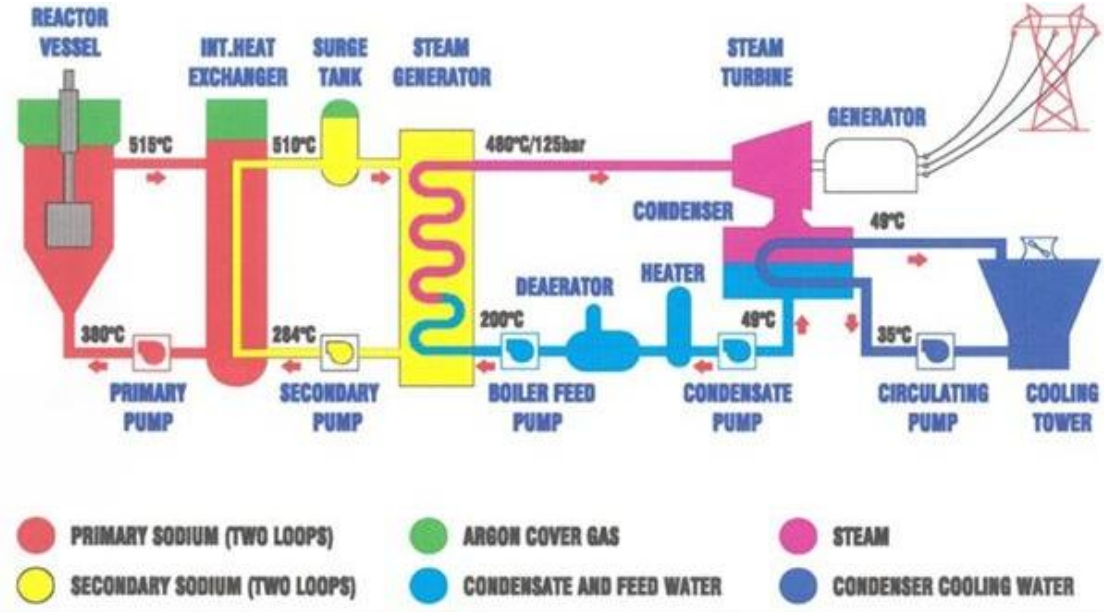

Fig: 1 FBTR heat transport system

Reactor core was gradually evolved by adding Mark II (55% PuC + 45% UC) and MOX Subassemblies (44% PuO_2 + 56% UO_2) and power was raised in steps up to 27.3 MWt. Experiments conducted showed that MOX fuel of this composition is compatible with sodium. Fuel pins of Mark I and Mark II compositions were irradiated in the reactor and discharged for Post Irradiation Examination (PIE) to assess the fuel performance. The MOX Fuel (29% PuO_2 + 71% UO_2- uranium enriched with U-233 to achieve the design linear heat rating of 450 W/cm) required for Prototype Fast Breeder Reactor (PFBR) was irradiated to 1,12,000 MWd/t to study its behavior. Post irradiation examination (PIE) of PFBR test fuel has been completed. More than 1800 pins of Mark I SA composition were irradiated up to a peak burn up of 1,55,000 MWd/t and 61 pins of a lead SA were irradiated up to a peak burn-up of 1,65,000 MWd/t so far. Towards designing and building future metallic fuelled reactors, irradiation of metallic fuel pins has been commenced from the 18^{th} campaign onwards**.** 24 irradiation campaigns have been completed so far and the 25^{th} campaign is in progress.

2.1 Reactor power control

The FBTR experience indicated that Fast reactor power could be controlled manually without any difficulty. Auto control for regulation of reactor power is not needed. Sufficient confidence has been gained on this by operating FBTR without any hassles and the same philosophy has been adopted for PFBR.

2.2 Experience with Control rod drive mechanisms:

2.2.1 Dropping of three control rods without effecting SCRAM order.

During initial period of commissioning, during start-up of reactor, there was an incident of three control rods dropped into the core without actuation of effective scram order. The investigation revealed spurious SCRAM signal in start up channels due to noise pulse (ON time of noise pulse < latching time of scram circuit) resulted in discharging and charging of EM coils of CRDM and progressive decrease in EM current and dropping of control rod. The latching time of the scram circuit was more (57 ms) compared to the response time of Electro magnet (45 ms to 72 ms). The latching time was modified to 40 ms. Based on this incident, a new LOR parameter “low current in EM coils” was incorporated.

2.2.2 Uncontrolled withdrawal of control rod:

There was an incident of one of the control rods continued to raise even after the raise command push button was released during approach to criticality in 1987. The control rod over shot by 9mm and manual LOR was ordered to shutdown the reactor. The cause of overshooting of control rod was due to sluggishness of the "Raise" contactor. Based on this incident, a new LOR input "Control Rod Level Discordance' was introduced.

2.2.3 Reverse rotation of CRDM motor during maintenance:

Any open circuit in power supply circuit of CRDM motor due to failure of any component namely fuse or thermal over load element, should not result in reverse rotation of CRDM. In one incident, investigation revealed that there was a reverse rotation of power drive of one of the CRDM motor during failure of overload relay. Burning of OLR (over load relay) coil of one of the phase of 415 V power supply of CRDM motor resulted in motor getting power supply through brake power supply. This caused change in phase sequence of power supply to the motor and resulted in reverse rotation of CRDM motor. To prevent similar incident, the brake power supply was tapped from upstream side of OLR and contacts of raise/ lower contactor with suitable interlock to energises/ de-energise the brake when raise/ lower contactor is energized/ de-energised were wired in the brake coil circuit. Also, the tapping for the brake power supply was staggered in different phases (R-Y-B) for the CRDMs to avoid common mode failure. The brake power supply was tapped across the same phases as the control power supply for the logic circuit of any mechanism. As part of safe operation practice, the operability of Control Rod Drive Mechanism (CRDM) is ensured by on-power friction force measurement.

2.3 Optimization of reactor trip parameters

Based on the operating experience, reactor trip parameters were optimized by modifications/additions/deletions in Scram and LOR circuit to avoid spurious trips and achieve maximum availability without compromising the safety of the reactor. For instance, "Low current in EM coil" LOR parameter was included to address dropping of control rods without SCRAM and "Control rod level discordance" was added as LOR input to prevent uncontrolled withdrawal of control rod. The incident analysis revealed that "Control rod level discordance" LOR parameter is sufficient to protect reactor against dropping of control rods without SCRAM and uncontrolled withdrawal of control rod. Hence, "Low current in EM coil" was deleted as LOR parameter but retained as an alarm.

2.4 FBTR Fuel irradiation with stage wise enhancement of LHR

The reactor was initially loaded with a small Mark I fuel core rated for 10.5 MWt at a Linear Heat Rating (LHR) of 250 W/cm. Being an untested fuel, the target burn-up was initially set at 25,000 MWd/t. The LHR and target burn-up values have been progressively enhanced to 400 W/cm and 1, 55, 000 MWd/t based on PIE of Mark I fuel at different burn-up levels[1]. At each stage of LHR & burn-up enhancement, rigorous theoretical analysis was carried out and safety clearances are being obtained.

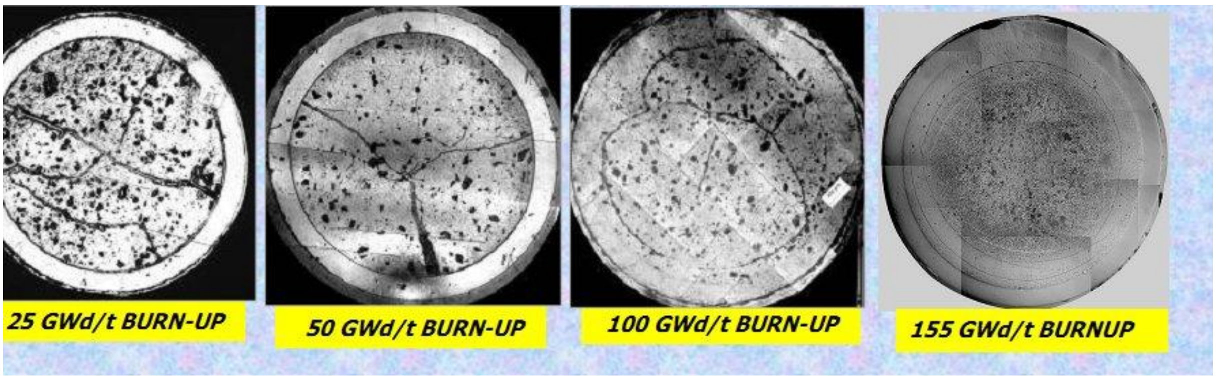

Fig 2 .Micrographs of MK-1 fuel at different burn up level

The lead Mark I fuel (70% PuC+30% UC) subassembly was irradiated upto the burn up of 1,65,000 MWd/t without fuel clad failure. The fuel has also been successfully reprocessed at IGCAR. The successful closure of the fuel cycle was also demonstrated when a fuel subassembly with Pu recovered from FBTR fuel was loaded back into the core of FBTR.

2.4.1 PFBR test fuel pin irradiation

As the fuel used in FBTR and PFBR are of different compositions, PFBR test fuel was irradiated to 1,12,000 MWd/t at the rated LHR of 450 W/cm. PIE of this subassembly was also completed. During the 15th irradiation campaign, one experimental fuel pin with MOX fuel pellets of PFBR composition was irradiated in FBTR for 13 days at a linear power of 400 W/cm for understanding the beginning of life gap closure behavior**.**

During PIE, the measurements at different locations indicated that the apparent gap had reduced from average pre-irradiation value of 75-110 microns to uniformly around 20 microns in all the locations. The gap reduction during beginning of life indicates the feasibility of increasing the LHR of PFBR fuel to the design value of 450 W/cm after the initial pre-conditioning of approximately 20 EFPD.

2.5 Experience with Fuel clad failure incident

During the 17th irradiation campaign at 18 MWt, one Mark I subassembly which has reached a burn-up of 148 MWd/t failed. The DND contrast ratio between west & east side during the reactor scram was observed to be more than 4.5 and the predicted burn up of the failed fuel subassembly based on the observed Kr^{85}/ Kr^{88}activity ratio was more than 100 GWd/t. Hence, from the above, it was inferred that any one of the highly burnt fuel subassemblies located in the west side of 3rd ring as the failed subassembly. By neutron flux tilting method, the failed fuel was identified and further confirmed by operating the reactor at high power after shifting it to the storage location. The subassembly was discharged from the reactor after its decay heat came down to acceptable limit for dry storage. PIE of failed SA indicated random failure.

2.6 Experience with Sodium systems [1][2][3][4][5]

Sodium systems have been operating for the past thirty one years and their performance has been excellent. The impurity levels in sodium was always <0.6 ppm and it was demonstrated that even without purification system in service for about 60 days, the impurity levels in primary system remained within limits. During commissioning of steam generator, one cold trap in secondary sodium loop had to be replaced due to impurity loading at the time of connecting the SG to the loop. One secondary sodium pump was replaced after 10,000 h of operation due to abnormal noise. Performance of all other pumps till now has been very good. There was no incident of oil leak from the pump seals to the sodium circuit so far. Performance of sodium pump drive system was not satisfactory initially. It improved significantly after air conditioning the control logic panels and carrying out certain logic modifications. The primary sodium was sampled periodically for trace element analysis and the nuclear grade purity was found to be well maintained. An Electro-chemical carbon meter is installed in one of the secondary sodium loops on experimental basis to measure the active carbon level in the system. Active carbon content was found within the limits.

There were incidents of ingress of mercury from the relief pot of primary cover gas system into the primary sodium system during vacuum pulling operation of primary sodium storage tank and CRD compressor operation with stuck closed NRV. This problem was overcome by modifying the layout of relief line from primary sodium storage tank and directly connecting the CRD compressor discharge line to effluent header.

2.5.1 Primary sodium leak incident[1][3]

In April 2002, while reactor was operating at 17.4 MWt, there was an incident of leak of 75 kg of primary sodium from the purification circuit. The leaked sodium had frozen on the cabin floor and pipelines and was manually cut and scooped out under inert gas purging. Leak was from the body of a 20 NB bellows-sealed valve, through one of the three blind holes used by the manufacturer for machining the valve body(Fig 3).

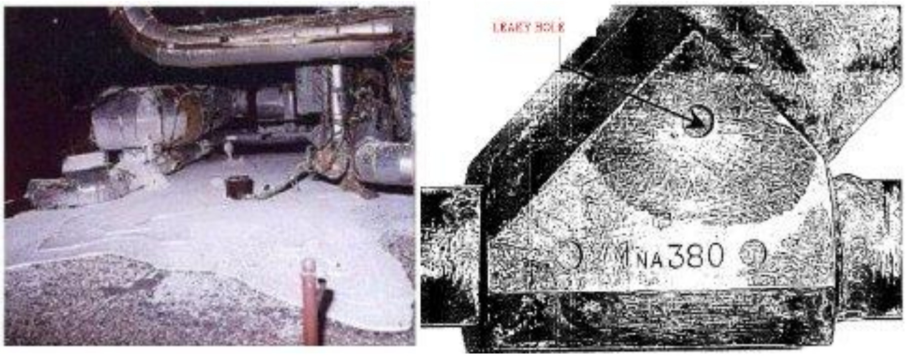

FIG.3 Sodium leak inside purification cabin due to hole in the valve body.

The valve was replaced. Since the problem is generic to the specific make, valves of this make used in the plant were inspected and similarly made valves were rectified by welding tight fit plugs. The sodium which leaked during the incident was converted to hydroxide, neutralized by ortho-phosphoric acid and disposed off as active liquid effluent. It is gratifying to note that a material thickness of just 0.1 mm was enough to hold sodium for 17 years which indirectly indicate the maintenance of high purity level in the Primary sodium system.

In one incident sodium leaked past the failed bellows came out through a crack in the weld joint of the nipple used for mounting spark plug detector. Later the faulty valve was cut and replaced with new valve.

2.5.2 **Sodium leak from nickel diffuser of West SGLDS** [1]

In Feb 2006, there was a reduction in sodium flow in one of the Steam Generator Leak Detection (SGLD) circuit (Fig 4). Heavy accumulation of frozen sodium was found out in the shell side and vacuum line of the nickel diffuser. Sodium has leaked from the tube to tube sheath weld joint of the nickel diffuser and got frozen in the shell side and pipe lines. This compressed and flattened the nickel diffuser tubes resulting in low flow. As no sodium leak detection system was provided in the vacuum lines, the leak has gone unnoticed.

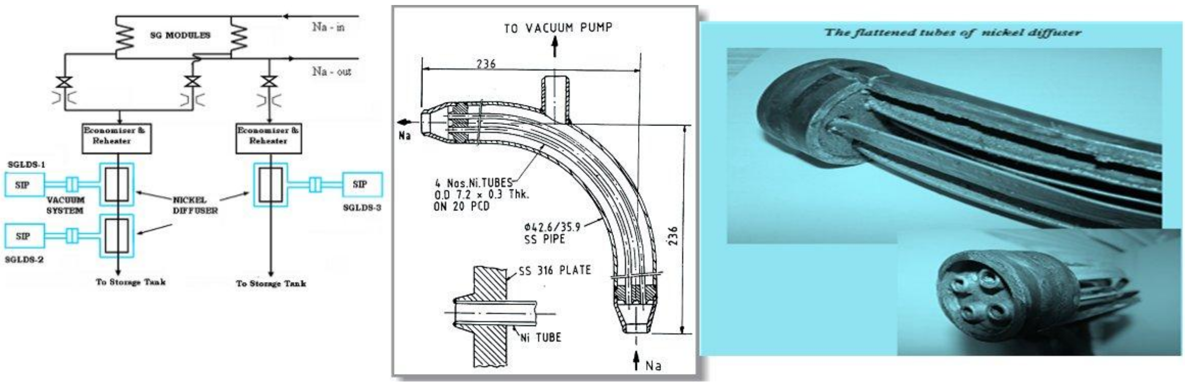

FIG.4 Triplicated SGLDS system and flattened tubes of Nickel diffuser

Hence, Mutual Induction (MI) type probe was provided downstream of the Nickel diffuser in the vacuum line to detect any leak in the initial stage itself.

2.5.3 Modification in the surge tank level maintenance circuit

In secondary sodium main circuit, the surge tank remained connected to expansion tank through a communication line with a motorized valve and a bypass line across it (Fig 5). A continuous flow of hot sodium from surge tank to expansion tank maintains this line hot and available for communication whenever needed.

As the hot sodium from surge tank mixes with the cold sodium in expansion tank, there was a possibility of thermal shock in the expansion tank where the hot sodium gets mixed with cold sodium. Identical arrangement existed in French reactor Pheonix and leaks were reported in this location due to thermal striping. In order to prevent recurrence of such a problem in FBTR, the sodium communication line with integral bypass line was converted to a hot argon communication line by modifying the circuit with a pneumatically operated valve which will open whenever communication is required for level maintenance or for fast dumping.

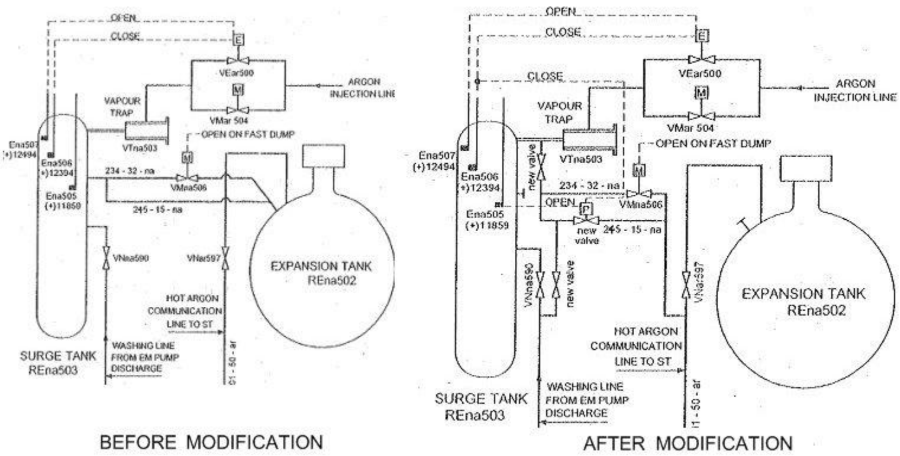

FIG. 5. Modification in surge tank.

2.5.4 Replacement of rupture disc assemblies in the secondary sodium [1]

Rupture disc (RD) assemblies are provided in the inlet and outlet sodium headers of steam generators and in the cover gas region of expansion tank to protect steam generator and IHX bottom tube sheet during a sodium water reaction. As per the technical specifications for FBTR operations, the RDs in the secondary sodium system are to be replaced periodically. As FBTR was operating at low sodium temperature and purity level of sodium has been maintained extremely well, there was no concern from corrosion effect at elevated temperatures and hence there was no need to replace rupture discs.

However, the regulators recommended replacement of rupture discs in one loop and subsequent burst testing of the removed rupture discs in order to ensure that there is no deviation in the set value due to ageing of the material. Accordingly, state-of-the-art scored type rupture disc assemblies were procured and replacement was done in the secondary west loop. The removed RD assemblies were burst tested and found to be rupturing at the design pressure even after 25 years of service.

2.5.5 Choking of hot argon line in the primary sodium system

Choking of hot argon communication line of primary capacities was encountered few times in the nozzle portion of the primary sodium storage tank due to deposition of sodium vapours in the cooler region. The cause of the loss of communication between storage tank and other primary sodium capacities is due to sodium deposition and choking near the storage tank nozzle. The choke was removed by cleaning after cutting the line. In order to keep the nozzle portion hot, heaters are laid in and around the nozzle portion. Also the humps and bends in the hot argon communication line were removed.

2.6 Experience with Reactor Assembly[2][3][4]

2.6.1 Deflection of reactor vessel during commissioning

During commissioning in 1985, when the sodium temperature was progressively raised to 350°C for isothermal tests, large azimuthal temperature difference (~ 80°C) in the cover gas region of reactor vessel (RV) was noticed. This resulted in tilting of reactor vessel and shift in grid plate as measured by displacement measuring device. This was investigated to be due to non-uniform natural convection currents in cover gas space. This was overcome by injecting helium to the argon cover gas to form a double layer above sodium level to suppress the rising convection currents. This works on the simple principle that heavy argon, getting heated up by the hot sodium, tries to rise up and as it encounters the less dense helium at the top and cannot raise any further. The convection gets only localized. This was found to be effective in bringing down the grid plate deflection and the temperature gradient in RV.

2.6.2 The Fuel handling incident[3]

During an in-pile fuel transfer for performing a low-power physics experiment in May 1987, a major fuel handling incident took place. The incident was due to a plug rotation logic remaining in bypassed state during fuel handling, resulting in the Rotatable Plugs being rotated with the foot of a fuel subassembly protruding into the core during the transfer. Reactor operation could be resumed only in May 1989. A mechanical swiveling lock, to keep the SA firm in the transfer position has since been installed in transfer flask. This condition has also been wired to the logics of plug rotation and fuel handling flasks.

2.6.3 Cocoon surrounding the reactor

A metallic cocoon was installed over the fixed floor with sealing arrangement covering LRP & SRP floors to reduce the background activity surrounding the top of the reactor. An exhaust blower was connected to the cocoon to suck the accumulated leaking gas surrounding the reactor and discharge it to RCB exhaust duct. With this arrangement, there was significant reduction of air activity level inside RCB

2.6.3 Water leak from BSC coils inside biological shield concrete [1][3]

The reactor vessel of FBTR is surrounded by a safety steel vessel and further by two types of concrete namely 600 mm thick biological shield and 900 mm thick structural concrete. The biological shield concrete is cooled by circulating water through 180 coils embedded in concrete. Crevice corrosion at the socket welds is considered to be the cause. Hence, proprietary formulation sealants were injected to arrest the leak points in the coils of the two sub headers. Following these incidents, there was six more leak incidents in different coils and was chemically sealed indigenously. After arresting the leaks, the leak rate from BSC system has come down drastically within acceptable limit.

2.6.4 Malfunctioning of Core Cover Plate Mechanism (CCPM)[1][3]

The Core Cover Plant 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 sub assemblies. The mobile plate got struck in the fuel handling position in 1996 and this could not be rectified.

The effect of CCPM struck at 75mm position resulted in a plenum hydraulics where in the hot sodium from the inner rings glided over to cooler sodium from outer rings of the core. Thus the sub-assemblies in the outer ring (third ring on wards) were reading much higher temperatures than the actual temperatures. Theoretical studies indicated that the detectable plugging levels were lower than allowable plugging levels and hence there was no safety concern. An eddy current flow meter was used for periodic checking of actual flow through fuel sub-assemblies during fuel handling state.

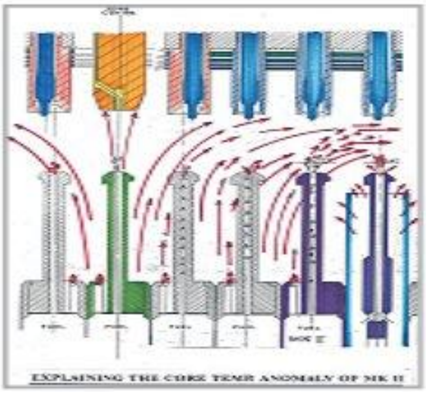

FIG.6. Schematic of mixing of streams with CCPM at 80 mm

2.6.5 Interchange of core thermocouples[4]

There were interchanges of core thermocouples between two sets of fuel and nickel sub assembly positions. These interchanges core of thermocouples were resolved in 1990 by replacing the affected nickel SAs one by one by an overcooled dummy SA and monitoring the temperature rise after start-up and raising the reactor power.

2.6.6 Trailing cable[5]

Frequent failures of cores of the trailing cable during plug rotation has been experienced over the years, resulting in bypassing of logics, replacement of failed cores with spare cores and replacement of cables after exhausting all spare cores. The cause of these failures was analysed and found to be due to the tortuous path of the trailing cables, resulting in multiple bents and large number of pulleys. The layout of the trailing cables was simplified by rotating the cable tower by 90^{o} and routing the cable in straight line from LRP to cable tower eliminating two sharp bends. This modification significantly reduced cable core failures and frequent replacement of the trailing cables.

2.7 Reactivity transients[4]

In 1998, when reactor was being operated at 8 MWt for irradiation of Zirconium-Niobium (Zr-Nb) alloy, reactivity transient which is repeatable in nature was encountered. The transient was self limiting and the power remained stable after the transient. The reactivity gain (30 to 40 pcm) while raising the power was lost while lowering the power. There was no permanent gain in reactivity. The power increase was around 700 to 800 kWt. The onset of transient occurred at a specific value of mean temperature gradient across the core ($\Delta\theta m$) of 90 to110^{o}C for a given primary sodium flow. The transient can be observed at any power level if the above $\Delta\theta m$ is reached. At primary sodium flow of equal to or more than 460 m^3/h, the transient did not occur. From the experiments conducted, it was concluded that the transient was due to slight thermally induced geometric changes in the core which happened at low flow rates.

2.8 Experience with steam generator leak detection system[1]

Sputter ion pump current which is a measure of total pressure in vacuum system responded in a manner similar to mass spectrometer signal during calibration of the system. Hence sputter ion pump current instead of mass spectrometer signal was used to initiate reactor trip in case of feed water leak in steam generator. With this, the need for periodic replacement of filament of mass spectrometer and consequent intervention in vacuum circuit and reactor down time were eliminated. SGLDS is a complex system having an ultra high vacuum system and complicated electronic signal processing circuit. On several occasions, there were spurious spikes in signal leading to reactor trip. As proving any such increase in signal as spurious involves elaborate testing and qualification, the system was triplicated and trip initiated on two out of three coincidence logic.

2.9 Experience with steam & water system

2.9.1 Orifice dislocation in SG

During the operation at 18.6 MWt in the 15th irradiation campaign, large variation was observed among the steam temperatures from the four SG modules. The bulk steam outlet temperature was only 430˚C as against 450˚C estimated. Investigation revealed dislocation of the spring loaded orifice assemblies (Fig:7) provided at the entry of the water tubes for providing flow stability to the steam generator modules. The spring loaded orifices were replaced with welded type orifices.

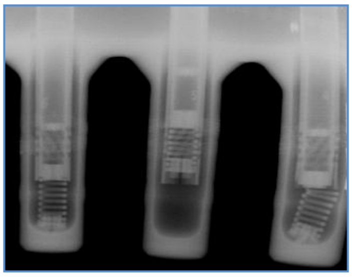

Fig:7 Dislocation of orifice in SG

2.10 Experience with detection of SG tube leak

In Oct 2016, when the reactor was operating at 27.3 MWt /5.8 MWe, there was an increase in Hydrogen level in west loop secondary sodium. This was detected by increase in Sputter Ion Pump current in all the triplicate channels of SGLD system. Subsequently, SGLDs system initiated Auto LOR due to Hydrogen level high parameter, sensed by increase in SIP current in two out of three channels. Following the LOR, West SG safe configuration took place on auto and nitrogen injection to SG tubes took place as per design intent.

2.10.1 Confirmation of SG tube leak:

Hydrogen and nitrogen content in expansion tank cover gas sample was analyzed and found to be 5 % & 6 % respectively after two hours of reactor trip. In order to rule out oil leak from secondary sodium pump as the cause for increase in hydrogen content, Methane in cover gas was also analyzed and found to be nil. Also oil level in secondary west sodium pump was steady. In order to rule out contribution from Hydrazine in feed water, SG inlet feed water sample was analyzed for hydrazine content and found to be less than 20ppb (normal). Plugging temperature of secondary west loop sodium was taken and found to be 112°C after 3 hrs of reactor shutdown. Thus all the investigations indicated that there is a genuine leak in west SG. To further confirm the tube leak, helium leak test was done by pressurizing tube side with He up to 5 bar and presence of helium was checked in the SG shell side after isolating both shell and tube side boundary valves. The sample showed 272 ppm of Helium confirming the tube leak.

2.10.2 Identification of defective SG module:

In each Secondary loop there are two SG modules and there is no individual isolation valves (both water/steam and sodium side) for identification of defective module. Gas tracing technique was employed to ascertain the defective module**.** In this technique, two different gases namely Helium and Nitrogen/Argon gas were filled at known pressure in the tube side of either modules (injected at the bottom of SG module flow meter tapings) and observed for the presence of Helium or Nitrogen/Argon in the SG shell side. The maximum duration of water/steam leaking into sodium side is estimated to be less than 10 minutes (i.e 3 minutes of transportation delay, 3 minutes for reaching the threshold value from background value for taking SG safe configuration safety action and another

3 minutes for SG depressurization). The defective SG module was replaced with spare SG module.

Fig: 8 Replacement of defective SG module

3.0 CONCLUSION

FBTR has been in operation for more than 30 years. The experience and confidence gained in operating FBTR gave the confidence to leapfrog to a stage where currently about 1400 T of sodium has been charged into the storage tanks of PFBR without any incident. The successful operation of FBTR, its excellent track record of nuclear and radiological safety and the design improvements based on the feedback from FBTR have helped in designing fast reactors of higher powers. At present the confidence level of the country in depending on fast reactors as a future energy source has become high.

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