Two Decades of Operating Experience with the Fast Breeder Test Reactor

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Abstract. Sodium cooled fast breeder reactors constitute the second stage of India's three-stage nuclear energy programme, for effective utilization of the country's limited reserves of natural uranium and exploitation of its large reserves of thorium. The Indira Gandhi Centre for Atomic Research was established at Kalpakkam, 80 km south of Chennai (India) in 1971, with the mission to develop the technology of sodium cooled fast reactors. The heart of this Centre is the Fast Breeder Test Reactor (FBTR), a sodium cooled test reactor, built to serve as a test bed for irradiation of fast reactor fuels and materials and to provide experience in large scale sodium handling and fast reactor operation. FBTR was built on the lines of the French Rapsodie-Fortissimo reactor. It went critical in Oct 1985 with a small core having 22 fuel subassemblies having a unique, high Pu monocarbide fuel. The core has been progressively expanded by loading additional fuel to compensate for reactivity loss due to burn-up.The current core has 48 fuel subassemblies of three different compositions, in addition to a test fuel subassembly simulating the composition of MOX fuel of the power breeder reactor. The high Pu carbide has reached a burn-up of 155 GWd/t without clad failure. During more than two decades of operation, FBTR has provided valuable experience which has been a major feedback in the launching of the 500 MWe Prototype Fast Breeder Reactor (PFBR) at Kalpakkam. The paper gives an overview of the operating experience of FBTR, including the performance of the systems, experiments carried out, major incidents and major system modifications to improve the reactor availability.

1. Introduction

The Fast Breeder Test Reactor is a 40 MWt, loop type, sodium cooled fast reactor. Fig.1 shows the schematic flow sheet of the heat transport circuits. 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 common steam water circuit comprising a turbine-generator and a 100% dump condenser. A brief description of the systems and equipment is available elsewhere [1].

The original design characteristics of the reactor are given in Table- 1. As per the initial design, the core was rated for 40 MWt with 65 MOX fuel subassemblies of Rapsodie driver fuel composition (30% PuO_2 and 70% UO_2, with the latter enriched in U^{235} to 85%). Due to non-availability of enriched uranium, the option of using an alternate fuel rich in Pu was studied. Compatibility of higher concentrations of PuO_2 (>30%) with sodium and difficulties in fuel fabrication led to the choice of the carbide fuel for the first core of FBTR. The fuel chosen was a mixed carbide fuel with 70%PuC & 30%UC, designated as Mark-I fuel. Being a unique fuel of its kind without any international parallel or irradiation data, the core was redesigned as a small carbide core, and it was decided to test it directly in the reactor as the driver fuel. Hence, as against the original design of 65 MOX fuel subassemblies rated for 40 MWt, the small carbide core was designed with 27 fuel subassemblies and was rated for 10.6 MWt. The carbide fuel was developed by Bhabha Atomic Research Centre, Mumbai [2].

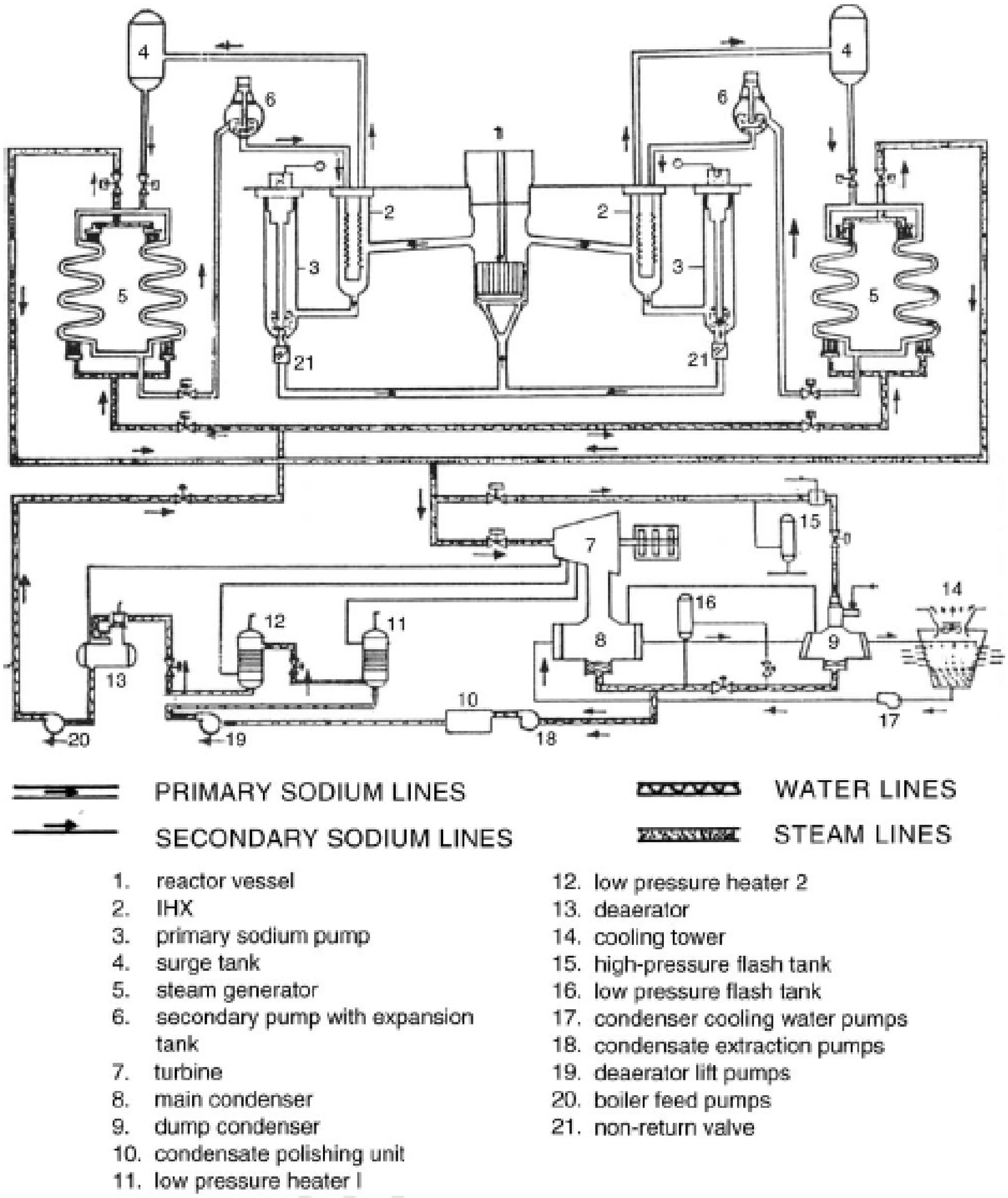

FIG. 1. Flow-sheet of FBTR.

2. Evolution of the Core

FBTR went critical on 18th Oct 1985 with the small core of 22 fuel subassemblies of Mark-I composition. The core has since been progressively enlarged by adding fuel subassemblies to compensate for reactivity loss due to burn-up. With a view to raise the reactor power to 40 MWt, it was decided, in 1995, to go in for a full carbide core of 78 fuel subassemblies of 55 % PuC + 45 % UC composition (designated as Mark-II fuel). Fuel subassemblies of Mark-II composition were inducted surrounding the Mark-I subassemblies in 1996. It was also decided to retain the Mark-I fuel till these subassemblies reach the maximum possible burn-up. Since the performance of the Mark-I fuel has been excellent, reaching a burn-up of 155 GWd/t without failure, full expansion to the Mark-II core could not be completed till now. In 2005, it was decided to induct high Pu MOX fuel in order to gain experience in the fabrication and reprocessing of MOX fuels, since MOX fuel will be used in the next few power breeder reactors in India. Eight MOX fuel subassemblies with 44 % PuO_2 were loaded in FBTR in 2006. These subassemblies have 37 fuel pins each, with the fuel pin and pellet diameters same as in the 500 MWe Prototype Fast Breeder Reactor (PFBR), coming up at Kalpakkam. In additon, a 37 pin fuel subassembly simulating the MOX fuel composition of PFBR (29% PuO_2) is also undergoing test irradiation. The current core of FBTR, therefore, has 49 fuel subassemblies- 27 Mark-I, 13 Mark-II, eight MOX subassemblies and one PFBR fuel test subassembly.

Table 1. Original Design Characteristics of FBTR

Parameters	**Original Design Values**
Reactor type	Sodium cooled, loop type
Reactor Power	40 MWt / 13.2 MWe
Fuel	MOX (30% PuO_2 + 70% UO_2 with 85% enrichment)
No. of Fuel Subassemblies	65
Fuel Pin diameter	5.1 mm
No. of pins / Subassembly	61
Maximum Linear Heat Rating	400 W/cm
Maximum burnup	100 GWd/t
Peak Neutron flux	3.5 E15 n/cm^2/s
No. of control rods	6
Control Rod Material	B_4C (90% enriched in B^{10})
Reactor inlet sodium temperature	380^{o}C
Reactor outlet sodium temperature	515^{o}C
Primary Sodium Flow	1100 m^3/h
Feed water temperature	190^{o}C
Steam Temperature	480^{o}C
Feed water flow	70t/h
Steam Pressure	125 kg/cm^2
Sodium Inventory	150 t
Steam Generator	Once-through type, 7 tubes in shell, serpentine shaped
Turbine-generator	16 stages, condensing type, 16.4 MWe, air cooled.
Bypass circuit	100% Dump Condenser

These core changes, vis-à-vis the original design, have resulted in the core power being limited to 17.4 MWt so far, with a corresponding reduction in Heat Transport System parameters. Typically, the maximum reactor inlet & outlet temperatures have been 350˚C & 444˚C respectively,as against the respective design values of 380˚C & 510˚C. The steam temperature is about 425˚C, as against the design value of 480˚C. .

Fig. 2 give the configuration of the core at first criticality and as of now.

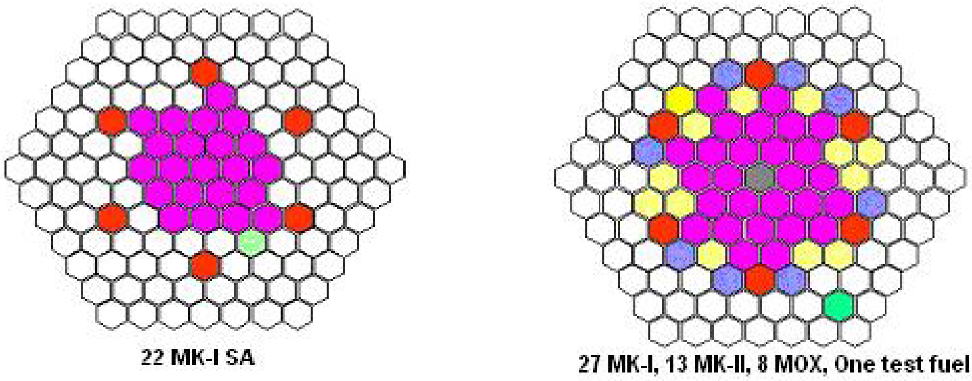

FIG. 2. Core configuration at first criticality and as of now

3. Operational History

Construction of FBTR started in 1972, and civil works were completed by 1977. Most of the components were manufactured by Indian industries, and were installed in 1984. Commissioning was done in phases. Initially, the primary and secondary sodium systems were commissioned, without the steam generators in place. After first criticality on 18th October 1985, low power physics experiments were conducted. During an in-pile fuel transfer for measuring the worth of the 23rd fuel subassembly in May 1987, a major fuel handling incident took place, discussed in para 8.0. Reactor operation could be resumed only in May 1989. The various major milestones crossed subsequently are given in Table-2. The cumulative operational statistics of the reactor, as of 31st March, 2007, are given in Table-3.

4. Fuel and Material Irradiation

In addition to being a self-driven irradiation facility for the Pu rich monocarbide fuel, the reactor has been utilised for studying the irradiation creep behaviour of Zr-Nb being used in the Indian Pressurised Heavy Water Reactors. The present mission of FBTR is to irradiate the MOX fuel (29 % PuO_2) chosen for PFBR to the target burn-up of 100 GWd/t at the design linear heat rating of 450 W/cm. To obtain this LHR with the lower flux level of FBTR, U^{233} has been added to the fuel. The test fuel has so far seen a burn-up of 62 GWd/t. Recently, a single pin of this composition was irradiated for 20 days at 400 W/cm to study its initial restructuring behaviour, since this is found to have a bearing on the economics of PFBR. Long term irradiation of D-9 alloy, which is the current core structural material for PFBR, has been started. Short term irradiation of SS-316 material used in the Grid Plate of FBTR is in progress as a part of material surveillance programme for the life extension of FBTR.

Table 2. Milestones

Milestone	**Date**
First criticality	18th Oct, 1985
Sodium valved in into Steam Generators	Nov `89
Water valved in into Steam Generators	Jan `93
Power of 10.5 MWt	Dec `93
High Power Safety Related experiments	94-95
MK-I reaches a burn-up of 25 GWd/t	May `96
Start of Loading of MK-II fuel	Oct `96
TG synchronised to grid	July `97
Zr-Nb irradiation	98-99
MK-I reaches a burn-up of 50 GWd/t	Apr `99
Power of 17.4 MWt	March 2002
MK-I reaches a burn-up of 100 GWd/t	Sep 2002
Start of PFBR Test Fuel Irradiation	July 2003
MK-I reaches the target burn-up of 155GWd/t	July 2006
Loading of Eight nos. of high Pu MOX	Feb 2007

Table 3. Cumulative Operational Statistics (as of 31st March 2007)

Parameter	**Value**
Total Operating time (h)	38313
Thermal energy produced (MWh)	279314
TG synchronisation time (h)	5285
Electrical energy generated (MWh)	5468
EFPD @ LHR of 320 W/cm (d)	856
Cumulative operation of four Sodium Pumps (h)	581,691
Steam Generator operation (h)	21529
Reactor Trips	411

5. Physics & Engineering Experiments

In addition to routine measurements of control rod worths and reactivity coefficients of inlet temperature and power after every core configuration change, physics experiments carried out are: reactor kinetics experiments, void coefficient measurements, response of delayed neutron detection system to detect clad failure and flux mapping in sodium above core. A series of safety related engineering tests was conducted in 1994-95, basically to validate the codes used in incident analysis. Normal plant incidents like off-site power failure and tripping of one pump in the primary, secondary or tertiary loops were studied 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. As a precursor to the station black out test, natural convection tests in the secondary and primary loops were separately carried out. In view of the high burn-up logged by the Mark-I fuel, the probability of fuel failure could be high. In order to validate the performance of the Delayed Neutron Detection System and verify the capability to identify any failed fuel, we recently conducted a series of experiments with a special subassesmbly with 19 perforated pins of natural uranium. The results of the experimens were satisfactory, and confirmed that the failed fuel could be easily detected by the DND system and the failed fuel location could be easily inferred from the contrast ratio of the counts from both the loops. As a part of the life extension programme, the fast flux at the location of the Grid Plate in the reactor has been measured.

In the next few years, FBTR will be used for conducting a series of tests, including validation of devices and systems used in PFBR. These include the neutron detectors, in-core detctors and acoustic probes planned to be deployed in PFBR.

6. Performance of the Carbide Fuel

Perhaps, the most gratifying is the performance of the Mark-I fuel [3]. Based on out-of-pile characterization, the fuel was initially rated for a peak LHR of 250 W/cm and allowable burn-up of 25 GWd/t. Based on further out-of-pile simulation, its LHR rating was later raised to 320 W/cm.

One Mark-I subassembly was discharged in July 1996 at a burn-up of 25 GWd/t [4]. The Post-Irradiation Examination (PIE) indicated the clad diameter to be within the original tolerance limits and there was no clad deformation. Metallographic examination indicated closure of fuel clad gap by fuel cracking. The stack length had increased by a maximum of 1.67%. The average fuel swelling rate was estimated as 1.2 % per atom percent burn-up. There was no measurable increase in the flat-to-flat distances of the hexagonal wrapper. Based on the observed reduction in the fuel-clad gap, designers raised the burn-up limit to 65 GWd/t and the LHR limit to 400 W/cm for burn-up exceeding 35 GWd/t.

Another subassembly was discharged at a burn-up of 50 GWd/t in 2000. PIE indicated no indication of fuel clad mechanical interaction. The fuel-clad gap was about 20-30 microns, which could accommodate further swelling of fuel. No restructuring of the fuel was noticed. The micro structure indicated that the swelling is accommodated by filling up of porosity. The radial cracks in the fuel pellet were still present. The linear fuel swelling rate was estimated as 1.5%. There was no significant increase in the hardness of the clad. The maximum fission gas release was estimated as 5-6%. There was no measurable dilation of the hexagonal wrapper.

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

430°C was 3%. Based on these observations, the allowable burn-up of the unique carbide fuel was further raised. To take care of any difficulties in discharging the SA due to wrapper dilation, extraction force was periodically monitored. The fuel reached 155 GWd/t burn-up without failure in July 2006. Fig. 3 gives the photomicrographs of the fuel at 25, 50 & 100 GWd/t burn-up and fig.4 gives a view of the pin bundles inside the hot cells.PIE of a subassembly at 155 GWd/t burn-up is in progress. The current indications are that further extension of burn-up will be difficult from considerations of residual ductility of clad and wrapper. It is planned to continue to irradiate one subassembly beyond this burn-up to study the endurance limit of this unique fuel and its behaviour on breach of clad.

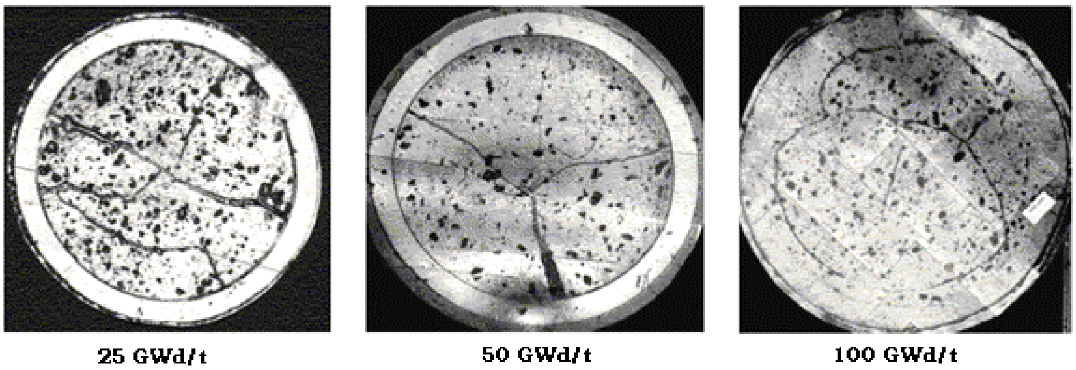

FIG. 3. Photomicrographs of Mark-I Carbide Fuel at Various Burn-up Levels

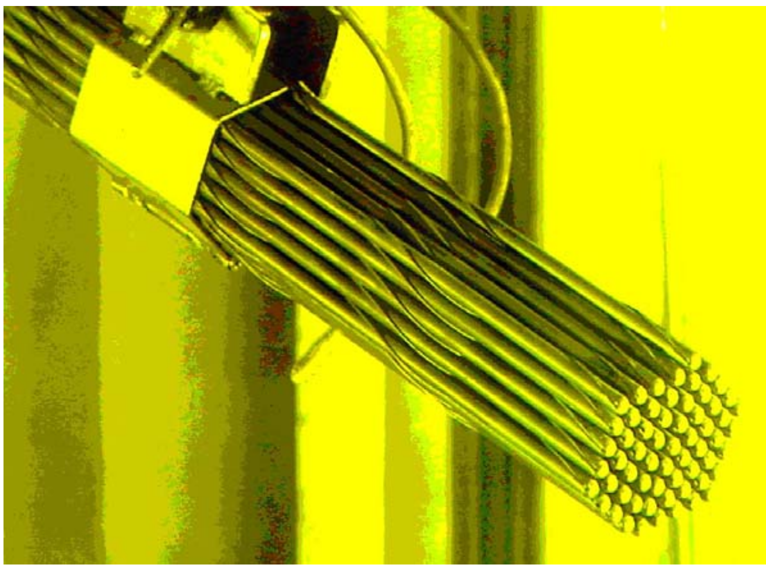

FIG. 4. Pin Bundle Inside the Hot Cells

7. Performance of the Reactor Systems

Performance of the safety critical and safety related systems has been excellent. Each sodium pump has operated for more than 125,000 h without any problem. Sodium purity is so well maintained that when viewed inside the reactor through the periscope, it shines as a mirror, so much so that many of the reactor internals could be inspected from their reflections in sodium. Fuel pins removed from the reactor after 18 years of residence time retain their shining appearance and even the identification numbers engraved on the pins could be easily read in the hot cells during post-irradiation examination.

Argon cover gas purity is so well maintained that even with the cold traps out of service for several months, the plugging temperature of sodium is found to be below 105˚C. The steam generators, which are critical for the success of fast reactor programme, have operated for more than 21000 h without any leak. It may be noted that the steam conditions-viz. pressure and temperature- in FBTR are the highest among all the reactors in India. The high level of feed water purity demanded of these once-through type steam generators has been successfully maintained. The TG has been in service for more than 5000 hours at off-normal steam conditions (420˚C as against the design value of 480˚C, flow of 20 t/h as against the design value of 70 t/h) conditions. Inspection of the internals indicates no erosion of blades or diaphragms.

8. Major Incidents

There were two major incidents viz. the fuel handling incident in 1987 and primary sodium leak in 2002. There were also three incidents of reactivity transients.

8.1. Fuel Handling Incident

During an in-pile fuel transfer for performing a low-power physics experiment in May 1987, a major fuel handling incident took place [5]. The incident was due to the plug rotation logic during fuel handling remaining in bypassed state, resulting in the Rotatable Plugs being rotated with the foot of a fuel subassembly protruding into the core during the transfer. The foot of the subassembly as well as the heads of 28 reflector subassemblies on the path of its rotation got bent. During the several manoeuvres to charge the subassembly, one reflector subassembly at the periphery got ejected by the bent foot of the fuel subassembly. During subsequent plug rotation, the sturdy Guide Tube which guides the gripper of the Charging Flask entangled with the ejected reflector subassembly and got bent by about 320 mm. The damaged fuel subassembly was retracted using extra force. The Guide Tube was cut below a set of equalisation holes using a specially designed remote cutting machine. All care was taken to ensure that the cutting chips do not fall into the reactor. The Guide Tube was removed in two pieces (fig. 5). The damaged reflector subassemblies in the path of rotation were identified using periscope with sodium drained to uncover the subassembly heads, and removed using specially designed grippers. Reactor operation could be resumed only in May 1989.

8.2. Primary Sodium Leak

In April 2002, there was an incident of leak of 75 kg of primary sodium from the purification circuit into the inerted purification cabin housing the circuit [6]. The leaked sodium had frozen on the cabin floor and pipelines (fig. 6) and was manually cut and scooped out under inert purging using long tools through the man-hole at the top of the cabin and another man- hole made on one side of the cabin for this purpose. Entry to the cabin was then made wearing air-masks to remove the residual sodium sticking to the pipelines. The cabin was then de-inerted and inspected. Leak was from the body of a 20 mm size bellows-sealed valve, through one of the three blind holes used by the manufacturer for machining the valve body. The valve was replaced and another similar valve rectified by welding tight fitting plugs. The system was normalised in July 2002. Since the problem is generic to the specific make, valves of this genre used in the plant were inspected in stages and rectified wherever under-thickness was seen. The sodium which leaked in the incident was converted to hydroxide, neutralised by orthophosphoric acid and disposed off as active liquid effluent. The man-rem expenditure for this incident was only 2.25 person-mSv. It is gratifying to note that a material thickness of just 0.1 mm was enough to hold sodium for 15 years.

8.3. Reactivity Incidents

There were two positive reactivity incidents- one in Nov. 1994 and one in April 95 [5]. Again, in 1999, during the irradiation of Zr-Nb alloy at 8 MWt, positive reactivity transients of a different nature were seen. The causes for these transients could not be established. They are suspected to be due to core deformations arising out of steep thermal gradients inherent in the small core. With the progressive expansion of the core, they are no longer seen. Dedicated data loggers have been installed.

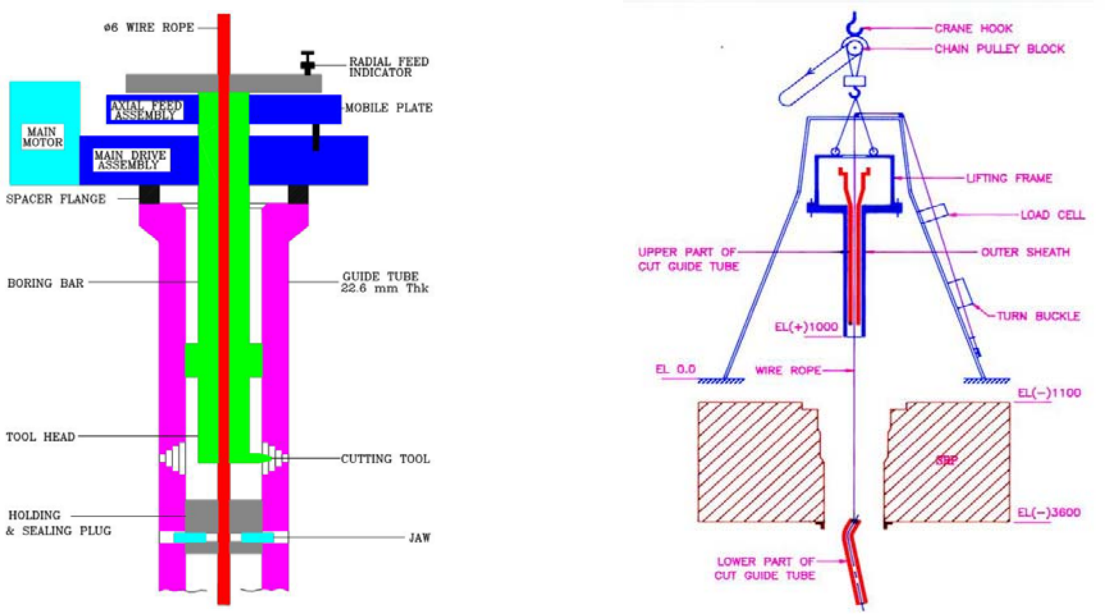

FIG. 5. Guide Tube Cutting and Retrieval

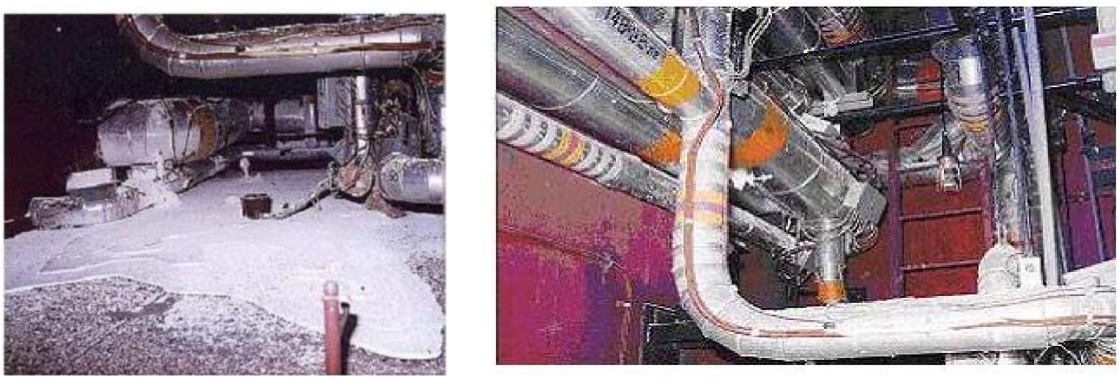

FIG. 6. Leaked Sodium on the Floor and Pipelines

9. Radiological Safety

The total cumulative dose to personnel so far is 78 person-mSv (7.8 man-rem) and the total activity discharged through the stack is 14.229 TBq (459 Ci) of Ar^{41}. During the past 20 years, there has been no significant event of abnormal radioactivity release, personnel or area contamination. These have given us the confidence that fast reactors are eco-friendly and clean sources of energy.

10. Modifications to Improve Reactor Availability

Two major modifications carried out in recent years have resulted in significant improvement in reactor availability. Most of the unscheduled outages were from the steam-water circuit. The contact type reheaters which were used in the steam water system were prone to level fluctuations, resulting in cascade tripping of the feed water pumps and ultimately in reactor trip. The reheaters were replaced by non-contact, shell and tube type design, and the system has now been found to be trouble-free.

The Steam Generator Leak Detection system in each secondary loop was having one detector, and any malfunctioning of the detector would result in spurious trip of the reactor. The system was triplicated in each loop and wired to trip the reactor in 2/3 mode. In addition to improving the plant availability, it also enables differentiation of any genuine leak signal from spurious signal. The triplication also improves system maintainability.

11. Future Plans

Mark-I fuel with a burn-up limit of 155 GWd/t will continue to be the driver fuel for FBTR, and hence the core is likely to continue to be of the current size for quite some time. In order to reach the rated temperatures with the current core, it is planned to effect some modifications in the steam generators. The current major mission of FBTR is to irradiate the PFBR MOX fuel to its target burn-up of 100 GWd/t. As a part of the development of advanced fuels for fast reactors, FBTR will be deployed for testing diverse fuels such as sol-gel based, vibro-compacted oxide fuel and sodium-bonded metallic fuel. For this purpose, assessment of the residual life of the plant is in progress.

12. Conclusion

The experience in construction, commissioning and operation of FBTR has been fruitful for India's fast reactor programme. The inherent flexibility in the design of fast reactor cores has been exploited for the design of varied cores and devlopment and testing of at least three novel fast reactor fuels so far- viz. two compositions of high Pu carbide fuel and one of high Pu MOX fuel. In addition, the MOX fuel of power reactor composition is also being tested in FBTR. The satisfactory performance of the fuels, sodium systems, steam generators and instrumentation during two decades of operation of FBTR has provided sufficient feedback to enable the launch of the 500 MWe PFBR Project. FBTR will also be a test bed for the sodium-bonded metallic fuels, which, by virtue of their better breeding gain, are contemplated for the future power breeder reactors in India.

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