

The Fast Breeder Test Reactor—Design and operating experiences

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Received 11 March 2004; received in revised form 26 September 2005; accepted 28 September 2005

Abstract

The Fast Breeder Test Reactor is a 40 MW t, loop type sodium cooled fast reactor, built at Indira Gandhi Centre for Atomic Research, Kalpakkam, as a fore-runner to the second stage of Indian nuclear power programme. The reactor design is based on the French reactor Rapsodie, with several modifications, which include the provision of a steam–water circuit and turbine-generator in place of sodium–air heat exchanger in Rapsodie. The reactor uses a high Pu monocarbide as the driver fuel, and went critical on 18th October, 1985. Being a unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. The fuel has performed extremely well, with the peak burn-up reaching 136 GW d/t as of 31st December 2004. This paper gives a description of the reactor and summarises the operating history of the reactor. The operating experience of FBTR has provided sufficient feed-back and confidence for India to launch upon the design and construction of a 500 MWe fast reactor.

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1. Introduction

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 Reactor Research Centre (renamed Indira Gandhi Centre for Atomic Research in 1985) was established at Kalpakkam, 80 km south of Madras (now Chennai), in 1971, with the mission to develop the technology of sodium cooled fast reactors. At the heart of the centre was proposed a sodium cooled test reactor, named Fast Breeder Test Reactor, which would serve as a test bed for irradiation of fuels and materials and provide experience in large scale sodium handling and reactor operation. An agreement was signed with CEA (France) for transfer of the design of the Rapsodie reactor, training of personnel in Rapsodie and transfer of manufacturing technology of critical components. The responsibility of construction was totally with India, with the participation of Indian industry (Bhoje et al., 1985).

The Fast Breeder Test Reactor has several design changes vis-à-vis Rapsodie. Major changes include the use of Pu–U monocarbide fuel and incorporation of steam generators and a turbine-generator.

2. Description

The Fast Breeder Test Reactor is a 40 MW t, 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. The major characteristics of the reactor are given in Table 1. Stainless steel (SS 316) is the principal material of construction for the reactor and coolant circuits.

2.1. Reactor core

The core (Fig. 2) consists of 745 closely packed locations, with fuel at the centre, surrounded by nickel reflectors, thoria blankets and steel reflectors. The core is vertical and freestanding, with the subassemblies supported at the bottom by the Grid Plate and held on to the latter by collapsible hold-down springs. The subassemblies are hexagonally shaped. As per the initial design, the core was rated for 40 MW t and had 65 fuel subassemblies, three test steel subassemblies, six control rods, 143 nickel reflectors, 342 thoria blankets and 163 steel reflector subassemblies. In addition, there are 23 storage locations in the outermost steel reflector row.

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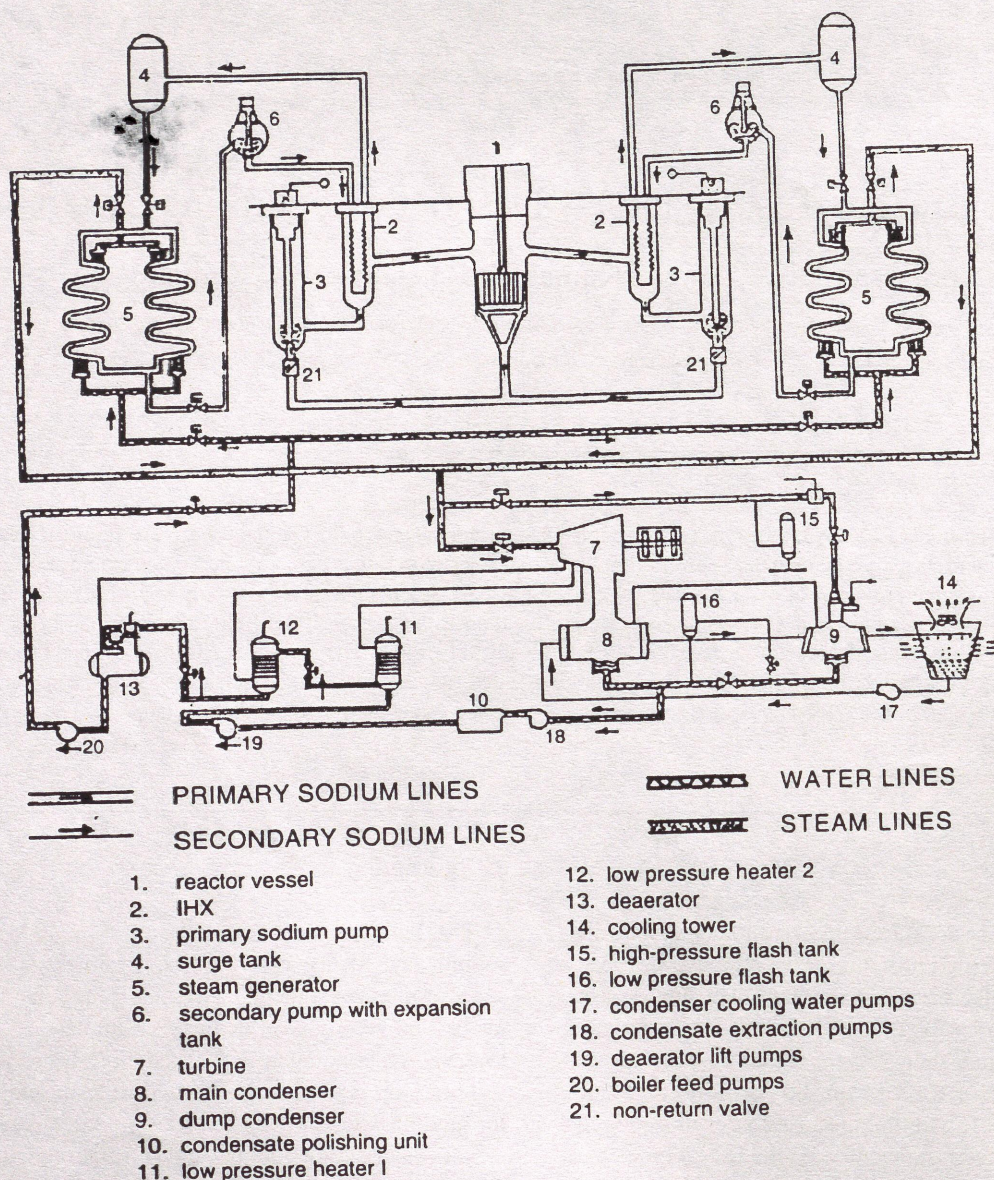


Fig. 1. Schematic flow diagram of the main heat transport system.

The fuel subassembly (Fig. 3) is 1661.5 mm long, with a width across flats of 49.8 mm and houses 61 fuel pins of MOX composition (30% PuO₂ + 70% UO₂ with 85% enrichment). The fuel pins are 531.5 mm long, with fuel stack length of 320 mm. Axial blankets of 263 mm length are provided on either side of the fuel pins.

2.2. Reactor Assembly (Fig. 4)

The Reactor Vessel houses the core and serves as a conduit for the primary sodium coolant flow through the core. Charging and discharging of subassemblies are done from the reactor top with the reactor in shutdown state. The sodium inlet pipe joins the Reactor Vessel at the bottom and two sodium outlet pipes radially branch out of the vessel above the core. The reactor is closed at the top by Large and Small Rotatable Plugs, which serve as top shields and also enable access to the core

locations for fuel handling. The Rotatable Plugs are cooled by nitrogen. Liquid Metal Seals of Tin–Bismuth alloy, backed with Inflatable Seals, isolate the reactor cover gas from the Reactor Containment Building atmosphere. The Liquid Metal Seals are frozen during reactor operation and melted during rotation of the plugs. The space between the Liquid Metal Seals and the Inflatable Seals, called the Interseal Space, is maintained in argon at a pressure higher than the reactor cover gas to prevent the leakage of active cover gas into the reactor building.

The Small Rotatable Plug houses the Control Plug which carries six Control Rod Drive Mechanisms and Core Cover Plate with thermocouples for monitoring the outlet temperatures of the fuel subassemblies. Ten neutron shields, each 25 mm thick, surround the core and minimize the incident flux on the Reactor Vessel. Thermal shields are provided inside the Reactor Vessel to minimize the thermal stresses due to cold and hot shocks.

Table I
Main characteristics of the Fast Breeder Test Reactor

Parameters	Initial design	Present small core
Reactor type		Sodium cooled, loop type
Reactor power	40 MW ν 13.2 MWe	15 MW ν 2 MWe
Fuel	MOX (30% PuO ₂ + 70% UO ₂ with 85% enrichment)	Mark I (70% PuC + 30% UC) Mark II (55% PuC + 45% UC)
No. of fuel subassemblies	65	25 Mark-I + 13 Mark-II
Fuel pin diameter (W/cm)		5.1
No. of pins/subassembly		61
Maximum linear heat rating (W/cm)	400	400
Maximum burn-up (GW d/t)	100	100
Peak neutron flux (n/cm ² /s)	3.5E15	2.2E15
No. of control rods		6
Control rod material		B ₄ C (90% enriched in ¹⁰ B)
Reactor inlet sodium temperature (°C)	380	330
Reactor outlet sodium temperature (°C)	515	425
Primary sodium flow (m ³ /h)	1100	500
Feed water temperature (°C)		190
Steam temperature (°C)	480	405
Feed water flow (t/h)	70	25
Steam pressure (kg/cm ²)		125
Sodium inventory (t)		150
Steam generator		Once-through type, seven tubes in shell, serpentine shaped
Turbine-generator		16 Stages, condensing type, 16.4 MWe air cooled.
Bypass circuit		100% Dump condenser

The radial and axial shifts of the Grid Plate are monitored by two displacement measuring devices.

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

2.3. Sodium systems

Primary sodium is pumped into the reactor by primary sodium pumps and flows by gravity to the intermediate heat exchangers and then back to the pump suction. The intermediate heat exchangers are vertical, counter-flow heat exchangers and transfer heat from the active primary sodium to the inactive secondary sodium. Primary sodium flows on the shell side and secondary sodium on the tube side. The shell is fixed and the tube bundles are removable. Secondary sodium is pumped into the intermediate heat exchangers by secondary sodium pumps. After removing heat from primary sodium, the secondary sodium enters the steam generators. A surge tank is interposed between the intermediate heat exchangers and steam generators as a buffer against pressure wave transmission to intermediate heat exchangers during sodium–water reaction in steam generators due to any water leaks. The four sodium pumps are vertical, single stage centrifugal pumps with axial suction and radial discharge. Each pump has a fixed shell and a removable assembly comprising

the impeller, diffuser and shaft. The shaft is supported by taper roller bearings at the top and guided by hydrostatic bearings at the bottom.

For safety reasons, there are no valves in the primary and secondary sodium main loops. Flow control is by controlling the speeds of the pumps. The pumps are driven by dc motors and powered by Ward Leonard drives. Flywheels mounted on the Ward Leonard drives provide sufficient inertia to run the pumps to ensure that fuel clad hot spot temperature is within limits in the event of power failure. The pump drives are provided with emergency power supply from the station diesel generators and battery backup is provided for the primary pump drives to provide adequate flow for safe removal of decay heat.

All the sodium capacities are provided with an inert cover of argon above the free sodium levels. Argon purity is maintained through NaK bubblers. Both primary and secondary sodium systems are provided with cold traps for sodium purification. Plugging indicators monitor sodium purity.

The steam generator modules are of once-through, counter-flow type, with sodium entering the shell side from top and water entering the tube side from bottom (Fig. 5). The modules have a serpentine configuration, with evaporation and superheating occurring in a single pass. Due to the absence of blow-down, feed water chemistry is maintained within very stringent limits. The steam generator modules are housed inside an insulated casing. By opening the trap doors of the casing, it is possible to remove decay heat from the reactor by natural convection.

The entire primary sodium circuit is provided with a nitrogen-filled envelope called Double Envelope, designed to minimize

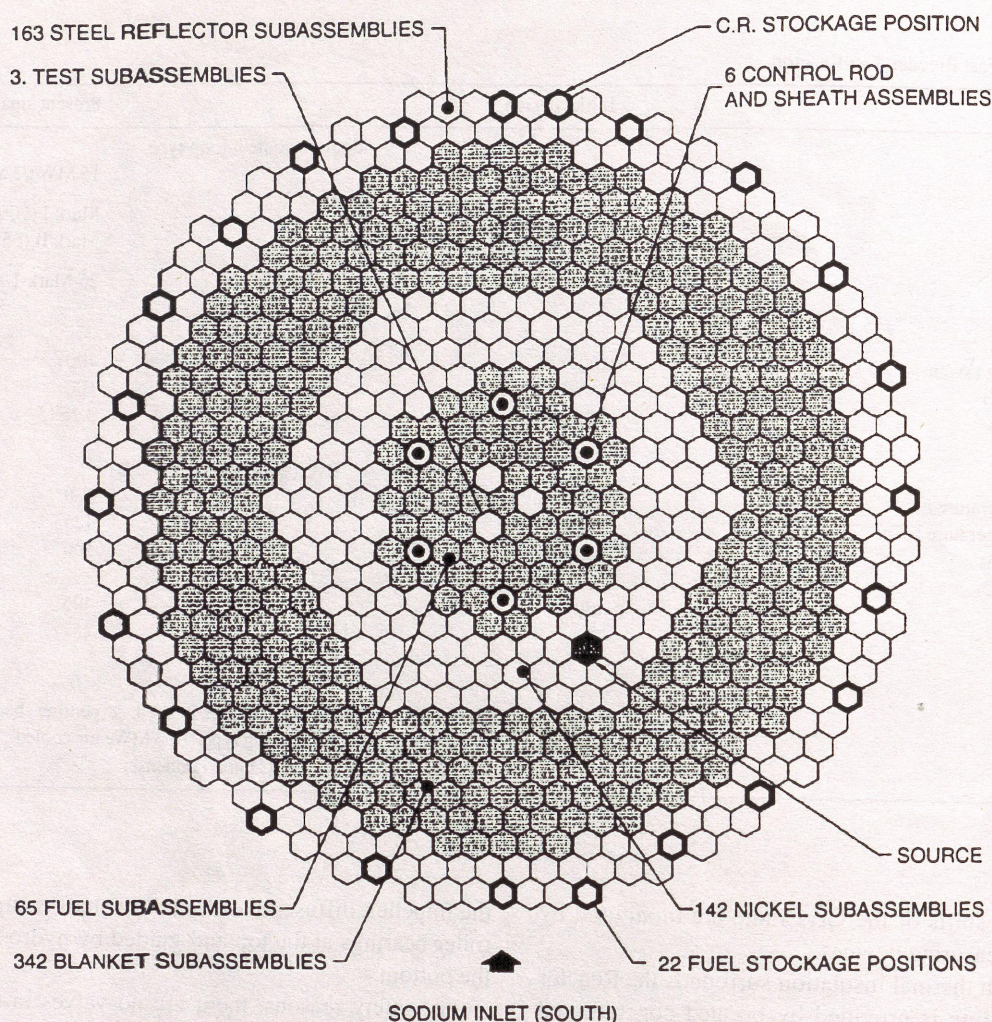


Fig. 2. Reactor core.

the sodium level drop in the reactor in the event of any sodium leak. The annular gap between the Reactor Vessel and its Double Envelope is used for emergency cooling of the core during the very-low-probability incident of simultaneous rupture of coolant boundary and its Double Envelope outside the reactor. Nitrogen is also used for sodium fire fighting in the cells housing the primary sodium system in the unlikely event of failure of the main coolant boundary and its Double Envelope.

2.4. Generating plant

The steam–water circuit consists of all the equipment in a conventional power plant. An on-line condensate polishing unit meets the stringent feed water chemistry requirements of the once-through steam generators. A cooling tower cooled by induced draft fans serves as terminal heat sink. The turbine is a single cylinder, 16 stage, non-reheat condensing turbine and is designed to produce 16.4 MWe with 72.5 t/h flow of superheated steam at 120 kg/cm² and 480 °C. The generator is rated for 19.3 MVA, 6.6 kV, 3 ϕ , 50 Hz, 0.85 pf, 3000 rpm. The rotor is air-cooled with a closed circuit. The generator

field is powered by a shaft driven exciter rated for 110 kW and 220 V (dc).

2.5. Instrumentation and control

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

A Central Data Processing System comprising two computers (one on-line and the other on auto-standby) monitors the core outlet temperatures and generates reactor trip on mean core outlet temperature, mean core temperature rise and plugging of any fuel subassembly. The Central Data Processing System also initiates lowering of rods on control rod level discordance and

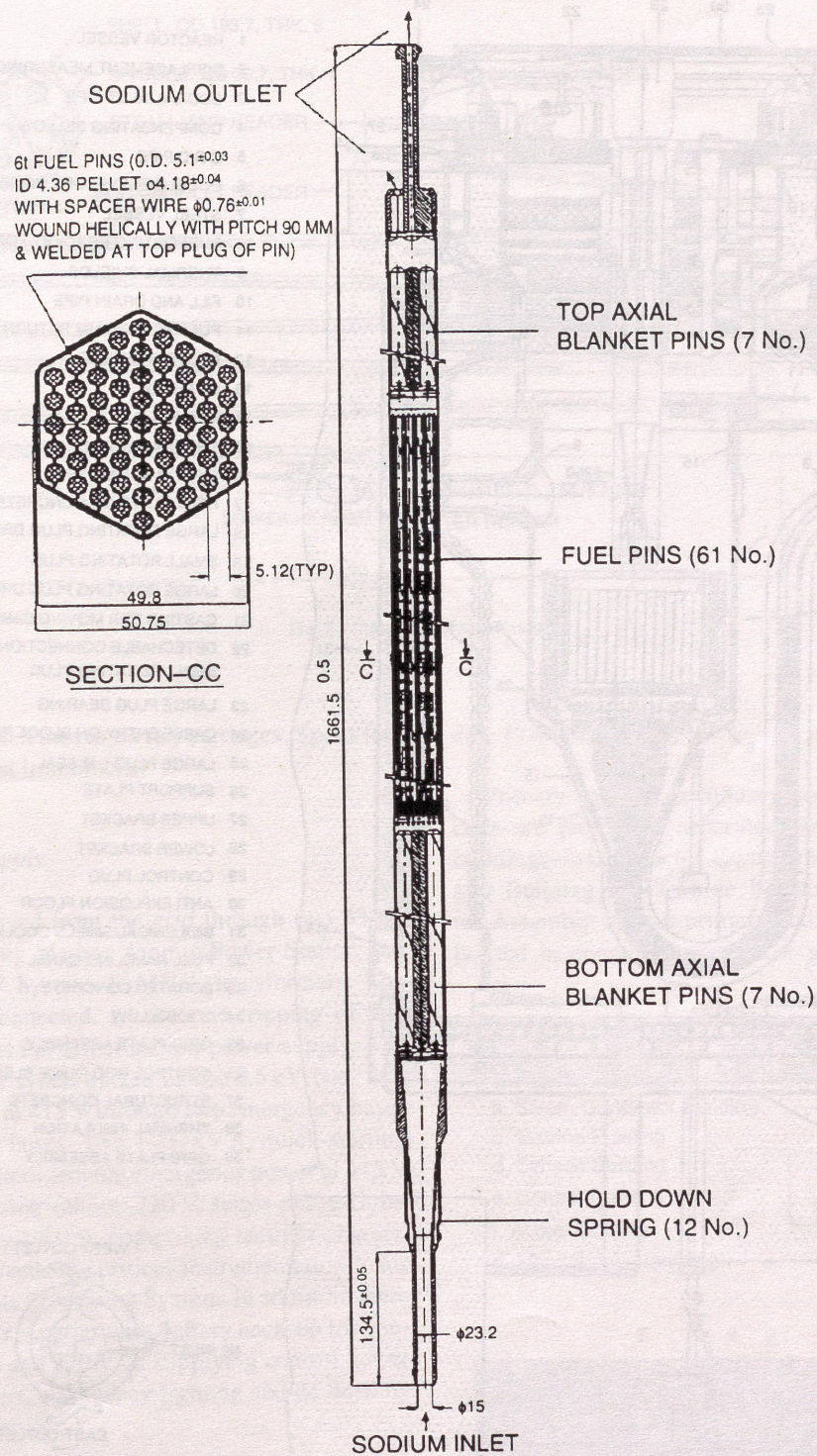


Fig. 3. Fuel subassembly.

rate of rise of Steam Generator Leak Detection System signal. In addition, it checks the healthiness of the reactor protection system by carrying out a Fine Impulse Test and supervises the analog and digital signals.

Sodium levels in the capacities are monitored by continuous and discontinuous level probes. Flows are monitored by permanent magnet type electro-magnetic flow meters. Sodium leak

is detected by spark plugs, wire type and ionization types of detectors.

Water leak into sodium in the steam generator at the incipient stage (50 mg to 10 g/s) is detected by a very sensitive Mass Spectrometer based Steam Generator Leak Detection System to measure hydrogen in sodium in ppb levels. At lower sodium temperatures, as during valving in water into the steam

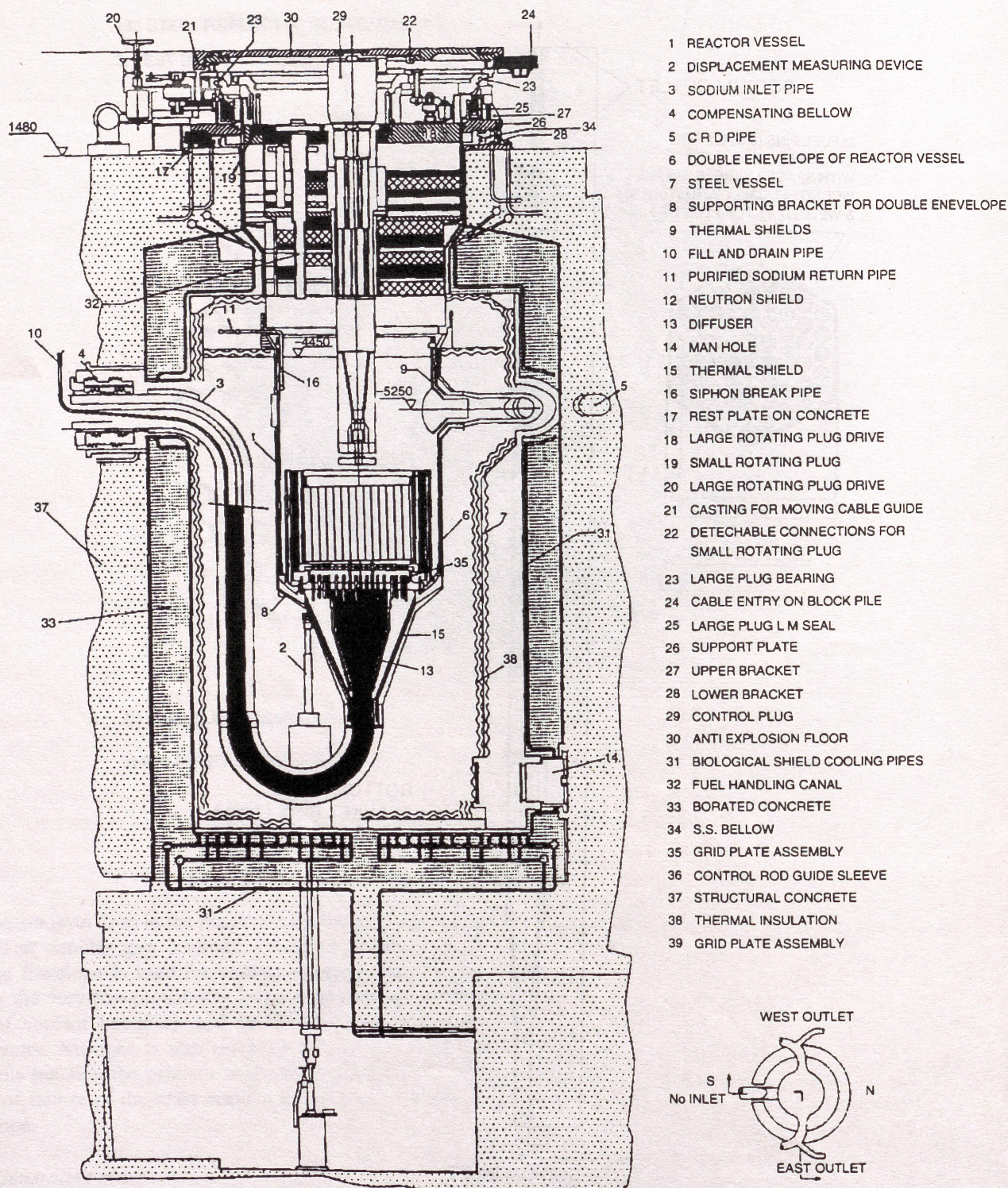


Fig. 4. Reactor Assembly.

generators during power raising, leak is detected by monitoring the hydrogen in the cover gas of expansion tanks. Medium leaks (0.01–0.5 kg/s) are detected by monitoring the expansion tank cover gas pressure. Quick closing valves isolate the steam generators on the sodium and water sides in the event of large leaks (>0.5 kg/s) and rupture discs provided in the sodium headers relieve pressure build up.

2.6. Fuel handling

Fuel handling is done off-line using Charging and Discharging Flasks. There are special flasks for handling of control rod drives, irradiation devices, pumps and intermediate heat exchangers. For moving the active components between Reactor Containment Building and the adjacent Active Building, there

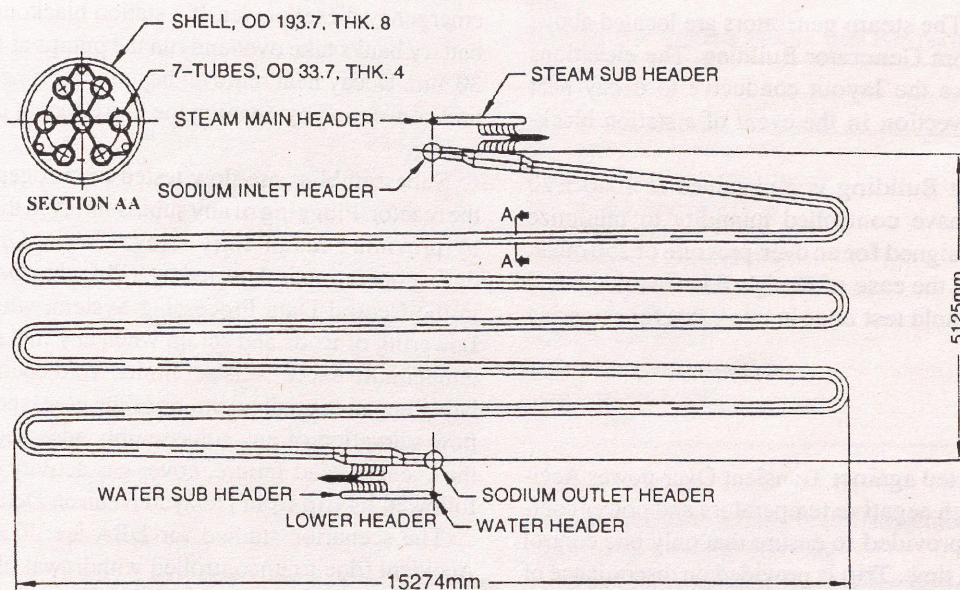


Fig. 5. Steam generator module.

is a Secondary Flask, which moves on rail carriages. Spent fuel is stored in air-cooled cast iron blocks.

2.7. Electrical power supply

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

2.8. Auxiliary systems

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

2.9. Plant layout (Fig. 6)

Primary sodium, secondary sodium and steam–water circuits are physically separated by locating them in separate buildings—viz. Reactor Containment Building, Steam Generator Building and Turbine Building, respectively. The Reactor Assembly and all primary sodium system components are housed in concrete cells below ground level in the Reactor

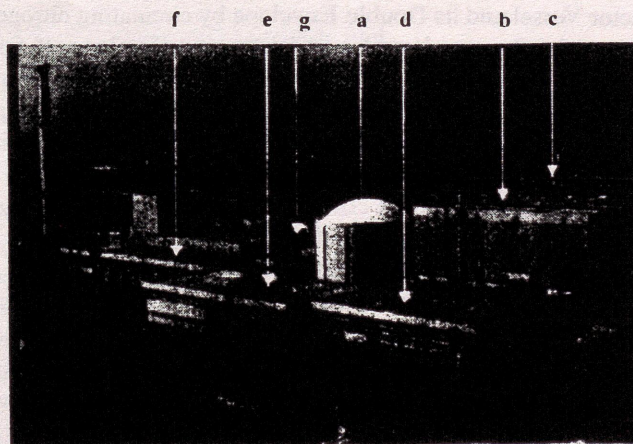
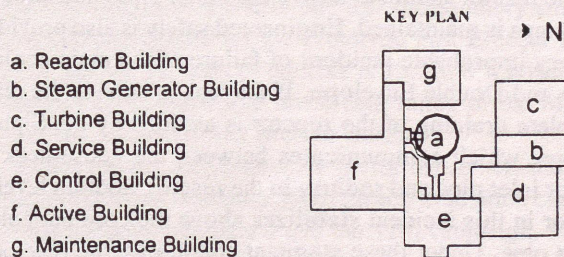


Fig. 6. Aerial view of FBTR with key plan.

Containment Building. The steam generators are located above ground level in the Steam Generator Building. The elevations of the components make the layout conducive to decay heat removal by natural convection in the event of a station blackout.

Reactor Containment Building is maintained at 1040 ± 20 mbar (a), in order to have controlled humidity to minimize sodium fire risks. It is designed for an over-pressure of 250 mbar, to contain the activity in the case of Design Basis Accident and is subjected to pressure hold test once in two years for assessing the leak rate.

3. Safety

The reactor is protected against Transient Over-power Accident by feedback through negative temperature and power coefficients. Interlocks are provided to ensure that only one control rod can be operated at a time. Trip is provided on discordance of control rod levels in the event of any uncontrolled withdrawal. The reactor is provided with redundant and diverse trips from neutronic and core thermal parameters, i.e., $\ln-P$, $\log-P$, period, reactivity, mean core outlet temperature and mean core temperature rise. Insertion of two control rods is sufficient to shut down the reactor from the most reactive configuration.

The Double Envelope of the primary sodium loops safeguards against Loss of Coolant Accident. Leak-before-break enables early detection before any leak escalation. Spark plugs at the bottom of the primary sodium capacities and ^{24}Na particulate activity monitors in the nitrogen in Double Envelope enable detection of leak at the incipient stage. Level probes in capacities confirm escalated leaks. In the event of any leak, sodium level in the reactor stabilises above the outlet pipes and circulation by pumps is maintained. Engineered safety is also provided for the very improbable incident of failure of the main coolant boundary and Double Envelope. If the leak is outside the reactor, complete draining of the reactor is avoided by the Siphon Break Pipe, which communicates between the goose-neck of the reactor inlet pipe and sodium in the reactor. Sodium level in the reactor in this incident stabilizes above the core but below the outlet pipe. Under these stagnant conditions, the reactor is cooled by circulating nitrogen in the annular space between the Reactor Vessel and its Double Envelope by circulating nitrogen between the Reactor Vessel and its Double Envelope. If failure of the main coolant boundary and Double Envelope occurs inside the Reactor Assembly, the leaked sodium collects in the steel vessel. There is a provision to flood the Reactor Vessel with sufficient sodium from a dedicated sodium flooding system to ensure submergence of the core. Biological shield cooling system removes heat from the steel vessel.

Absence of any valve in the loops and operation of two pumps in parallel safeguard against Loss of Flow Accident. Trips are provided from total flow through the reactor, individual loop flow, individual pump trip, core outlet temperature and core temperature rise. During any offsite power failure, the reactor trips and primary pumps coast down due to the inertia of the flywheels, providing at least 30% of the rated flow, 40 s after power failure. The pumps are meanwhile taken over by the

emergency diesel power. If a station blackout occurs, dedicated battery banks take over and run the pumps at 150 rpm for at least 30 min. Decay heat removal beyond this time is possible by natural convection by opening the trap doors of the steam generator casing.

Subassemblies are flow tested in nitrogen before loading in the reactor. Plugging of any subassembly in the reactor is avoided by providing radial entry sleeves in the Grid Plate. Plugging during operation is detected by a Plugging Detection Subroutine in the Central Data Processing System, which initiates alarm, Lowering of Rods and scram when any fuel subassembly outlet temperature exceeds its set limits. Void coefficient of reactivity being negative at all positions of the core, sodium boiling due to flow starvation of any subassembly adds negative reactivity. In the event of clad failure, cover gas activity alarm annunciates, followed by trip from Delayed Neutron Detection System.

The scenarios studied for DBA are: Transient Over-power Accident (due to uncontrolled withdrawal of control rod), Loss of Coolant Accident and Loss of Flow Accident, without protection, i.e., without the actuation of the shut down system. The resulting scenario under any of these conditions is partial core melting, rapid rise of power due to additional reactivity caused by slumping of core, fuel vaporisation and core disassembly. Theoretical calculations indicate the maximum fuel fraction that will vaporise as 6.5% and the thermal energy generated before the core becomes subcritical by disassembly as 272 MJ. With conservative allowances for fuel-coolant interaction, the mechanical energy release due to instantaneous isentropic expansion of the fuel vapour has been calculated as 14.5 MJ or less than 3.5 kg of TNT. Reactor Vessel can safely contain 8.8 kg of TNT. The energy released through the top will be absorbed by the tie rods of the Anti-Explosion Floor, to limit the lifting of the rotating plugs to 800 mm. The quantity of sodium that will be ejected has been calculated as 27 kg. The Reactor Containment Building has been designed for an over-pressure of 250 mbar, resulting from the combustion of 100 kg of sodium, and will get isolated by signals from pressure and activity monitors. There is provision for controlled release of activity through the stack.

4. Construction and commissioning (Bhoje et al., 1991; Kapoor et al., 1994, 1996; Ramalingam, 1999; Suresh Kumar et al., 2002a)

Construction 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. The reactor was made critical on 18th October 1985 and low power physics experiments (Lee et al., 1990) were conducted. Steam generator modules were then connected to the secondary sodium circuits.

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 the plug rotation logic during fuel handling, remaining in bypassed state, resulting in

the Rotatable Plugs being rotated with the foot of a fuel sub-assembly protruding into the core during an in-reactor 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. In a complex mechanical interaction that took place during subsequent plug rotation, the sturdy Guide Tube which guides the gripper of the Charging Flask 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. 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.

Reactor was operated up to a maximum power of 1 MW t till 1992 for intermediate power physics and engineering experiments. The steam–water circuit was commissioned, steam generators were put in water service and power was raised to 8 MW t in January 1993. After completing high power physics and engineering experiments, power was raised to 10.2 MW t, generating superheated steam suitable for admission to the turbine-generator. After completing the commissioning activities of TG and auxiliaries, power was raised to 11.5 MW t and the turbine-generator was synchronised to the grid in July 1997.

5. Reactor operation summary (Kapoor et al., 1994, 1996; Ramalingam, 1999; Suresh Kumar et al., 2002a; Chetal, 2003)

Rapsodie used MOX fuel with 30% PuO₂ and 70% UO₂ (with the latter enriched to 85%) as the driver fuel. This design was originally proposed for the core of the Fast Breeder Test Reactor. Since Indian nuclear programme does not envisage the use of enriched uranium at any stage, investment in a high enrichment facility for the sole purpose of fuelling the Fast Breeder Test Reactor core was considered neither expedient nor viable. Hence, the option of using an alternate fuel rich in Pu was studied. Compatibility of higher concentrations of PuO₂ with sodium and difficulties in fuel fabrication, especially with respect to maintaining the stoichiometry (oxygen/metal ratio) and uniformity of phases, ruled out the choice of oxide fuel. Hence, the carbide option was studied and found to be feasible, notwithstanding the extra need for inert atmosphere during fabrication (Ganguly et al., 1986). Being a unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. Hence, the core was redesigned as a small carbide core. As against the original design of 65 MOX fuel subassemblies rated for 40 MW t, the small carbide core had 22 fuel subassemblies with 70% PuC and 30% UC composition (designated as Mark-I fuel) during first criticality. This small carbide core was rated for 10.2 MW t, with the peak linear heat rating limited to 250 W/cm. The core has since been progres-

Table 2
Summary of performance statistics from 1985 (up to 31st December 2004)

Parameter	Cumulative value since first criticality
Maximum power (MW t/MWe)	17.4/2.8
Maximum linear heat rating (W/cm)	400
Bulk sodium temperature (°C)	420
Operating time (h)	34,520
Thermal energy produced (MWh)	230,000
Electrical energy generated (MWh)	4260
TG synchronisation time (h)	4400
Peak burn-up (GW d/t)	136
Longest operating campaign (d)	54
No. of lowering of rods	235
No. of scrams	155

sively enlarged by adding fuel subassemblies to compensate for reactivity loss due to burn-up.

With a view to raise the reactor power to 40 MW t, it was decided, in 1995, to go in for a full carbide core of 78 fuel subassemblies. The fuel composition chosen was 55% PuC + 45% UC (designated as Mark-II fuel). Fuel subassemblies of Mark-II composition were inducted in 1996. A gradual transition to the full carbide core was envisaged. The Mark-I fuel in the centre was retained to continue the irradiation for assessing its ultimate burn-up capability before phasing it out. Mark-II fuel was added at the periphery. The allowable peak linear heat rating of the Mark-I fuel has also been revised up to 400 W/cm based on the fuel performance. The reactor power for the small carbide core has been progressively increased, reaching the highest power of 17.4 MW t in 2002. The present core has 28 Mark-I (70% PuC + 30% UC) and 13 Mark-II (55% PuC + 45% UC) subassemblies, in addition to one test subassembly simulating the MOX fuel chosen for the Prototype Fast Breeder Reactor. The present power limit of the core is 15.7 MW t, arising out of limitation in the allowable linear heat rating of the MOX test fuel of the Prototype Fast Breeder Reactor.

Table 2 gives the cumulative performance statistics of the reactor as of 31st December 2004. Eleven irradiation campaigns have so far been completed and the 12th campaign, started in October 2004, is in progress.

6. Performance of the carbide fuel

The performance of the Mark-I carbide fuel has far exceeded our initial expectations. Being a unique fuel, the initial limits on linear heat rating and burn-up were set at modest 250 W/cm and 25 GW d/t, respectively. Subsequently, out-of-pile tests simulating various values of linear heat rating confirmed that a definite scope exists for raising the allowable linear heat rating to 320 W/cm. The reactor was operated at this peak linear heat rating from 1994 till 2001.

Post-irradiation examination of a subassembly discharged at 25 GW d/t burn-up in 1996 (Baldev et al., 1997) indicated the fuel-clad gap to be intact (Fig. 7); there was no significant fission gas release into the plenum. The target burn-up was hence revised to 64 GW d/t.

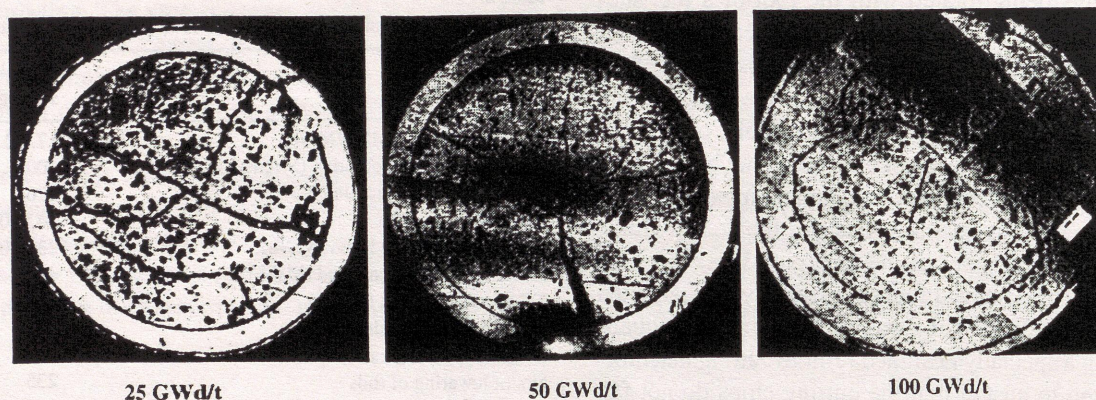


Fig. 7. Photomicrographs of Mark-I fuel (70%PuC + 30%UC) at different burn-up values.

Another subassembly was discharged at 50 GW d/t burn-up in 1999. Post-irradiation examination results indicated that the fuel–clad gap had closed, but there were sufficient voids in the fuel to accommodate further fuel swelling before start of significant Fuel–Clad Mechanical Interaction (Fig. 7). About 20% of the fission gases was found released. The results gave the scope for raising the burn-up limit to 94 GW d/t. The improved conductance due to closure of the fuel–clad gap, vis-à-vis the reduction in the bond gas (He) conductivity due to dilution by fission gases, indicated that the linear heat rating could also be raised to 400 W/cm beyond a burn-up of 35 GW d/t. The 10th irradiation campaign in 2002 was with this peak linear heat rating, leading to increase in reactor power to 17.4 MW t. Five fuel subassemblies crossed a burn-up of 100 GW d/t in September 2002 without failure. One subassembly with a burn-up of 101.5 GW d/t was discharged for post-irradiation examination at the end of this campaign. This location was loaded with a fresh subassembly. Peak linear heat rating of the core was hence reverted to 320 W/cm for the Mark-I fuel.

Post-irradiation examination of the Mark-I fuel subassembly discharged at the burn-up of 101.5 MW d/t was completed in March 2004. Metallographic examinations carried out on fuel pins indicated that the fuel–clad gap has completely closed at the centre of the fuel column. Most of the voids in the fuel had also been exhausted. No significant clad carburisation was observed where the fuel had come in contact with the clad. The percentages of fission gas released to the plenum was 14%. The fuel had an average volumetric swelling of 5.2%. Based on these results, safety clearance has been obtained to operate the reactor till the first clad failure or till the Mark-I fuel reaches a burn-up of 150 GW d/t, whichever is earlier. As of 31st December 2004, the Mark-I fuel has seen a peak burn-up of 136 GW d/t without any clad failure. A system to identify the age of the failed fuel by measuring the $^{85}\text{Kr}/^{88}\text{K}$ ratio in cover gas has been installed in the cover gas rejection circuit.

7. Irradiation programme

The Fast Breeder Test Reactor has been used as a self-driven irradiation facility for the Pu rich monocarbide fuel. Seven pins of Mark-I and Mark-II compositions were irradiated during 1995–1997 to study their initial restructuring behaviour.

With its flux one order higher than in the Pressurised Heavy Water Reactors, the Fast Breeder Test Reactor was used for studying the irradiation creep behaviour of Zr–Nb being used in the Indian Pressurised Heavy Water Reactors. Pressurised Zr–Nb capsules were irradiated during 1998–2000, simulating their operating stresses and temperatures. For this purpose, reactor was operated at 8 MW t, with inlet temperature lowered to 308 °C. The creep behaviour was found to be in accordance with published data.

The present mission of the Fast Breeder Test Reactor is to irradiate the MOX fuel (29% PuO_2) chosen for the Prototype Fast Breeder Reactor to the target burn-up of 100 GW d/t at the design linear heat rating of 450 W/cm. To get the required linear heat rating of 450 W/cm in the high Pu core of the Fast Breeder test Reactor, ^{233}U has been added to the test fuel. Since ^{233}U will be used in the third stage of India's nuclear energy programme, the irradiation is expected to provide some insight into its behaviour as well.

8. Physics and safety related engineering experiments (Suresh Kumar et al., 2002b)

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.

The void coefficients were measured in the third, fourth and fifth rings by loading special helium filled and sodium vented subassemblies and found to be negative. Using these values, the void coefficients were computed for the central, first, second and third ring subassemblies, modelling the exact geometry of fuel. The void coefficients for these locations are, respectively, found to be -77.0 ± 51.7 , -65.33 ± 18.8 , -32 ± 18.3 and -48.7 ± 18.3 pcm, confirming that the void coefficient is negative at all locations in the core.

Delayed Neutron Detection System count rates were measured using a specially designed subassembly with perforated pins with natural uranium, at low power levels (up to 1.8 MW t). The results indicate that the recoil area for the Delayed Neutron Detection signal to reach scram threshold at the test conditions

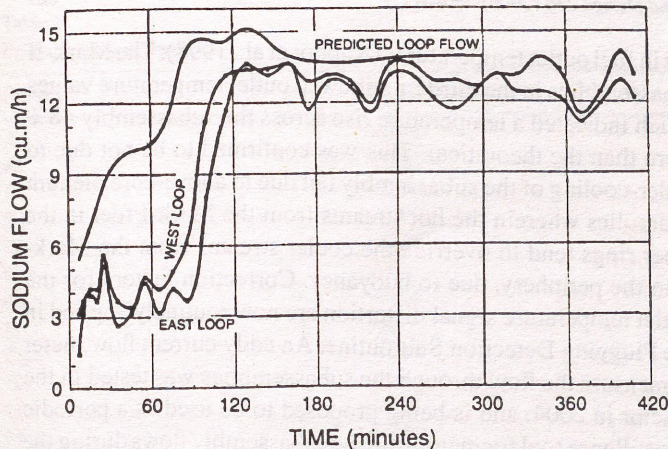


Fig. 8. Natural convection flow in secondary system.

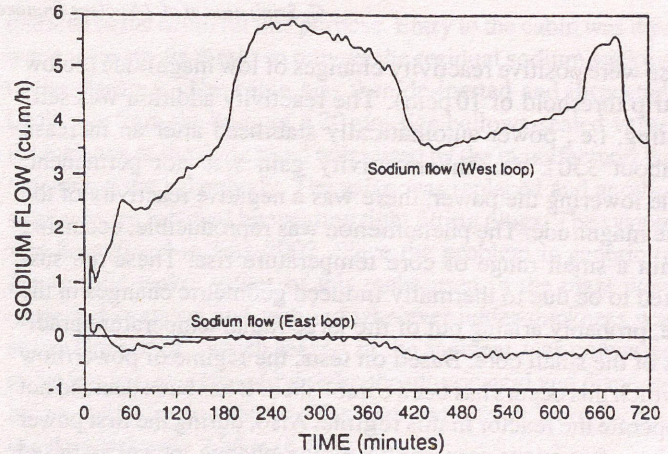


Fig. 9. Natural convection flow in primary sodium system.

is 12 cm^2 . The signal was found to increase with inlet sodium temperature and primary sodium flow. Based on these experimental results, the Delayed Neutron Detection signal level for scram for the 15 MW t conditions (350°C inlet and $500 \text{ m}^3/\text{h}$ flow through the reactor) corresponds roughly to about 4 cm^2 recoil area.

For validation of computational methods used in fast reactor shield design, reaction rates were measured at 25 kW t, 400 kW t and 10 MW t in the sodium region above the core and in the rotating plug using special irradiation devices. The measured reaction rate profiles agree very well with the predicted reaction rate profiles in sodium above core. The measured values in the rotating plug region are being used to fine-tune the codes.

A series of safety related engineering tests was conducted in 1994–1995, basically to validate the codes used in incident analysis (Vaidyanathan et al., 1994). Normal plant incidents like tripping of one pump in the primary, secondary or tertiary loops and offsite power failure were studied and the sequence of events confirmed to be as per safety logic. Temperature transients seen by the components during these incidents were recorded. 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 intention.

As a precursor to the station black out test, natural convection tests in the secondary and primary loops were separately carried out. Reactor power was stabilised at 170 kW t before tripping both the pumps in either secondary or primary. Control rods were not operated and reactor power was allowed to evolve due to the feedback coefficients. In the secondary loops, natural convection flow was established after some delay (Fig. 8), attributed to initial stagnation of hot sodium in the capacities, i.e., surge tanks and expansion tanks. In the primary circuit, sufficient flow for decay heat removal got established in one loop after some delay, while the sodium was stagnant in the other loop and started cooling off due to surface heat losses (Fig. 9). In both the tests, power fluctuations due to inlet temperature feedback of reactivity were observed. The test results were used for improving the codes used for prediction of incident scenario in the safety analysis of the reactor. Further tests including station blackout test are planned.

The evolution of the temperatures of biological shield concrete and structural concrete of the Reactor Assembly in the absence of shield cooling was studied. While the bulk concrete temperature was within limits (80°C) up to a period of 6 h, temperatures in the air-cooled structural concrete penetrations of inlet and outlet pipes were crossing the thresholds in about 3 h. Since the design codes permit higher local temperatures ($\sim 120^\circ\text{C}$) for concrete, regulatory authorities have been approached for relaxation of the limits on concrete temperature.

9. Performance of the systems (Kapoor et al., 1994, 1996; Ramalingam, 1999; Suresh Kumar et al., 2002a)

9.1. Reactor core

The isothermal temperature and power coefficients of reactivity, measured for each core configuration, have been found to be negative. Average values for the present 41 fuel subassembly core are: $-4.3 \text{ pcm}/^\circ\text{C}$ and $-15.37 \text{ pcm}/\text{MW t}$, respectively. The power coefficient, though negative, has been found to show a dip, i.e., less negative values, in the intermediate power ranges (5–9 MW t); this is being analysed. The loss of reactivity due to fuel burn-up has been measured as $1.24 \text{ pcm}/\text{MW d}$.

In November 1994, there was an incident of anomalous positive reactivity, which was controlled below trip limit by the operator. A series of tests to study 15 postulated causes like operator error, changes in process parameters and ingress of moderator, was conducted, but the reason could not be found out. Reactor operation was resumed on an experimental basis. In April 1995, the reactor underwent scram on positive reactivity during power raising. Investigations did not reveal the cause. Reactor operation was again resumed, on experimental basis, with dedicated fast data acquisition systems to record the various parameters during a transient, if it recurs. The reactivity trip threshold was also raised from 10 to 30 pcm, to enable data acquisition for a longer time, but the incident did not recur. The trip setting was then reverted to 10 pcm. In both these incidents, there was a small permanent gain of reactivity by the core.

During the irradiation of Zr–Nb capsules at 8 MW t in 1999–2000, reactivity anomalies of a different nature occurred.

These were positive reactivity changes of low magnitude (below the trip threshold of 10 pcm). The reactivity addition was self-limiting, i.e., power automatically stabilised after an increase of about 350 kWt. The reactivity gain was not permanent; while lowering the power, there was a negative reactivity of the same magnitude. The phenomenon was reproducible, occurring within a small range of core temperature rise. These are suspected to be due to thermally induced geometric changes in the core, probably arising out of the large radial temperature gradients of the small core. Based on tests, the regime of power/flow in which this occurs has been established. It has been decided not to operate the reactor in this regime. Also, during the first power raising after every core configuration change, power is raised close to the over-power trip threshold of 110% to check for the absence of this phenomenon before resuming regular operation.

In May 2003, one control rod was found to be repeatedly dropping during power raising. The control rod and its outer sheath were discharged for post-irradiation examination. Profiling of the geometry of the control rod and outer sheath and dimensional measurement of the foot of the control rod for swelling are in progress.

9.2. Reactor Assembly

During commissioning, Reactor Vessel deflection was found high, beyond a sodium temperature of 270 °C, due to circumferential variation of the Reactor Vessel temperatures in the cover gas region. It was suspected to arise due to convection loops formed by the cover gas in the annular gap between Large Rotatable Plug and Reactor Vessel. This was solved by injecting helium in the gap, while admitting argon cover over the free sodium level below. The resulting density gradient was able to suppress the convection loops in the annulus and minimise the Reactor Vessel deflection (Srinivasan et al., 1988).

During initial power raising, the Reactor Vessel temperatures at the main flange supporting the Grid Plate were found closer to the reactor outlet temperatures, and verified to be due to recirculation of sodium from the top plenum. Extensive 3-D analysis and water model studies for this unexpected phenomenon revealed that the recirculation is due to the downward deflection of core outlet streams after impingement with the Reactor Vessel. Stress analysis for the actual temperature profiles indicates that stresses are within limits.

The Inflatable Seals failed during commissioning. Hence, the interseal argon pressure could not be maintained. Whenever the Liquid Metal Seals are melted, the Interseal Space is periodically flushed with argon to prevent oxidation of the Liquid Metal Seal. Visual inspection of the Liquid Metal Seals indicates that the seal has not degraded by oxidation on the reactor side. The silicone oil kept above the alloy on the interseal argon side has been found to get solidified into a gel. The oil is topped up periodically. Both the Inflatable Seals were replaced in 2004. However, it has not been possible to maintain the interseal argon pressure due to leaks from some elastomer seals (O-rings) which are difficult to replace.

In 1996, the Core Cover Plate, housing the core thermocouples, got stuck in the fuel handling position, resulting in anom-

lies in fuel outlet temperatures (Kapoor et al., 1998). The Mark-II subassemblies in the third ring showed outlet temperature values which indicated a temperature rise across the subassembly 48% more than the theoretical. This was confirmed to be not due to under-cooling of the subassembly but due to above-core plenum hydraulics wherein the hot streams from the Mark-I fuel in the inner rings tend to override the cooler streams from the Mark-II in the periphery, due to buoyancy. Correction factors for the outlet temperature signal distortion are now routinely applied in the Plugging Detection Subroutine. An eddy current flow meter to measure the flow through the subassemblies was tested in the reactor in 2004, and is being proposed to be used as a periodic surveillance tool for measuring the subassembly flows during the fuel handling campaigns. The Central Canal Plug housing the thermocouples of the central subassembly was replaced in 1997, since two out of its three thermocouples started giving erratic readings. This was again replaced by a longer plug in 2002, to provide accurate measurement of the central subassembly in the present stuck condition of the Core Cover Plate.

Periodic visual inspection of the Reactor Vessel internals using a periscope has confirmed no abnormalities. There is no significant sodium/oxide deposition on the Reactor Vessel or at the bottom of the Rotatable Plugs. By running one primary pump with the reactor partially drained, flow through the Siphon Break Pipe is visually seen, confirming the non-plugging of the pipe.

In 2000, water leaks were seen from the biological shield cooling coils. The leaks were chemically sealed. To prevent stagnation of any leaky water in the annular gap between the borated biological shield concrete and structural concrete resulting in external corrosion of the pipes, drain holes have been drilled through the concrete. The annular gap is also periodically flushed with instrument air.

Failure of a few cores in the Trailing Cable system on the reactor has been occurring during plug rotation. These cables wrap around the Rotatable Plugs and are held taut by counterweights through a system of pulleys. The failure is due to the stresses arising out of this complex system. Cables are being periodically replaced.

9.3. Sodium systems

The four main sodium loops have been in continuous operation since 1985. Sodium purity has been maintained well, with the plugging temperature always below 105 °C. On two occasions, plugging temperature of secondary sodium went up. One was during connection of the steam generators to the sodium header of the loop and the other was during intervention in the system for welding the on-line carbon meter. Plugging temperature also goes up during calibration of Steam Generator Leak Detection System, when controlled quantities of H₂ are injected in sodium. Required purity could be restored in a few days on all these occasions using the cold traps. Primary cold traps were out of service for about four months during a sodium leak incident, but plugging run after initial refilling of the purification circuit indicated plugging temperature less than 105 °C, testifying to the integrity of the cover gas system and efficacy of the NaK bubblers.

Table 3
Primary sodium analysis

Element	Value (ppm)
Ag	<0.02
Al	0.21
Ba	<0.02
Be	<0.02
Bi	0.05
C	31.6
Co	<0.02
Cr	<0.3
Cu	0.03
Fe	0.8
Li	<0.02
Mn	0.02
Mo	0.06
Ni	0.1
Pb	0.14
Rb	<0.02
Sb	<0.02
Sn	<0.02
U	<0.02

The hot argon lines feeding cover gas capacities have been periodically tested and found to be free from plugging by sodium aerosols or oxides. The sodium flooding provision for the reactor is also periodically checked by injecting controlled quantities of sodium and visually seeing the flow through the flooding pipe in the reactor using the periscope. These checks have confirmed that the flooding lines are free from plugging.

Initially, a sudden reduction in flows through secondary cold traps was observed. The cause was traced to restricted area at the region of cold sodium entry into the central uptake pipe. Cold traps were removed, decontaminated, modified and put back in service.

There was an incident of mercury ingress from the cover gas pressure relief pot into primary sodium. Remedial measures have been taken to avoid recurrence. Periodic primary sodium samples have been showing mercury activity due to this incident. Periodic chemical analysis of primary and secondary samples show that the nuclear grade purity is well maintained (Table 3). Radiometric survey was carried out on the primary capacities and pipelines in October 2002. Sodium was drained from all capacities except the Reactor Vessel. The survey indicated activated corrosion product deposition, comprising mostly ^{54}Mn , ^{58}Co , ^{60}Co and ^{65}Zn . There were two minor incidents of NaK leak, one due to failure of a spark plug in a secondary cold trap and the other due to backing up of NaK from the NaK bubblers into the service argon system. Suitable design changes have prevented their recurrence. There were three incidents of minor sodium leaks from the secondary—one from the ferrule joint of sodium sampler and two from failure of bellows in valves.

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 (Chetal, 2003). The leaked sodium had frozen on the cabin floor 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 make 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 man-mSv.

The four sodium pumps have been operating satisfactorily, with a cumulative operating time of 525,000 h. One secondary pump was replaced in 1987 after 10,000 h of operation, consequent on metallic ringing noise heard near the discharge. The removed pump was decontaminated, dismantled and inspected. The source of noise is suspected to be loose fit between the hole in the shaft and the plug fitted therein to admit high pressure sodium to the hydrostatic bearing.

Oil seals in the pumps have been performing well. Leaked oil is being recovered periodically. The total inventory of unaccounted oil is less than 1000 ml so far. Seals were replaced for both the primary and secondary pumps when the oil leak rates went up.

During commissioning, stems of the motorised valves were found breaking before actuation of the torque limiter. For critical applications like sodium flooding system, the valves were replaced by pneumatic valves. For others, the valve is closed up to the travel limit by the motor and then manually shut tight.

The steam generator modules have been in water service for 18,000 h and have seen 140 cycles of depressurisation. There has been no water leak into sodium. During initial power raising, water leaks into the steam generator cabin occurred due to defects in the caps housing the orifices of the tubes. The caps were replaced. There were generic water leaks past the flanged joints of the orifice flow meter assemblies in the water sub-headers. These were arrested by changeover to welded assemblies. All these leaks were annunciated by the spark plugs kept in trays below the modules for detecting sodium leaks. The steam generator cabin has now been provided with a moisture detector. There was another incident of water leak through a stub for housing the temperature sensor at the steam outlet tubes. The stub was blanked.

Wire type leak detectors have been installed on the steam generator modules to detect sodium leaks into the cabin. A dedicated argon bank to inert the cabin in the event of any sodium leak from the steam generator shell has been provided.

9.4. Generating plant

The steam–water circuit is in operation since 1993. The feed water heaters of this circuit are of contact type, making them prone to level fluctuations during process parameter changes. These fluctuations result in cascade tripping of the condensate

pumps, leading to reactor trip. In fact, most of the reactor trips have been originating from the steam–water circuit. Hence, these heaters were replaced by surface type heaters in 2004.

During commissioning, one main boiler feed pump seized due to loss of thrust. This was traced to possible loss of Net Positive Suction Head. The pump was replaced by a maintenance friendly pump with lower allowable Net Positive Suction Head.

The performance of the turbine-generator has been satisfactory. A system to log the operating time in off-normal frequency has been put in service.

9.5. Instrumentation and control

The reactor protection system has been adopted from Rap-sodie. During the initial phase of operation, several spurious trips due to noise pick-up by the channels and cables were encountered. These were systematically studied and several improvements like provision of shielding for cables carried out. The neutronic channels were replaced by state-of-the-art design in 1997–1998. Since then, the performance has been satisfactory, except generic failures in the low voltage (± 15 V) supply modules. Replacement of these modules is being carried out in stages. To obviate the need for a neutron source for start-up after log outages, two pre-startup channels with boron coated chambers were inducted in service in 1992.

Originally, interlocks were provided to stop raising of any control rod for more than 3 s to limit the reactivity addition. Automatic Lowering of Rods on control rod level discordance was set at 20 mm. These were overloading the duty cycles on the control rod drive motors and relays and prolonging the time for start-up and power raising. The 3 s interlock was deleted and control rod level discordance limit raised to 40 mm. Provision was made to inhibit the reactivity trip by a dead-man push button on the console during power raising. These have enabled quick start-up and power raising.

There were spurious alarms from the log-*P* channels even when the lin-*P* was less than this value. This was found to be due to the poorer resolution of the log-*P* scale at higher power levels. Hence, the trip setting from the log-*P* channel was raised from 110 to 125%. An interlock from the lin-*P* at 10% has also been provided to ensure the take-over by the log-*P* channel.

The performance of the Control Rod Drive Mechanisms has been satisfactory. Lower parts of the mechanisms were replaced twice, once in 1992 and another in 1997, due to leaks from the primary metallic bellows which operate in sodium. Friction force and drop time of all the mechanisms are measured half-yearly and have been found to be normal. A new system has been installed to measure the friction force on power, by exercising the control rods, i.e., raising one mechanism while simultaneously lowering another.

Due to obsolescence, both the computers of the Central Data Processing System were replaced—one in 1988 and the other in 1992. Investigations of a reactor trip in September 2003 revealed that the standby computer was the source of spikes, due to malfunctioning of aged relay cards. The computer has been taken out of service and is being replaced by a new generation computer sub-system. The safety critical and safety related functions

will be separate in this sub-system. Over the years several new features have been added to the Central Data Processing System. These include additional Personal Computer based systems for monitoring Reactor Assembly temperatures, radiation and air activity levels and steam generator/turbine data. In addition, dedicated fast recording systems for reactivity incidents and an interrupt Personal Computer to scan all trip contacts every 20 ms and dump the data on scram have been provided. An integrated message storage system to store all messages has also been added.

The performance of the Steam Generator Leak Detection System has been excellent. Major modifications were carried out on this system during commissioning. The surface type sodium heaters in the system were failing frequently and were replaced with immersion heaters. Drift and oscillations in the output of the Mass Spectrometers were found to be due to higher ambient temperature at the analyser RF head and were corrected. Software noise rejection techniques were incorporated in the computer to avoid electronic and process noise. The system is calibrated annually by injecting known amounts of H₂ into sodium in three stages and consistent results have been noticed. Based on operating experience, the signals of sputter ion pump current have been wired in the protection logic, instead of Mass Spectrometer signals used earlier. To avoid spurious reactor trips, it was decided to triplicate the Steam Generator Leak Detection System in each secondary loop and wire it to the reactor trip circuit in two out of three voting logic. One additional leak detection system was added to each loop in 2004 and the third system is scheduled to be added in May 2005. To detect water leaks at low sodium temperatures during power raising and hot shutdown, a thermal conductivity based hydrogen detection system has been installed in the cover gas of secondary sodium pumps. Hydrazine in feed water has been found to be a major source of Steam Generator Leak Detection System signal increase, and is being controlled.

The sodium instrumentation has been performing well. In October 1992, a lightning incident initiated many spurious alarms resulting in damage to ten alarm annunciation cards. Surge protection diodes were provided across the dc power supplies of alarm annunciators and sodium leak detector electronics and the grounding system was improved. The control power supplies of the sodium instrumentation were segregated loop-wise in 1997–1998, to avoid common mode failure. Several improvements were carried out in sodium pump speed control system to control the pump speed within ± 1 rpm in auto-mode operation.

9.6. Fuel handling

After the fuel handling incident of 1987, a mechanical swivelling lock was installed in the Charging Flask and wired to the plug rotation logic to avoid recurrence of the incident. Logic checking is being carried out before fuel handling operations. All fuel handling procedures have been reviewed and prepared in the check-list format. All fuel handling operations for the past 17 years have been smooth.

In 2004, there was a minor incident in which the pot containing an irradiated control rod got disconnected from its

top shield plug during transfer from the Reactor Containment Building to the Active Building. This was investigated and found to be due to the presence of sodium deposits in the pot, resulting in the shield plug not getting locked properly. Procedures for pot handling have been modified in the light of this incident.

9.7. Electrical power supply

The reliability of offsite power has been good. Performance of the emergency diesel generators is excellent, with almost 100% compliance to the demands to start and load. All the battery banks were replaced in 1994–1995, and again in 2005. The original 220 V ac Uninterruptible Power Supply system was found erratic during changeovers, and was replaced by a state-of-the-art system with higher rating in 1998–1999. Failure of power cables in the penetrations to Reactor Containment Building have occurred ten times; cable joints were remade. Assessment of the residual life of cables in the wake of a cable fire revealed that the cables have not aged and have a residual life of at least 20 years.

9.8. Plant layout

Leak tests of the Reactor Containment Building at 150 mbar (g) have been conducted once in two years since 1985. The leak rates have been found to be well below the acceptance level of 0.077% per hour. Cable penetrations and hair line cracks on the concrete surface have been found to be the major contributors to the leaks. The leaks are arrested by sealants.

The fire alarm system was replaced by a new, computer based analogue addressable system in 1998.

9.9. Plant auxiliaries

In the initial years, frequent pin hole leaks were occurring in the service water pipelines. Several pipelines were replaced. Carbon steel heat exchangers were replaced with cupro-nickel. Service water chemistry is now being maintained by using proprietary chemicals and the system has been trouble-free. The circulating water return line from cooling tower to the suction pit was found to have larger pressure drop than estimated in design. Since the resulting flow was inadequate for high power operation, a concrete tunnel was constructed in 1995 to return the cooling tower basin water to the plant.

The Demineralised water plant capacity (1 m³/h) was found inadequate for meeting the flushing requirements of steam–water circuit. A new plant of 3 m³/h capacity was commissioned in 1995. Demineralised water storage capacities have also been augmented. Due to obsolescence, the old cryogenic nitrogen plant was replaced by a new Pressure Swing Adsorption plant of 25 m³/h capacity in 1996.

With many systems refurbished/augmented, several teething problems encountered in the initial years of operation have been overcome. Improvements are planned in Air-conditioning and Ventilation System and fire water system.

9.10. Radiological safety

The general radiation levels in all the accessible location in Reactor Containment Building during operation at 17.4 MWt power was 0.1 mR/h. On top of the reactor, streaming field of 1.5 mR/h was seen at local spots. During the initial years of operation, higher radiation levels (45 mR/h max) were measured at some local spots in the Reactor Containment Building and shielding was augmented in these locations. The maximum activity released to atmosphere in a calendar year has been 128 Ci of ⁴¹Ar and maximum exposure around 800 person-millirem which is negligibly small compared to the allowed limits. During the past 19 years, there has been no significant event of abnormal radioactivity release, personnel or area contamination thus fortifying the worldwide view that this reactor concept gives low radiation doses to operating personnel and low releases to environment.

10. Conclusion

Though the Fast Breeder Test Reactor has not so far been operated at its nominal power and temperatures, the experience in construction and commissioning and the satisfactory performance of the fuel, sodium systems, steam generators and instrumentation during 19 years of operation have demonstrated the mastering of this multi-disciplinary technology for energy production and provided sufficient feedback to enable the launch of the Prototype Fast Breeder Reactor. About 400 modifications have been carried out in order to improve safety, reliability and availability. The feedback of these has been added input in the design of the Prototype Fast Breeder Reactor.

Towards realising the nominal design power, the feasibility of having a heterogeneous core, with carbide fuel at the central zone and MOX fuel in the outer zone, is under active consideration. This will also provide experience in the use of high Pu oxide fuels, which is the fuel choice for the commercial reactor.

References

- Baldev, R., et al., June 1997. Post Irradiation Examination of Mixed (Pu, U) C Fuels Irradiated in Fast Breeder Reactor—IAEA-TECDOC-1039 on Influence of High Dose Irradiation on Core Structural and Fuel Materials in Advanced Reactors, Obninsk, Russia.
- Bhoje, S.B., et al., 1985. Design and construction of the Fast Breeder Test Reactor. In: Proceedings of the IAEA International Symposium on Fast Breeder Reactors, Lyons, France.
- Bhoje, S.B., et al., 1991. Commissioning of Fast Breeder Test Reactor. In: International Conference on Fast Breeders and Related Fuel cycles, Kyoto, Japan.
- Chetal, S.C., May 2003. Status of Fast Reactor Development in India, April 2002–March 2003. In: TWG-FR Technical Committee Meeting on Review of National Programmes on Fast Reactors and Accelerator Driven Systems (ADS), Taejon, Republic of Korea.
- Ganguly, C., et al., January 1986. Development and fabrication of 70% PuC–30% UC fuel for Fast Breeder Test Reactor in India. Nucl. Tech., 72.

- Kapoor, R.P., et al., May 1996. Operating experience of Fast Breeder Test Reactor and its utilization as an experimental facility. In: Fifth Asian Symposium on Research Reactors, Taejon, Korea.
- Kapoor, R.P., et al., November 1998. Unusual occurrences in Fast Breeder Test Reactor. In: Technical Committee Meet IAEA-TECDOC-1180 on Unusual Occurrences During LMFBR Operation, Vienna, Austria.
- Kapoor, R.P., et al., October 1994. Safety related operating experience with Fast Breeder Test Reactor. In: International Topical Meeting on Sodium Cooled Fast Reactor Safety, Obninsk, Russia.
- Lee, S.M., et al., April 1990. Core physics parameters measured in the carbide fuelled Fast Breeder Test Reactor. In: International Conference on the Physics of Reactors—Operation, Design and Computation, Marseille, France.
- Ramalingam, P.V., April 1999. Operating experience of FBTR. In: Seventh International Conference on Nuclear Engineering, Tokyo, Japan.
- Srinivasan, G., et al., October 1988. Thermo-mechanical behaviour of FBTR reactor vessel due to natural convection in cover gas space. In: Fourth International Conference on Liquid Metal Engineering and Technology, Avignon, France.
- Suresh Kumar, K.V., et al., March 2002. Fast Breeder Test Reactor—15 years operating experience. In: Technical Meeting on Operational Experience and Decommissioning Experience with Fast Reactors, Cadarache, France.
- Suresh Kumar, K.V., et al., March 2002. Physics and engineering experiments in Fast Breeder Test Reactor. In: Technical Meeting on Operational Experience and Decommissioning Experience with Fast Reactors, Cadarache, France.
- Vaidyanathan, G., et al., October 1994. Dynamic tests related to under-cooling events in FBTR. In: International Topical Meeting on Sodium Cooled Fast Reactor safety, Obninsk, Russia.

