

Fast Breeder Test Reactor

20 Years
Operating Experience



**Indira Gandhi Centre for Atomic Research
Kalpakkam – 603 102
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20 Years Operating Experience

For India's growing energy needs, contribution from nuclear energy plays a significant role for providing energy security and diversity. Recognising the exploitable resources of Uranium and Thorium in the country, the Department has formulated three stage nuclear power programme. The first stage is based on Pressurised Heavy Water Reactors (PHWR) where natural Uranium is used as fuel. In the second stage, the fertile isotope of Uranium (U-238) is planned to be utilized to enhance energy production by a factor of over 100 through Fast Breeder Reactors operating on the U-Pu cycle initially. In the third stage, large thorium reserves will be utilized for energy production.

The Indira Gandhi Centre for Atomic Research (IGCAR), set up by DAE to develop fast breeder technology, has elaborate program in the areas of reactor design and R&D in the field of engineering development, inspection technologies, materials, radiochemistry, safety and fuel reprocessing.

The principal facility in the Centre is the Fast Breeder Test Reactor (FBTR), which not only provides experience in this area of high technology, but also serves as a test bed for developing fuels, materials for commercial reactors and training ground for human resource development.

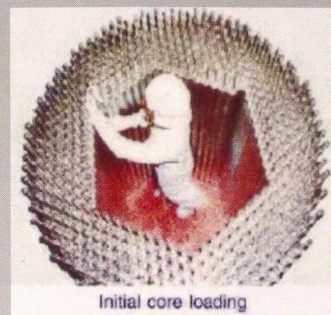
The construction of the reactor was started in 1974 and completed in 1984. The indigenous component of the reactor is more than 80%. After commissioning activities, the reactor achieved first criticality in October 1985. The reactor was operated up to 1 MWt for understanding physics and engineering of the system and also completing mandatory safety tests. The steam generator was commissioned in Jan 1993 and turbo-generator was synchronized to the grid in July 1997. The reactor has operated upto a power level of 17.4 MWt / 2.2 MWe.

FUEL PERFORMANCE

FBTR uses a unique indigenously designed fuel, a mixture of carbide of Uranium and Plutonium for driving the reactor. The fuel was designed indigenously and manufactured by BARC. The core has been kept small to assess fuel performance. Since first criticality, the reactor has completed 13 irradiation campaigns with an operating time of 1540 days.

The indigenously developed fuel has performed very well and reached a burn up level of 155 GWd/t (about 17 atom %), without any failure. This is far in excess of the set design value of 25 GWd/t.

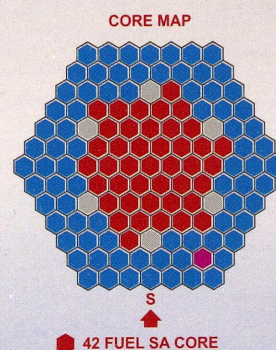
The progressive enhancement in fuel performance was achieved based on out-of-pile simulation tests, study of plant operation data, post irradiation examination (PIE) at intermediate stages and rigorous theoretical analysis.



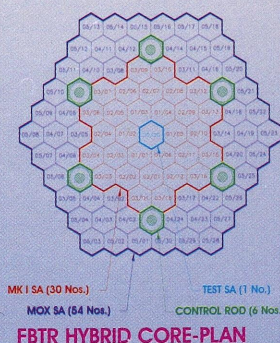
Initial core loading



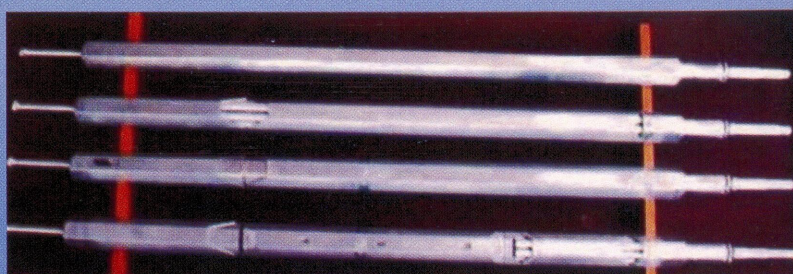
Reactor Vessel installation



42 FUEL SA CORE

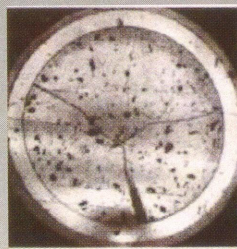


FBTR HYBRID CORE-PLAN

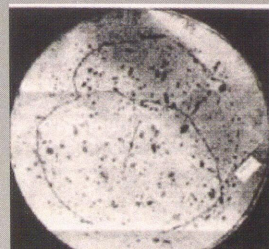




25000 MWd/t



50000 MWd/t



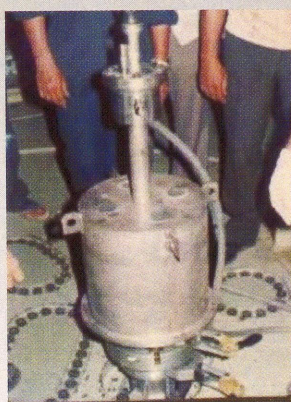
100 000 MWd/t

COOLANT CIRCUITS AND COMPONENTS

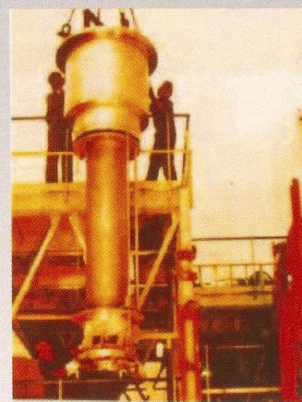
FBTR uses liquid sodium as coolant, which is reactive in contact with air/water. Maintenance of nuclear grade purity of the coolant is very important to minimize corrosion of structural materials and also avoid plugging of narrow flow passages in the reactor core and coolant circuits. This is achieved by controlling the routes for impurity ingress mainly through the inert argon cover gas and sodium pump seals and also by on-line purification and monitoring system to limit oxygen (< 10 ppm), hydrogen (< 2 ppm) and carbon (< 30 ppm) impurities in sodium.

Diverse leak detection methods for sodium leaks have been provided and procedures for handling leaks have been well understood and demonstrated in the plant. An incident of 75 kg of sodium leak in 2002 from a defective sodium valve in the primary sodium purification system housed in a nitrogen-inerted cabin was effectively handled and reactor brought on line within three months. Sodium service

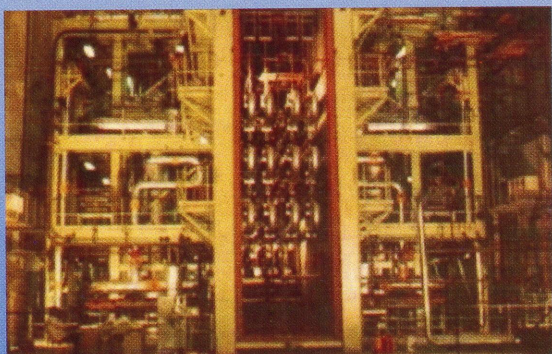
components especially the mechanical sodium pumps and sodium heated steam generators (SG) have performed exceptionally well. The pumps are vertical, centrifugal pumps, the shaft guided at the bottom by a hydrostatic bearing and at the top by a roller thrust bearing. The four sodium pumps have logged cumulative operation of 5,50,000 h without any major intervention. Sodium heated steam generators (SG) have operated satisfactorily for about 21,000 h without any SG tube leak. In case of leak, sodium-water reaction produces hydrogen which is monitored in ppb level by diverse methods viz. diffusion type and electrochemical hydrogen meters, pressure sensors and rupture discs to initiate safeguard actions to protect this critical equipment from damage. The diffusion type leak detection system in sodium has been triplicated to improve reliability. The superheated steam produced by SG is fed to a compact turbine, specially designed and manufactured by BHEL, to generate electricity. The TG has operated for 5228 h and generated 5.4 MU of electricity.



Sampling sodium from reactor



Primary Sodium Pump



Secondary Sodium Circuit with SG modules

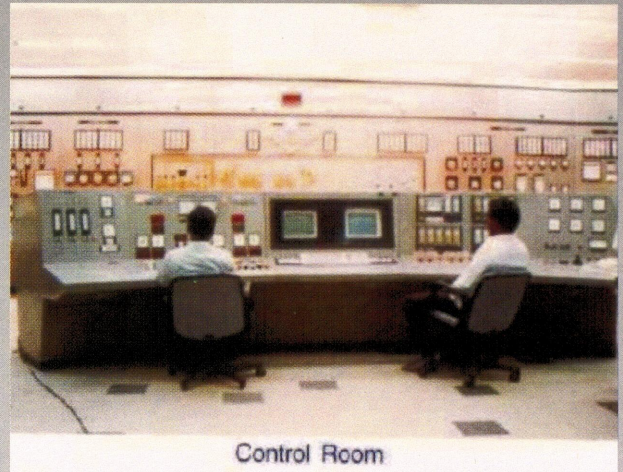


Turbine Generator

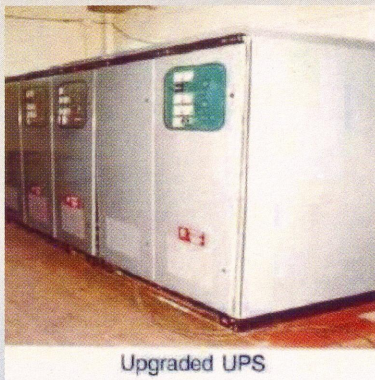
INSTRUMENTATION AND CONTROL

Six boron carbide control rods are used for start up, power control and shutdown of the reactor. The control rod drive mechanisms are being monitored for increase in friction force on-line and rod drop time. Two control rods are sufficient to shutdown the reactor. The power is controlled in manual mode by adjusting control rod to compensate for the loss of reactivity due to burnup and the operation in this mode has been found stable. The central data processing system processes the temperature signal at the outlet of individual fuel subassemblies and generates trip signals to limit fuel and clad hot spot temperatures. Due to upgradation in technology, these

systems have been replaced with state of art systems twice. Similar upgradation in technology is also carried out for neutronic instrumentation, control power supply and fire alarm systems. Based on operating experience, optimization in trip parameters has been carried out to improve reactor availability without compromising safety.



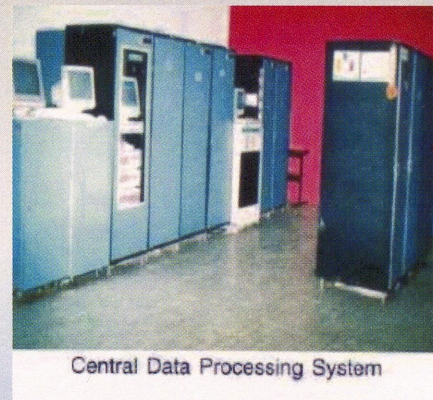
Control Room



Upgraded UPS



State-of-art Neutronic Channels



Central Data Processing System

COMPONENT HANDLING

Fuel handling is a group of operations consisting of charging, in-pile transfer and discharging of subassemblies. It is done with charging and discharging flasks with the reactor in shutdown state at 180°C . Based on experience, significant improvements in this system have been carried out and as a result about 500 in-pile operations with these flasks have been completed satisfactorily. Handling and replacement of sodium service components and in-reactor components have been successfully demonstrated by following basic principles of maintaining leak tightness of the system to prevent



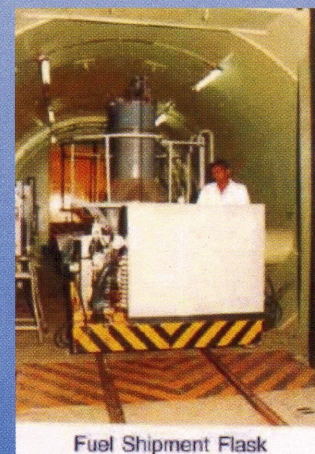
Fuel and Component



Dismantling Cell



Flask for sodium pump

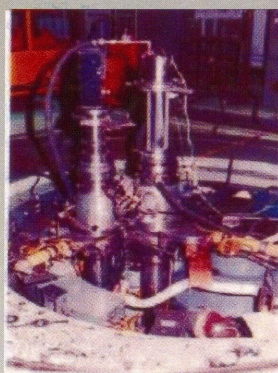


Fuel Shipment Flask

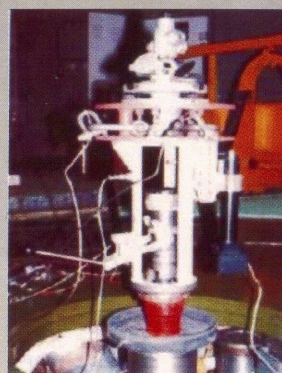
leak of radioactive cover gas and also retain nuclear grade purity of sodium. Decontamination and dismantling of sodium service components have also been successfully demonstrated.

IN-SERVICE INSPECTION

An elaborate in-service inspection and surveillance programme is in vogue in the plant. It consists of on-line monitoring and periodic inspections mainly oriented towards ensuring



Visual inspection of Reactor Vessel by Periscope



Inspection tool for checking FSA misalignment



In-vessel Cutting Machine

integrity of multiple barriers against release of radioactivity to environment, safe shutdown of the reactor under off-normal/incidental situations and availability of diverse provisions for safe removal of decay heat.

Gadgets like periscope for visual inspection of reactor vessel internals, ultrasonic under sodium scanner for inspection below the opaque liquid sodium level, core coordinate measuring device for checking misalignment between core subassemblies and top structure and several novel remote inspection/intervention techniques have been developed and successfully used.

SAFETY

Safety related physics and engineering tests were carried out to validate the design and also to ensure reactor safety under various postulated incidental situations. The data obtained from these tests is also used for improving mathematic modeling. Physics tests conducted include measurement of temperature, power and void coefficient of reactivity and these were negative as predicted, thus emphasizing the inherent safety features built in the design. Other experiments include capability to detect and localize failed fuel, foil irradiation for neutron flux measurement and kinetics experiments to ensure reactor stability. Engineering experiments demonstrated the capability of normal and emergency cooling systems to safely remove the reactor decay heat under all postulated incidental scenario especially the conduciveness of primary and secondary sodium circuits to operate in natural convection mode during station blackout.



Foil Irradiation experiment

RADIOLOGICAL PROTECTION

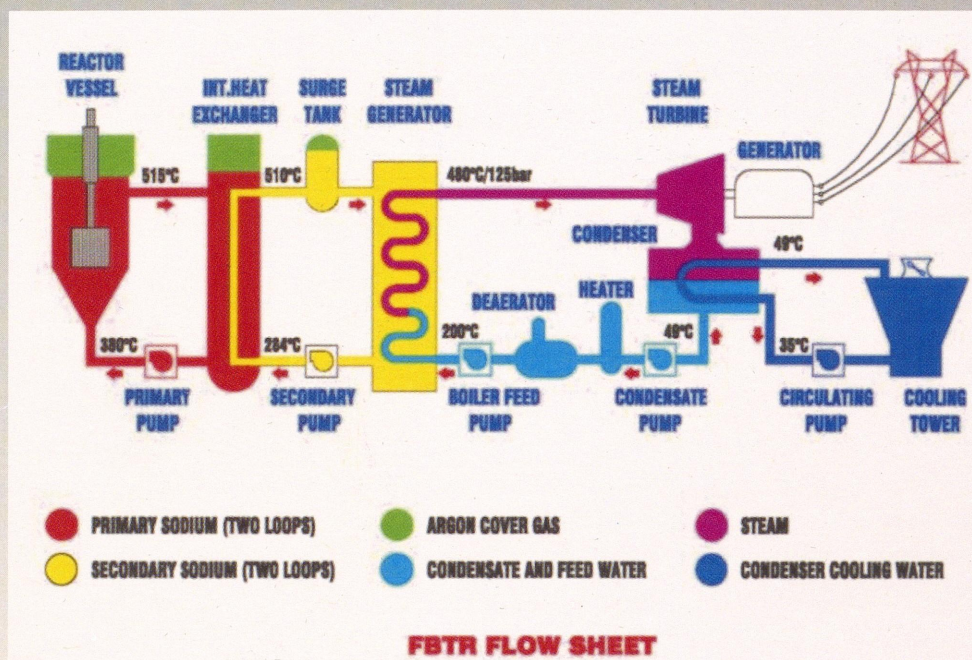
The radioactive primary coolant circuits are located in shielded cells. Hence, the general radiation level in all the accessible location in reactor containment building varies from 0.1 to 1.5 mR/h at various power levels. The total activity released to atmosphere in 20 years is 440 cu of Ar⁴¹. The cumulative occupational exposure is 6.7 man-rem during twenty years, which is negligibly small. Reactor containment building integrated leakage rate test is conducted biennially to ensure that the containment integrity is well within the specified limits of 0.1%/h. During the reactor operation history, there has been no incident of abnormal radioactivity release, personnel or area contamination thus fortifying the worldwide view that sodium cooled fast reactors are eco-friendly.

FUTURE PLANS

Presently FBTR is being used for irradiation of MOX fuel simulating the MOX fuel composition of PFBR fuel. It has logged a burn-up of 59 GWd/t without failure.

It is proposed to expand the core by adding MOX fuel of 44% PuO₂, surrounding the carbide fuel in stages to reach the final core configuration of 30 MK-I and 54 MOX subassemblies. The hybrid core is rated for 31 MWt / 10 MWe. Aging management studies for plant life extension have also been initiated to keep this test facility operational for fuel and structural material irradiation for the next 20 years. Operating experience generated at FBTR has been utilized for the design of PFBR.

Schematic Flow Sheet of FBTR



MAIN CHARACTERISTICS

<i>Parameters</i>		<i>Hybrid Core</i>	<i>Small Core</i>
Reactor Power	MWt/MWe	31 / 10	17.4/2.2
Fuel		Mark I 70% PuC + 30% UC MOX 44% PuO ₂ + 56% UO ₂	Mark I 70% PuC+30% UC Mark II 55% PuC+45% UC
No. of fuel SA		84	42
Maximum LHR(Mark I / MOX)	W/cm	400 / 250	400 (Mark I)
Maximum burnup	GWd/t	150 for Mark-I, 100 for MOX	150
Peak neutron flux	n/cm ² /s	3.5 E15	2.2 E15
Control rod material	90% B ¹⁰	B ₄ C	B ₄ C
Sodium temperature:			
Reactor Vessel inlet	°C	380	330
Reactor Vessel Outlet	°C	490	425
Primary sodium core flow	m ³ /h	942	600
Secondary sodium loop flow	m ³ /h	245	180
Sodium inventory	t	150	150
Feed water flow	t/h	46	30
Feed water / steam temp.	°C	190/480	190/405
Steam pressure	kg/cm ²	125	125
Steam generator	4 modules of once through type, 12.5 MWt capacity each.		
Turbine generator	16 stages, condensing type, 16.4 MWe air cooled,		
TG bypass circuit	100% capacity with dump condenser		