Available online at www.sciencedirect.com
SCIENCE DIRECT
Nuclear Engineering and Design 236 (2006) 852–860
Nuclear Engineering and Design
www.elsevier.com/locate/nucengdes

The design of the Prototype Fast Breeder Reactor

S.C. Chetal *, V. Balasubramaniyan, P. Chellapandi, P. Mohanakrishnan, P. Puthiyavinayagam, C.P. Pillai, S. Raghupathy, T.K. Shanmugham, C. Sivathanu Pillai

Indira Gandhi Centre for Atomic Research, Kalpakkam 603102, India

Received 11 March 2004; received in revised form 26 September 2005; accepted 29 September 2005

Abstract

India has a moderate uranium reserve and a large thorium reserve. The primary energy resource for electricity generation in the country is coal. The potential of other resources like gas, oil, wind, solar and biomass is very limited. The only viable and sustainable resource is the nuclear energy. Presently, Pressurised Heavy Water Reactors utilizing natural uranium are in operation/under construction and the plutonium generated from these reactors will be multiplied through breeding in fast breeder reactors. The successful construction, commissioning and operation of Fast Breeder Test Reactor at Kalpakkam has given confidence to embark on the construction of the Prototype Fast Breeder Reactor (PFBR). This paper describes the salient design features of PFBR including the design of the reactor core, reactor assembly, main heat transport systems, component handling, steam water system, electrical power systems, instrumentation and control, plant layout, safety and research and development.

1. Introduction

The nuclear energy program in India is being implemented in three stages. Natural uranium fuelled Pressurised Heavy Water Reactors (PHWR) are in operation and under construction in the first stage. Plutonium generated from PHWRs will be multiplied through breeding in Fast Breeder Reactors (FBR) in the second stage. This will facilitate launching of a large scale Th–^{233}U fuel cycle in the third stage. FBRs also utilize natural uranium fuel very effectively (∼75%) through breeding and thus provide a rapid energy growth potential (300 GWe for about 30 years). They also constitute a clean source of power unlike fossil fuel power stations. Several FBRs worldwide (like EBR-II, Phenix, PFR, BN reactors) are witness to their environment friendly performance. The use of thorium in FBRs in the third stage will make it a much larger resource (1500 billion tonnes of coal equivalent) than the combined coal, oil and gas resources. Thus FBRs will provide long term energy security utilizing the indigenous uranium and abundant thorium reserves.

Indira Gandhi Centre for Atomic Research, Kalpakkam is responsible for the establishment of fast breeder technology in the country. The commissioning of a 40 MWt/13 MWe Fast Breeder Test Reactor (FBTR) at Kalpakkam in 1985 marked the beginning of a FBR programme in India. The synchronization of FBTR with the grid was achieved in July 1997. Since then, considerable operating experience has been gained in the operation of sodium systems and steam water system including the steam generators. This has given confidence to commence the next phase of the FBR programme, i.e., the construction of the Prototype Fast Breeder Reactor (PFBR).

2. Prototype Fast Breeder Reactor

The PFBR is a 500 MWe, sodium cooled, pool type, mixed oxide (MOX) fuelled reactor having two secondary loops. The reactor is located at Kalpakkam, close to the 2 × 220 MWe PHWR units of the Madras Atomic Power Station (MAPS). Kalpakkam is situated at 68 km south of Chennai on the coast of Bay of Bengal. The primary objective of the PFBR is to demonstrate techno-economic viability of fast breeder reactors on an industrial scale. The reactor power is chosen to enable adoption of a standard turbine as used in fossil power stations, to have a standardized design of reactor components resulting in further reduction of capital cost and construction time in future and compatibility with regional grids. Mixed carbide fuel of PuC-UC

* Corresponding author. Tel.: +91 4114 280105; fax: +91 4114 280068.
E-mail address: chetal@igcar.ernet.in (S.C. Chetal).

0029-5493/$ – see front matter © 2006 Published by Elsevier B.V.
doi:10.1016/j.nucengdes.2005.09.025

Table 1
Main characteristics

Thermal power (MWt)	1250
Electric output (MWe)	500
Core height (mm)	1000
Core diameter (mm)	1900
Fuel	PuO_2–UO_2
Fuel pin outer diameter (mm)	6.6
Pins per fuel subassembly	217
Fuel clad material	20% CW D9
Diameter of main vessel (mm)	12900
Primary circuit layout	Pool
Primary inlet/outlet temp (°C)	397/547
Steam temperature (°C)	490
Steam pressure (MPa)	16.6
Reactor containment	Rectangular
Plant life (years)	40
No. of shutdown systems	2
No. of decay heat removal systems	2

is used in Fast Breeder Test Reactor due to non-availability of enriched uranium for MOX fuel option. The PFBR being a commercial demonstration plant, a proven fuel cycle is essential. The MOX fuel is selected on account of its proven capability of safe operation to high burnup, ease of fabrication and proven reprocessing. Detailed studies were carried out to compare the advantages and disadvantages of pool and loop type designs. Better safety features of pool type, i.e., main vessel with no nozzles leading to high integrity of the vessel, relatively large thermal inertia leading to ease in design of decay heat removal with lower heat removal capacity requirements and availability of more time for the operator to act, and large diameter of the main vessel with internals leading to significantly lower strain in the main vessel in case of core disruptive accident, led to selection of pool type design for primary circuit configuration. The pool type concept also enables further extension of the design to larger power reactors in the future. The excellent performance of sodium components in fast breeder reactors that obviates the need for easy maintenance on these components, which is possible in a loop type reactor. The main vessel is made of highly ductile AISI 316 LN material and it satisfies leak before break criteria. A two loop design has been adopted in view of its economical benefits and it meets the safety requirements. The main characteristics are given in Table 1.

The steam cycle selected is steam reheat with integrated steam generator (steam generator) instead of sodium reheat to simplify the design of the steam generator and associated circuits and for ease of operation. A dedicated safety grade decay heat removal system is provided and it eliminated the necessity of designing secondary sodium and steam water systems as safety grade systems. The reactor is designed with core sodium outlet temperature of 547 °C which is made possible due to use of SS 316 LN as structural material and capability to perform inelastic analysis for creep-fatigue damage assessment. Based on the choice of the steam generator material as ASTM Gr 91 (modified 9Cr–1Mo) and optimization studies, the steam parameters at turbine stop valve of 16.6 MPa and 490 °C have been fixed. The overall cycle efficiency is 40%.

All reactor structures, systems and components are classified systematically based on their safety functions and the requirements under seismic events have also been identified. The reactor is designed to meet the regulatory requirements of the Atomic Energy Regulatory Board (AERB). The computer codes developed in-house have been validated and approved by the regulatory authority. The safety of Prototype Fast Breeder Reactor is evaluated by analyzing the plant response to various events affecting the plant. The events with a frequency of occurrence $\geq 10^{-6}$ year^{-1} are considered as a Design Basis Event (DBE) and these have been further classified into categories I–IV events. The safety criteria adopted satisfies the requirement that a design basis event should be detected by at least two diverse parameters and the reactor should be automatically shut down before the design safety limits on coolant, clad and fuel are reached. The design safety limits on the coolant, clad and fuel temperatures are listed in Table 2 for the different categories of design basis events. Analysis of the design basis events have been carried out and it is seen that the fuel clad and bulk sodium temperatures are well within the respective category limits for all the design basis events.

3. Reactor core

Fig. 1 shows the core configuration. A homogenous core concept with two fissile enrichment zones of 21/28% PuO_2 is adopted for power flattening. The active core where most of the nuclear heat is generated consists of 181 fuel subassemblies. Each fuel subassembly contains 217 helium bonded pins of 6.6 mm diameter. Each pin has 1000 mm column of annular MOX fuel pellets and 300 mm each of upper and lower blanket columns. The pin diameter of 6.6 mm is selected considering

Table 2
Design safety limits of fuel, clad and sodium under various design basis events

DBE category	Mean SA hotspot sodium temperature (°C)	Bulk sodium temperature (°C) in		SA clad hotspot temperature (°C)		Fuel hotspot temperature (°C) or melt fraction
		Hot pool	Cold pool	In active core	In storage locations	
I	580	547	397	700	500	2700
II	650	–	540	800	600	2700
III	No bulk sodium boiling	–	–	900	650	10% melting in the maximum rated fuel pin
IV	No bulk sodium boiling	–	–	1200	950	50% melting in the maximum rated fuel pin

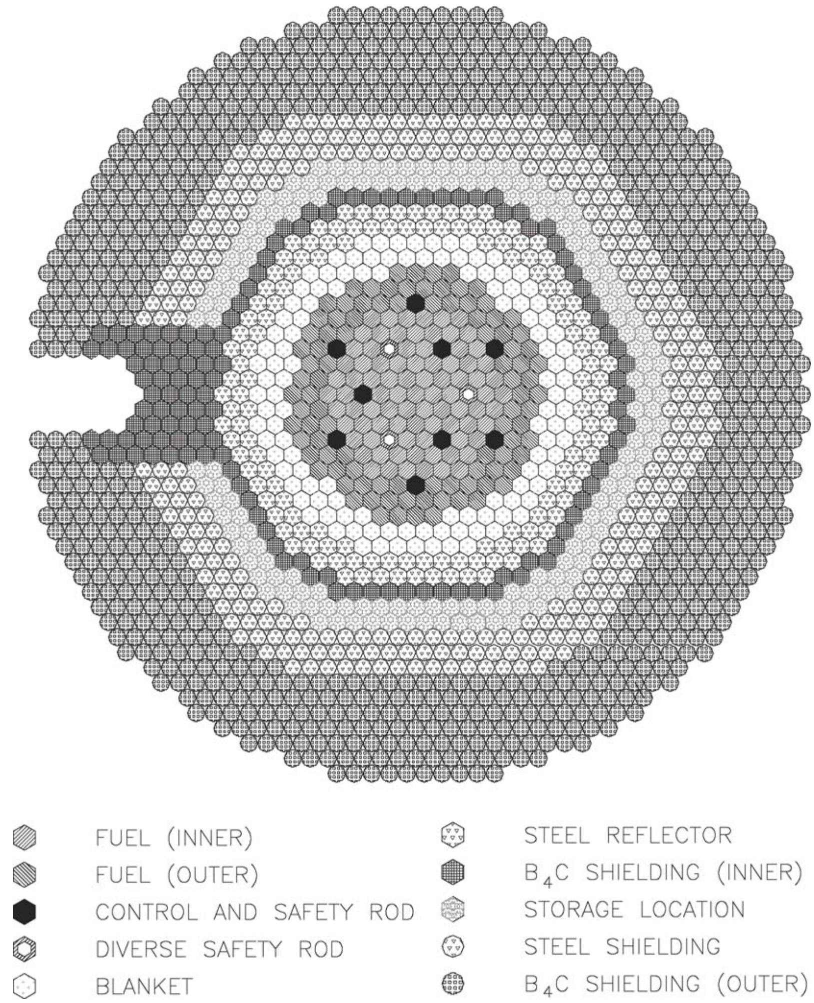

Fig. 1. Core configuration.

the Pu inventory, fuel fabrication cost and breeding. The clad material used is 20%CW 15Ni–14Cr–2Mo + Si + Ti (D9). The maximum linear power in the fuel pin is 450 W/cm. The initial peak fuel burnup is limited to 100 GWd/t due to deformation and swelling behaviour of the D9 material used. In the long run, use of improved clad material of D9 (with addition of tramp elements phosphorus and boron) and use of ferritic steel of 9Cr–1Mo type for wrapper material are planned to achieve a targeted burnup up to 200 GWd/t. Spent subassemblies are stored for one campaign in internal storage with one-third of the active core being replaced during every fuel handling campaign. Two rows of blanket subassemblies are provided surrounding the inner and outer fuel regions (Fig. 1). Twelve absorber rods, i.e., nine control and safety rods (CSR) and three diverse safety rods (DSR) are arranged in two rings. Two independent and diverse shut down systems are provided for ensuring safe shut down of the reactor even when one system is not available. Both the systems are designed to shutdown the reactor in less than 1 s. In addition, axial shielding is provided within the subassemblies and radial shielding subassemblies are provided within the core. These are optimized to order to have the required flux at in-vessel neutron detector locations. They also serve the purpose of limiting the activation of the secondary sodium, radiation damage of grid plate and helium production in core cover plate.

4. Reactor assembly

The entire primary sodium circuit is contained in a large diameter vessel (Ø 12,900 mm) called main vessel and consists of core, primary pumps, intermediate heat exchanger and primary pipe connecting the pumps and the grid plate (Fig. 2). The vessel has no penetrations and is welded at the top to the roof slab. The main vessel is cooled by cold sodium to enhance its structural integrity. The core subassemblies are supported on the grid plate, which in turn is supported on the core support structure. A core catcher provided below the core support structure, is designed to take care of melt down of seven subassemblies and prevents the core debris from coming in contact with the main vessel. The main vessel is surrounded by the safety vessel, closely following the shape of the main vessel, with a nominal gap of 300 mm to permit robotic and ultrasonic inspection of the vessels. The safety vessel also helps to keep the sodium level above the inlet windows of the intermediate heat exchanger ensuring continued cooling of the core in case of a leak of main vessel. The inter

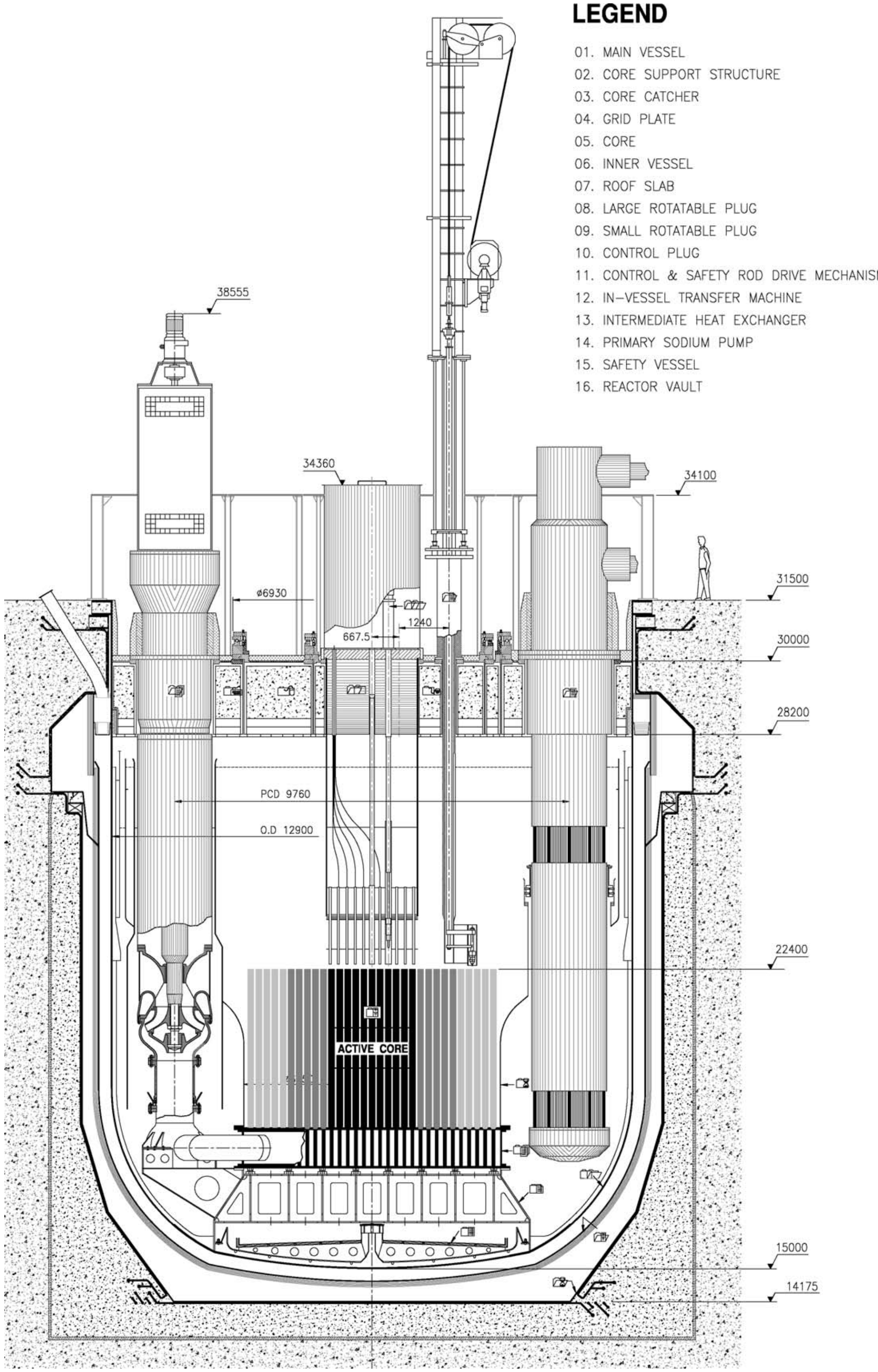

Fig. 2. Reactor assembly.

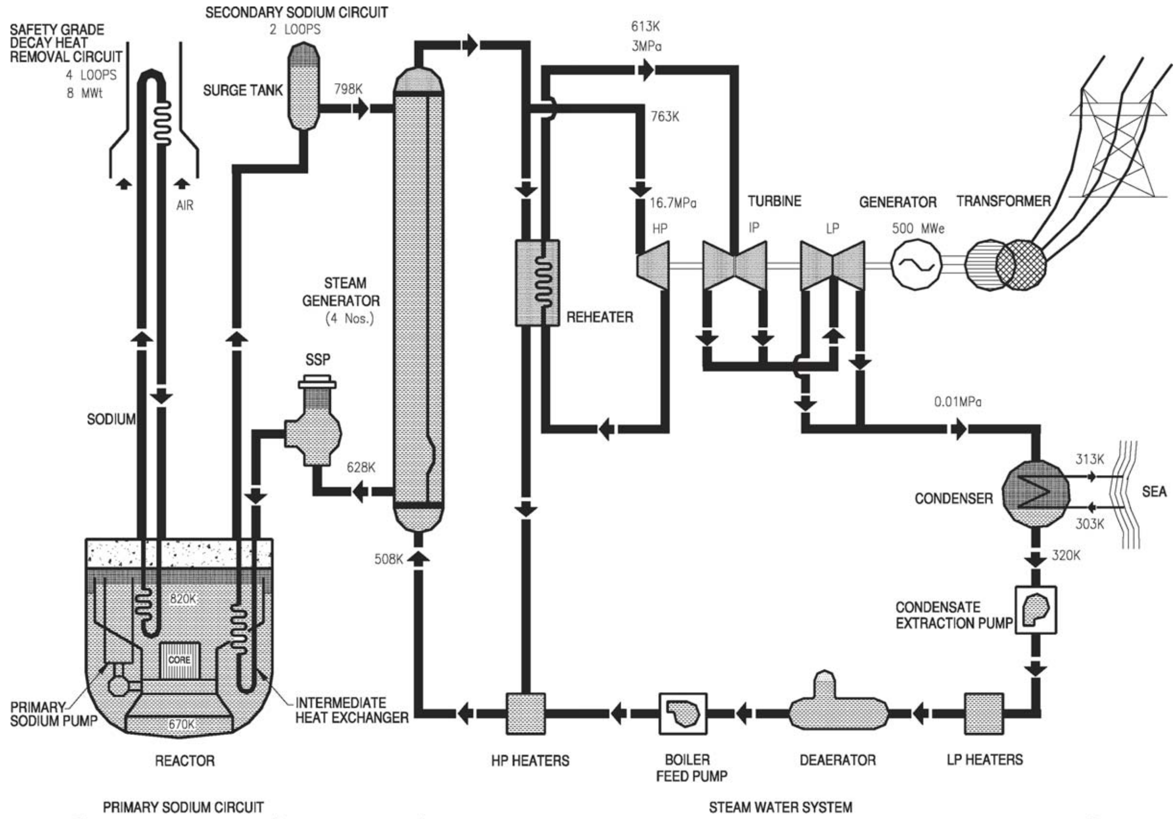

Fig. 3. Flow sheet.

space between main and safety vessel is filled with inert nitrogen. An inner vessel separates the hot and cold pools of sodium. The main vessel is closed at its top by top shield, which includes roof slab, large and small rotatable plugs and control plug. The top shield is a box structure made from special carbon steel plates and is filled with heavy density concrete ($\rho = 3500\,\mathrm{kg/m^3}$) and provides thermal and biological shielding in the top axial direction. The principal material of construction is SS 316 LN for the vessels and boiler quality carbon steel for top shield. The biological shielding in the radial and bottom axial direction outside the main vessel is provided by the reactor vault concrete.

5. Main heat transport system

Fig. 3 shows the flow sheet of main heat transport system. Liquid sodium is circulated through the core using two primary sodium pumps. The sodium enters the core at 397 °C and leaves at 547 °C. The hot primary sodium is radioactive and is not used directly to produce steam, instead it transfers the heat to secondary sodium through four intermediate heat exchangers. The non-radioactive secondary sodium is circulated through two independent secondary loops, each having a sodium pump, two intermediate heat exchangers and four steam generators. The choice of four steam generators per loop is based on overall optimization studies carried out considering capital cost, outage cost and operation cost with three steam generators in the affected loop in case of a leak in one steam generator. The primary and secondary pumps are vertical, single stage and single suction centrifugal type, with variable speed ac drives and are provided with flywheels to meet the flow coast down requirements of 8 and 4 s, respectively. An ac pony motor of 30 kW rating is additionally provided for each of the primary pumps. The steam generator is a once through integrated type design using straight tubes and an expansion bend in each tube. The decay heat is removed using the operation grade decay heat removal system of maximum 20 MWt capacity in the steam water system under normal conditions. In case of off-site power failure or non-availability of steam water system, the decay heat is removed by a passive safety grade decay heat removal circuit consisting of four independent loops. Each safety grade decay heat removal loop is rated for 8 MWt and consists of a decay heat exchanger immersed in the hot pool, one sodium/air heat exchanger, associated sodium piping, tanks and air dampers. Diversity is provided for decay heat exchanger, air heat exchanger and dampers. The circulation of sodium and air is by natural convection.

6. Component handling

Fuel handling is done after 185 effective full power days with reactor in shutdown condition at a sodium temperature of 200 °C. Two rotatable plugs and a transfer arm are provided for in-vessel handling of core subassemblies (Fig. 4). For ex-vessel handling, an inclined fuel transfer machine and cell transfer machines are used (Fig. 5). The preheated fresh subassemblies are transferred to the core using cell transfer machine and inclined fuel transfer machine. The spent fuel subassemblies are stored inside the

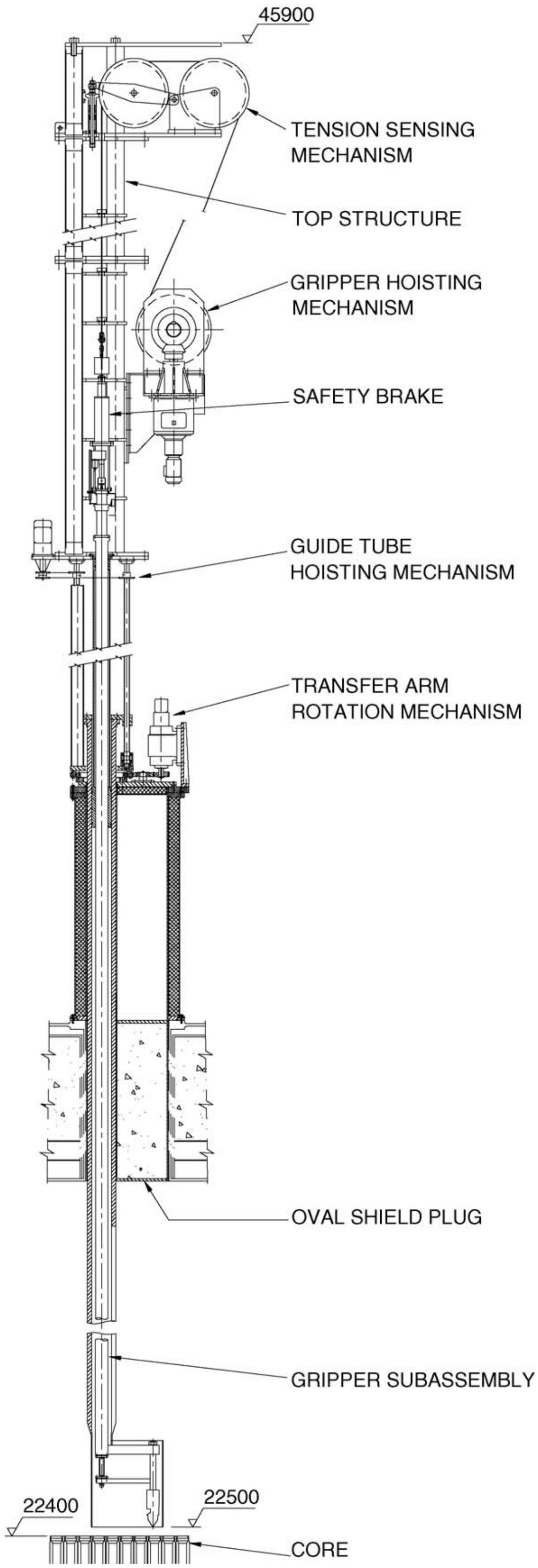

Fig. 4. Transfer arm.

main vessel for one campaign and then shifted to a dematerialized water filled spent subassembly storage bay pool located in fuel building. Sodium sticking to subassembly is washed in spent subassembly washing facility. Leak tight shielded flasks are provided for special handling of components like primary sodium pumps, intermediate heat exchangers, decay heat exchangers, absorber rod drive mechanisms and transfer arm. The components are decontaminated in a separate facility provided within reactor containment building before they are taken for maintenance. After decontamination, the above components are shifted to a separate building for maintenance purposes.

7. Steam water system

The steam water system uses a reheat, regenerative cycle using live steam for reheating. High pressure superheated steam from the steam generators drives a turbo alternator of 500 MWe capacity. The turbine design is standard, used in fossil fired thermal power plants. $3 \times 50\%$ capacity boiler feed pumps are provided to deliver feed water to steam generators. Two of the pumps are turbo driven and the remaining one is motor driven. Feed water is heated in six stages, five in surface type feed water heaters and one in direct contact deaerator. A steam separator is provided at the common outlet of the steam generator for startup purpose. A turbine bypass of 60% capacity is also provided. Condenser is cooled by seawater in a once through system. The condenser tubes are made of titanium.

8. Electrical power systems

The plant is provided with both off-site and on-site electrical power supply systems. Power is transmitted at 220 kV. The plant is connected to the southern regional grid. A 220 kV substation with five transmission lines and double circuit ties to Madras Atomic Power Station (MAPS) 220 kV bus is provided. An indoor switchyard is selected to safeguard and increase the reliability of the electrical equipment against the saline atmosphere. Class III power supply of the plant is provided with standby emergency diesel generators. Four diesel generator sets, each rated to supply 50% of the total emergency power supply demand with a rating of 3 MVA are provided as on-site sources of ac power. Power supplies for instrumentation and control equipment are provided by Class I no-break 48 V and 220 V dc and Class II no-break 240 V, 50 Hz, 1-phase power supply.

9. Instrumentation and control

The Reactor power is controlled manually. The burnup compensation of reactivity is very small (22 pcm/d). Six fission chambers with a sensitivity of 1 cps/nv are provided above the core, at the bottom of control plug from safety considerations. Cover gas activity and delayed neutrons in the primary sodium are monitored for failed fuel detection. Sodium samples from each fuel subassembly outlet are taken using three selector valves for locating failed subassembly. Two chromel–alumel thermocouples are provided to monitor the temperature of sodium at the outlet of each fuel subassembly. Flow delivered

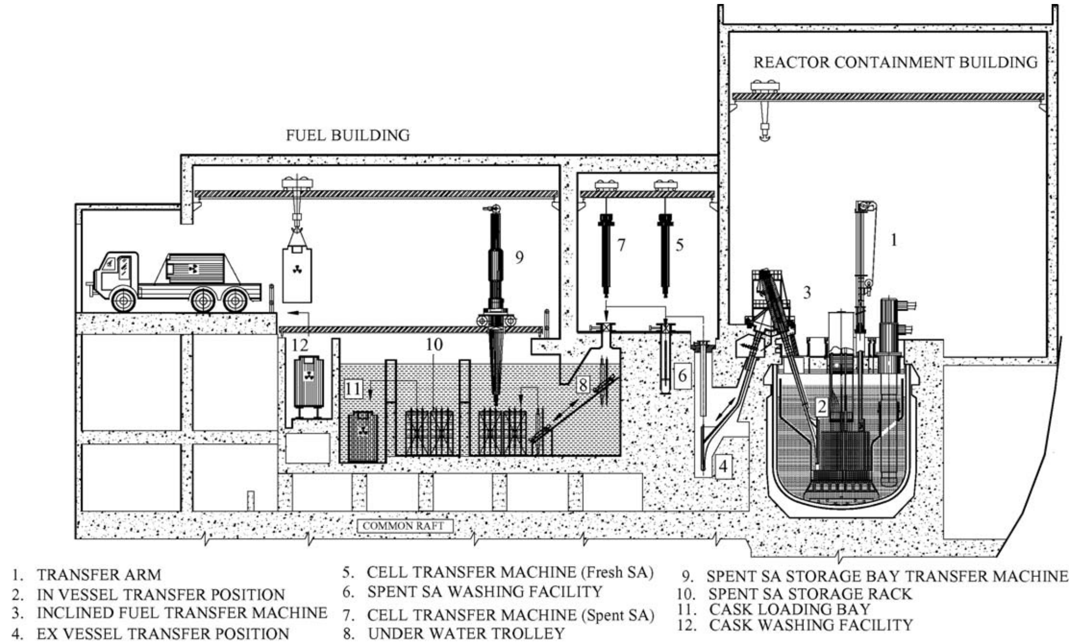

Fig. 5. Spent fuel handling scheme.

by the sodium pumps are measured using eddy current flow meter and safety action is taken on power to flow ratio. These provisions ensure that there are at least two diverse safety parameters to shut down the reactor safely for each design basis event. Ten SCRAM parameters from core monitoring systems and heat transport systems are connected to plant protection system to automatically shutdown the reactor (Fig. 6) in case any parameter crosses the limit. Steam generator tube leaks are detected by a leak detector (hydrogen in sodium) provided at the outlet of each steam generator module and an additional detector provided in the common outlet header. Two 'Hydrogen in Argon' detectors are installed in the cover gas space of surge tank.

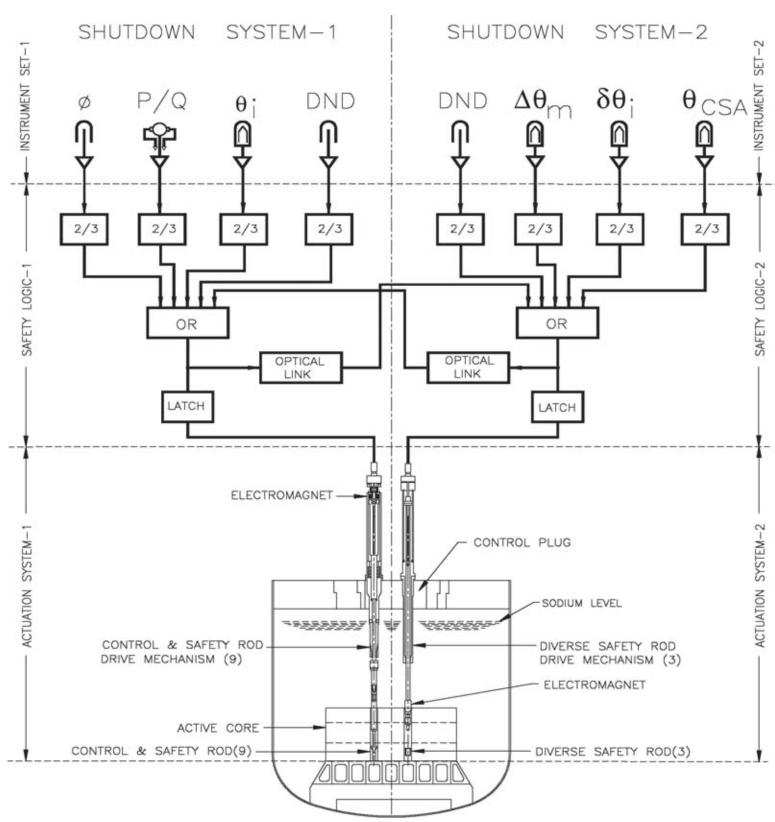

Fig. 6. Shutdown system.

Acoustic leak detectors are also installed at various locations on the outer shell of steam generator. Crack opening of air heat exchanger dampers and sodium flow monitoring ensures poised condition for safety grade decay heat removal whose operation is automatic. Separate backup control room and fuel handling control rooms are also provided. Instrumentation directly concerned with reactor safety is designed using hardwired systems except core thermocouples, which are processed by real time computers. Non nuclear systems use state of the art distributed digital control system to take advantage of multiplexed signal transmission and reduced cabling leading to cost savings. Safety signals are converted into digital form and are connected to the distributed digital control system for display in control room.

10. Plant layout

The plant layout is evolved on the basis of a single unit. The reactor assembly, primary sodium purification, primary argon cover gas system including its tanks and cover gas purification and decontamination facility are housed in a rectangular reactor containment building. Each of the two steam generator buildings houses four steam generators and associated components and piping. The reactor containment building, steam generator building and fuel building are connected and laid on a common base raft (Fig. 7). This minimizes the differential movement in piping and facilitates satisfactory working of inclined fuel transfer machine. In addition, control building, two electrical buildings and radwaste building are also laid on the common raft and connected to form a nuclear island, to reduce the magnitude of structural response under seismic loads and length of cables. The elevation of the raft is +12 m for reactor containment building and steam generator buildings and +14 m for the other buildings of nuclear island from functional, economic and seismic considerations (finished floor elevation is +30 m). A service building is provided to cater to the needs of plant services. The turbine building layout is selected such that the turbine missile trajectory is outside the safety related buildings. The finished floor levels of all safety related structures are above the design basis flood level estimated for 1000 years return period. The finished floor levels of non-safety related structures is based on design basis flood level of 100 years and these structures are located 1.5 m lower than the safety related structures from cost considerations. The diesel generators are housed in two separate safety related buildings. A 100 m tall stack is located close to the radwaste building.

11. Safety

To ensure safety, a defense-in-depth philosophy, consisting of three levels of safety, i.e., design with adequate safety margin, early detection of abnormal events to prevent accidents and mitigation of consequences of accidents, if any, is adopted. The reactor is designed with various engineered safety features such as two independent fast acting diverse shutdown systems and decay heat removal systems with passive features of natural circulation of intermediate sodium/air, along with diversity in design of decay heat exchanger and air heat exchanger. Core catcher and containment are provided as defense in depth for beyond design basis events. In the event of a total instantaneous blockage of a single subassembly, fuel melting starts and further the melting progresses at the most to the neighboring six subassemblies before the reactor is shut down by delayed neutron detection system. The molten fuel and debris are collected in a suitably dispersed manner to avoid criticality and ensuring long term cooling. The amount of molten fuel released in the melting of seven subassemblies is only 0.3 t. On the other hand, calculations indicate that 1 t of fuel in the most reactive configuration is required to achieve recriticality. Therefore, the

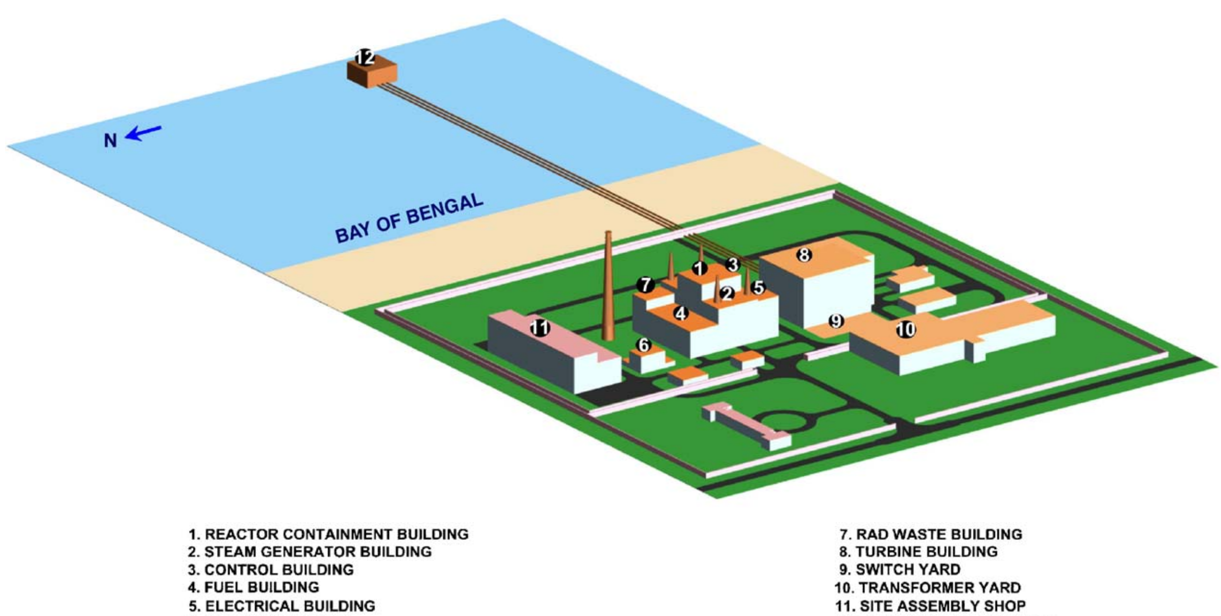

Fig. 7. Plant layout.

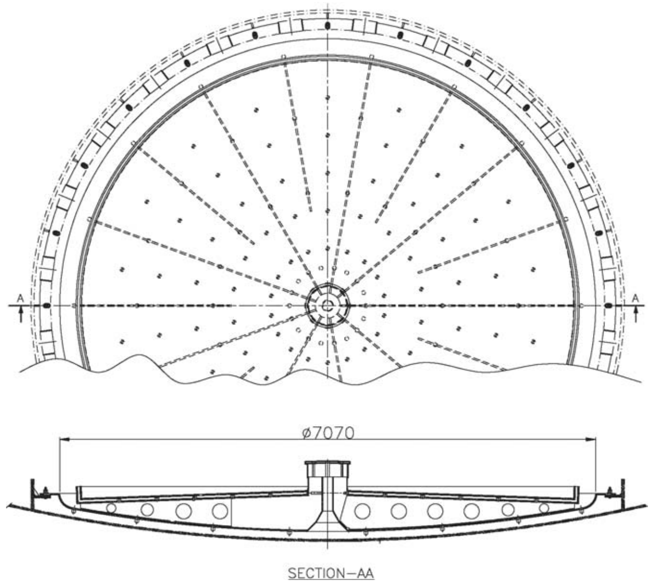

Fig. 8. Core catcher.

slumped molten fuel of seven subassemblies does not have any recriticality potential. Despite this, a core catcher is provided below the core support structure as a defense in depth measure. A chimney is provided at the center of the core catcher to aid natural convection flow of sodium (Fig. 8). The temperature of core catcher bottom plate is estimated as 543 °C.

Selection of design features, detailed design analysis and rigorous manufacturing specifications, minimize the risk of sodium leaks from components, piping and leaks resulting in sodium–water reaction in steam generator. Nevertheless, provisions have been made for early detection of sodium leaks and sodium–water reaction in steam generator and safety actions to minimize the consequence of the leaks. Additionally, the design also provides for in-service inspection of the main and safety vessels, secondary sodium piping and steam generator.

Atomic Energy Regulatory Board safety code on safety in nuclear power plant site states that a study on the probability of occurrence of an aircraft crashing on the nuclear power plant shall be made taking into account the flight frequencies at the nearest airfield and its distance from the site. The code stipulates that if the probability is more than 10^{-7} year^{-1}, then the site shall be deemed unsuitable. Appropriate screening distance value may be used to obtain the above probability value. In the absence of site specific data, screening distance values have been specified and the rejection standards are less than 8 km from major airports and 5 km from small airfields (screening distance value specified in the safety guide is based on probability of aircraft crash on the nuclear plant to be less than 10^{-7} year^{-1} for major airports having flights ~100,000 year^{-1}). The reactor site for Prototype Fast Breeder Reactor is located at a distance of 47 km from the nearest Chennai airport, and accordingly aircraft crash is not considered in the design.

The plant site is classified as zone III and there is no capable fault nearby. The plant is designed for safe shutdown earthquake and all sodium systems irrespective of safety classification are designed for both operating basis earthquake and safe shutdown earthquake to avoid sodium fire. All the sodium piping inside the reactor containment building are provided with double envelope with nitrogen inerting to avoid sodium fire. The structural integrity of primary containment, intermediate heat exchanger and decay heat exchanger is assured under core disruptive accident, which results in an energy release of 100 MJ, the theoretically assessed upper bound value for energy release. The use of leak before break approach is justified and provisions for leak detection and mitigation in case of a sodium fire are made systematically. A rectangular, single, non-vented, reinforced concrete containment designed for 25 kPa, is provided. The maximum pressure inside the containment is estimated to be 20 kPa with the conservative assumption of instantaneous burning of all the sodium that is ejected above roof slab under a core disruptive accident. The containment is designed such that the dose limits at site boundary for design basis accident of 100 mSv is not exceeded under core disruptive accident.

12. R&D

R&D is carried out both in-house in the various laboratories set up at Indira Gandhi Center for Atomic Research and in collaboration with other R&D organizations and industries on reactor fuels, sodium technology, reactor engineering, reactor physics and shielding, component development, materials, non-destructive examination, structural mechanics, thermal hydraulics, instrumentation and control, reprocessing, safety, etc. Facilities have been set up to test the prototype components in air, argon and sodium to validate the design of components.

13. Summary

India's energy requirements are large and only nuclear energy produced through fast breeder reactors can meet a significant part of the requirements. The Prototype Fast Breeder Reactor has been designed based on FBTR experience. The design is validated by large amount of R&D carried out in various disciplines. The construction of PFBR at Kalpakkam started in August 2003. Site excavation and geotechnical investigations were carried out followed by first pour of concrete in October 2004. The first criticality of the plant is planned in 2010 and the commercial operation is expected in 2011. PFBR will be the forerunner of a series of fast reactors planned in the coming decades, which will provide energy security to the country.