DESIGN OF 500 MWe PROTOTYPE FAST BREEDER REACTOR

S. Raghupathy, Om Pal Singh, S. Govindarajan, S.C. Chetal & S.B. Bhoje
Indira Gandhi Centre for Atomic Research

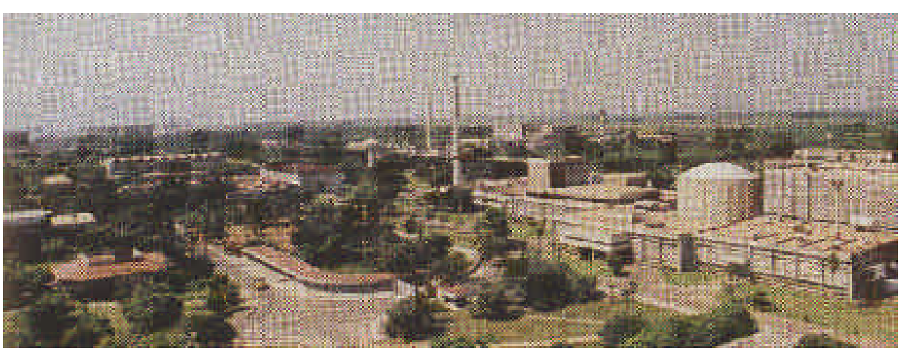
Electricity needs of India are presently being met primarily by coal resources with supplementary contribution from hydro resources. However, future needs are high and can not be met by coal resources for more than 50-70 years. Nuclear resources have potential to meet the needs for many centuries and they are superior to coal from environmental considerations. Nuclear technology in India has now matured and is poised to make a substantial contribution to the country's electricity needs.

Nuclear power programme is being implemented in three stages considering limited uranium resources and vast thorium resources in the country. The first stage consists of investing natural uranium in pressurized heavy water reactors (PHWR). This stage has the potential of 10 GWe. The second stage would use, in a closed cycle, the plutonium and depleted uranium extracted from the PHWR spent fuel in the Fast Breeder Reactors (FBR). This stage has a potential of 300 GWe.

Indira Gandhi Centre for Atomic Research (IGCAR) was instituted at Kalpakkam in Tamil Nadu in 1971, with the mission of developing the technology of FBR. A host of multi-disciplinary laboratories was established in the centre around the central facility of the 40 MWt Fast Breeder Test Reactor (FBTR). The FBTR which went critical in 1985, has given valuable experience in the operation of sodium systems including steam generators, and has provided a test bed for various experiments and fuel irradiation. Design of 500 MWe prototype fast breeder reactor (PFBR) was started in the 80's. The design incorporates state-of-art safety criteria and has benefited from the operating experience of the FBTR and the economic studies undertaken in the 90's by the French and the Russians to make FBR economically competitive vis-à-vis contemporary thermal reactors.

The purpose of constructing the PFBR is to demonstrate on an industrial scale the techno-economic viability and indigenous design capability of an FBR power plant in India. The current proposal is to build four more reactors of the PFBR design before 2020 A.D, and this has dictated the need for economic competitiveness of the reactor though being a prototype. The general guidelines that have been followed in the design are as follows:

- The concepts selected for the PFBR should be based on the operating experiences of FBR. Any innovative design concept is to be incorporated only after thorough research and prototyype test.
- The design must be standardized to a fair extent so that the follow on plants will have minimum modifications.
- High breeding ratio is not an essential requirement for PFBR. It should be a breeder (BR > 1). The fuel cycle economics is an important factor at this stage of development.
- The reactor shall meet the safety criteria issued by the Atomic Energy Regulatory Board (AERB) for PFBR.
- Design simplification, without compromise on safety and reliability, should be aimed at in order to facilitate speedier construction and easy operation.



Panoramic view of the Indira Gandhi Centre for Atomic Research, Kalpakkam, Tamil Nadu

Main Options

Reactor Power

The power generation capacity of PFBR has been chosen as 500 MWe. Specific capital cost is lower for 500 MWe units than for 250 MWe units.Scaling up the unit size to 500 MWe from FBTR was considered in detail. Manufacturing technology development has been completed successfully through industries, on full scale key nuclear components. Successful operation of about 20 units of 500 MWe coal fIred thermal power plants and construction of 500 MWe PHWR units currently in the country support this choice.

MOX Fuel

FBTR uses mixed carbide fuel due to certain constraints. PFBR being a commercial demonstration plant, a proven fuel cycle is essential. The reprocessing technology for oxide fuel is well proven, and experience is available in India. The MOX fuel has shown excellent perfonnance in all FBR worldwide, with a high burn-up of 200 gigawatt day/ tonne having been achieved on full size sub-assemblies. Considering all the these aspects, MOX fuel has been chosen for PFBR.Low fuel cycle cost is also important.

Pool Concept

Better safety features of the pool type concept have led to adoption of this concept for PFBR. The high thermal inertia of the large mass of sodium in the pool, containment of all radioactivity in a single vessel with no penetrations for nozzles and adaptability to reliable decay heat removal by independent dedicated sodium loops are the benefits of this choice. In the fast reactor designs, two types of reactor systems have evolved in different countries. U.K., France and Russia have adopted the "Pool" type concepts for large reactors while U.S.A. and Germany, have taken to both "Loop" type and `Pool' type designs. Pool concept is slightly cheaper. Operating experience in other countries confirm the choice.

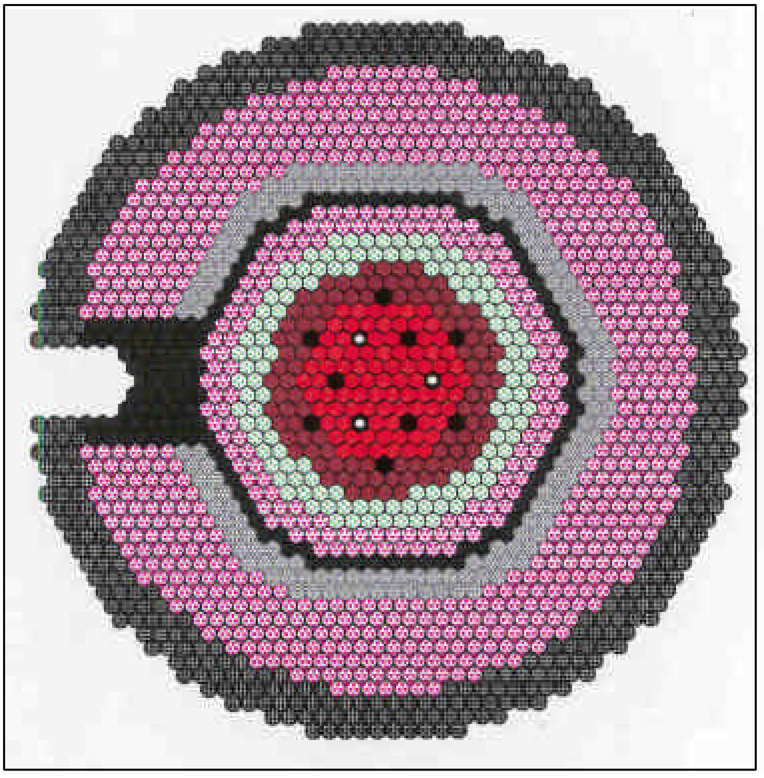
Number of Secondary Sodium Loops

The reduction in number of components helps to reduce the capital cost, construction schedule and outage time due to generic failure of components and also gives a compact layout. A two loop arrangement has been chosen with two primary pumps, four intermediate heat exchangers (IHX), two secondary pumps and eight steam generators (SG). Cooling of the reactor during various design basis events analysis has confirmed the choice.

Table 1: Plant Design Specifications

Reactor coolant	Sodium
Thermal power	1,253 MW
Electrical power (gross)	500 MW
Fuel	PuO_2 - UO_2
Core height	1 m
Fuel pin dia	6.6 mm
Peak fuel burnup	100 GWd/t
Max linear pin power	45 kW/m
Sodium temperature at reactor inlet	670K
Steam condition at TSV	763 K at 16.7 MPa
Reactor shutdown systems	2
Decay heat removal systems	2
Containment building	RCC/rectangular
Reactor life	40 Years

Fig.1 : PFBR Core Configuration



Operating Temperatures and Design Life

Detailed thermal hydraulic and plant dynamic studies and inelastic analysis of hot pool components, like control plug, inner vessel and Intermediate Heat Exchanger (IHX) using advanced material, constitutive models have been done to assess the margins available in the design of these components. Reactor design life of 40 calendar years with capacity factor of 75%, has been considered.

The design temperature chosen are: at the hot pool 820 K, at the cold

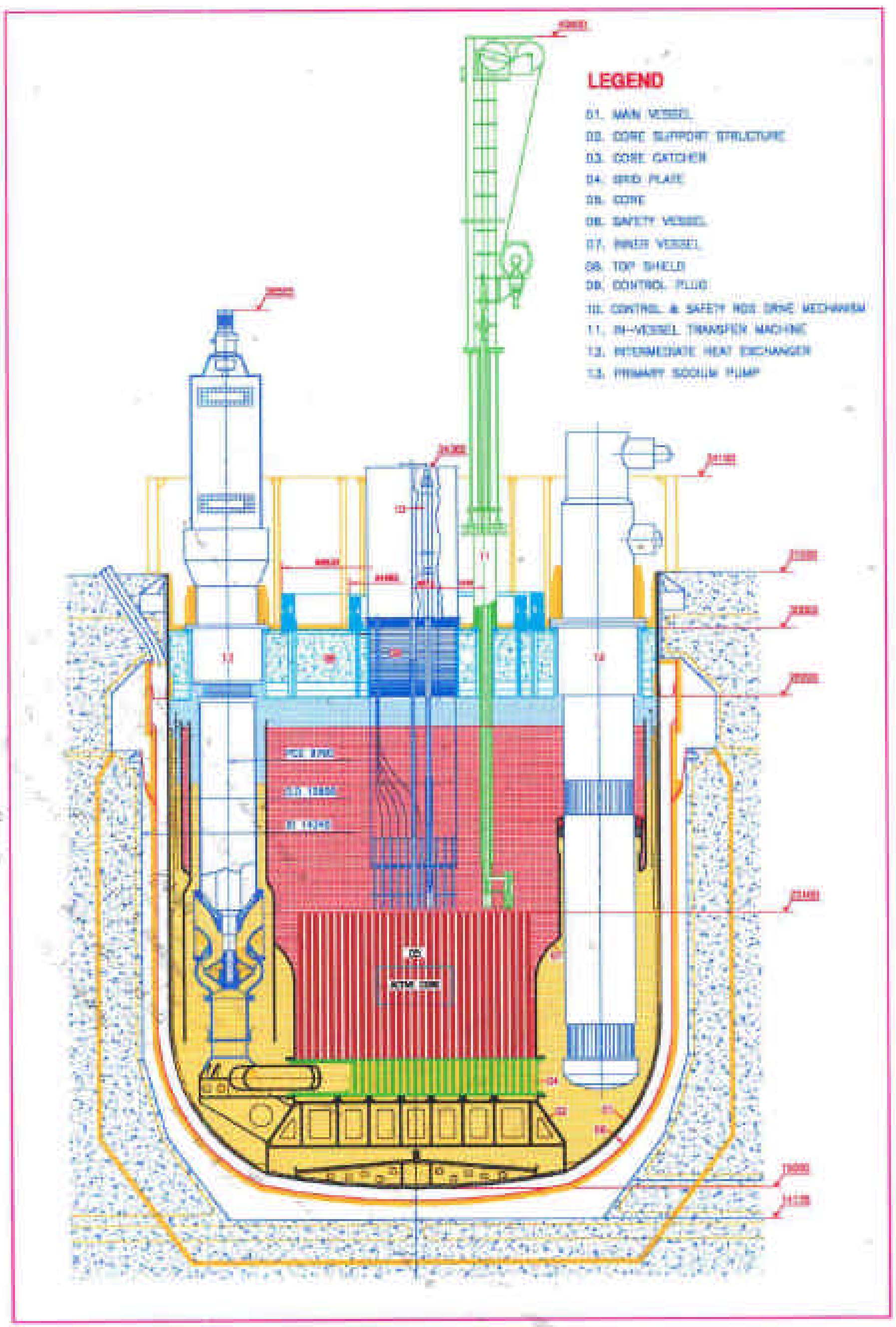


Fig.2: PFBR Reactor Assembly

pool 670 K, at the IHX inlet 628 K, at the IHX outlet 798 K and steam temperature at the turbine inlet 763 K. These are based on optimisation studies and materials limitation. Table 1 summarizes the main design specifications of the reactor.

Nuclear Steam Supply System

Reactor Core

The reactor core is made up of 1,758 subassemblies, arranged in a hexagonal lattice as shown in Fig. 1. Of these, 181 fuel subassemblies (FSA) form the active core. There are two enrichment regions in the active core for power flattening. There are 2 rows of radial blanket subassemblies and 12 absorber rods, comprising 9 control and safety rods (CSR) and 3 diverse safety rods (DSR) arranged in two rings. Enriched boron carbide is used as the absorber material. The radial core shielding is provided by stainless steel and B_4C subassemblies. They limit the secondary sodium activity and radiation damage and activation of the primary circuit components to acceptable levels. Neutronics analyses estimate the isothermal temperature coefficient and power coefficient to be negative at –1.8 pcm/K and –0.64 pcm/MWt respectively, which results in stable operation.

Reactor Assembly

The reactor assembly (Fig.2) refers to all the structural components, which help support the core and the allied mechanisms for heat removal, power control and shut down. It consists of grid plate, core support structure, inner vessel, main vessel, safety vessel, roof slab, rotatable plugs, control plug and absorber rod drive mechanisms.

A large stainless steel vessel, 12.9 m in diameter, called “main vessel”, houses the internals of the reactor. The vessel holds the large pool of liquid sodium weighing about 1,100 t. The main vessel has no penetration and the bottom closure shape is designed to enhance its structural rigidity under core load and sodium pressure. It is supported at the top by welding to the outer shell of the roof slab and is free to expand downward to accommodate thermal expansion.

The grid plate locates the core subassemblies. It also forms the inlet plenum for distribution of coolant flow from the pumps to the core. The grid plate itself is supported by the core support structure, which is welded to the bottom of the main vessel. This structure, together with the inner vessel, acts also as a barrier between the hot and cold pools of sodium.

The reactor assembly has two other vessels besides the main vessel namely the safety vessel and the inner vessel. The main function of the safety vessel is to contain sodium in the event of a leakage from the main vessel, limiting the fall in the sodium level and thus assuring cooling of the core. The gap between the vessels is 300 mm which permits robotic visual and ultrasonic inspection of the vessels.

The main function of the inner vessel is to separate the hot and cold pools of sodium.The lower part of the inner vessel surrounds the core. It has penetrations for the IHX and the primary sodium pumps. To minimize sodium leakage from hot pool to cold pool at the penetration of IHX, a mechanical seal with "piston rings" is used.

The roof slab, which forms the top shield, supports the main vessel, rotatable plugs, control plug, in-vessel fuel handling machine, primary sodium pumps, IHX and heat exchangers of the safety grade decay heat removal system. Air is used for cooling the roof slab. A "warm top shield" concept (T>383 K) has been chosen to avoid deposition of sodium in the annular gaps. Rotatable plugs provide access to all the core subassemblies, which require handling. Concrete is used as the shielding material. Top shield limits the radiation dose in RCB to 25 μSv/h enabling controlled access to RCB.

The control plug supports twelve absorber rod drive mechanisms~~s~~, thermocouples, three failed fuel identification modules and neutron detectors.

Sodium Circuits

Heat Transport Flow Sheet

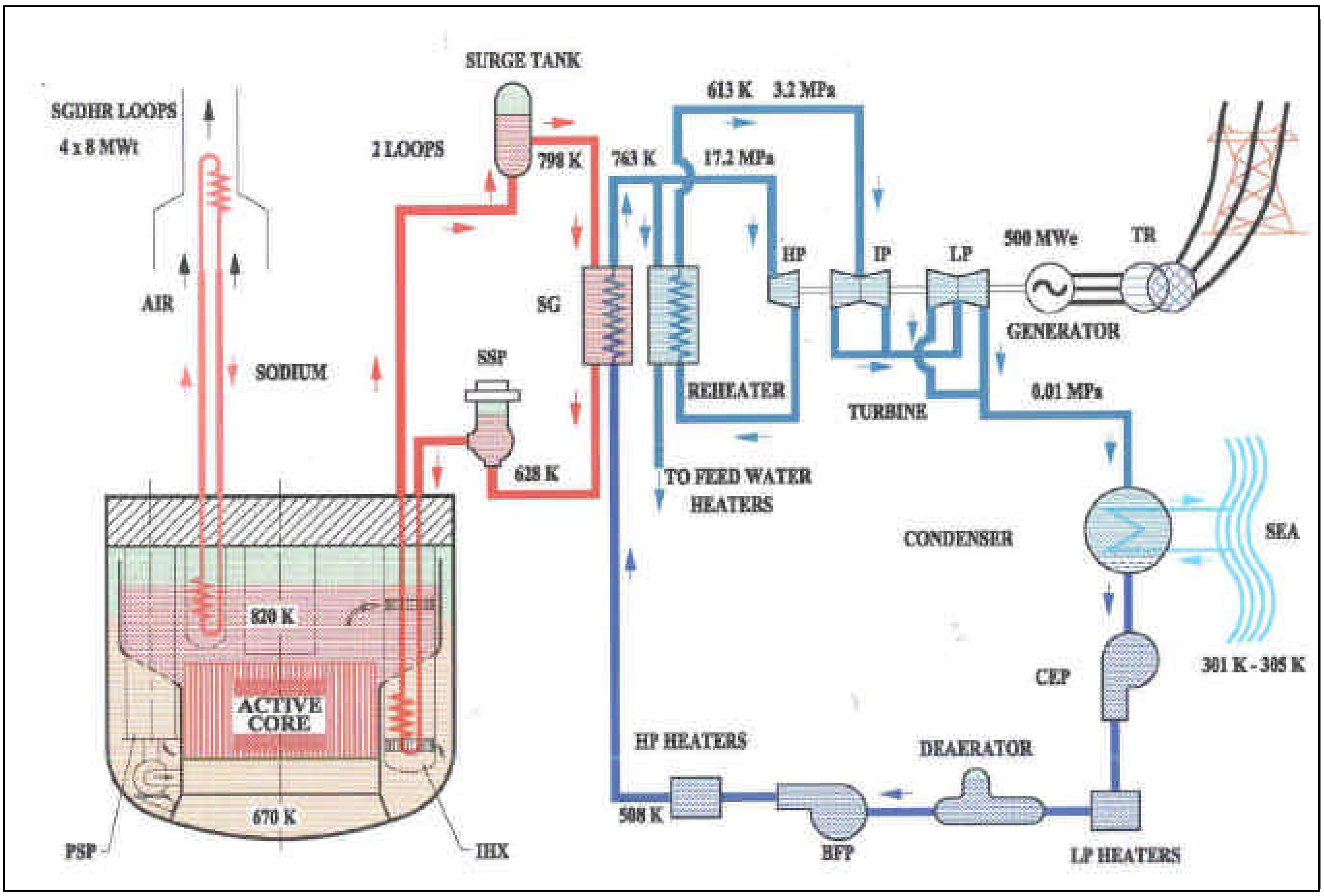
The heat transport system (Fig.3) consists of primary sodium circuit, secondary sodium circuit and steam - water system. Primary heat transport from the core is facilitated by two pumps, which drive sodium from the cold pool through the core. The hot sodium flows through the IHX, transfers its heat to the secondary sodium and finally returns to the cold pool at the bottom, completing the flow circuit.

Use of an intermediate secondary sodium circuit to transfer heat to the steam water circuit prevents the possibility of steam/water leak into the primary system, in the event of a leak in the steam generator tube. This also minimizes the radiation level in the Steam Generator Building, providing better access for maintenance.

Components and Piping

The primary sodium pump is a single stage, top suction centrifugal pump delivering a flow of 4.13 m^3/s at a head of 75 mlc and at an operating speed of 590 rpm. Each pump is powered by a 3.6 MW induction motor and can be operated at a regulated speed that can be varied con-

Fig.3 : Heat Transport Flow Sheet



tinuously between 15-100 % of nominal speed. Each pump is provided with a flywheel ensuring a flow-halving time of 8 s. Emergency power supply and a pony motor supplied by battery are provided to run the pump at 15 % speed in case of power failure. Secondary sodium pump is a single stage, bottom suction, centrifugal pump delivering a flow of 3.34 m^3/s at a head of 65 mlc and at an operating speed of 960 rpm.

Straight tube IHX design with primary sodium on shell side has been selected. Use of relatively small size tubes (19 mm OD) and mechanical seal arrangement around IHX results in reduction in overall size.

As regards steam cycle, reliable turbines are available for both sodium and steam reheat options. Steam reheat has been chosen considering factors like savings in the Steam Generator (SG) capital cost, reduction in construction time and simplicity in design and operation. This outweighs the marginal advantage in cycle efficiency of about 1.3% associated with sodium reheat cycle. A straight tube configuration with an expansion bend at the lower region of sodium flow path has been selected for SG. Tube to tube sheet joint is of internal bore weld type with raised spigot to enhance the reliability of the weld joints. Provision is made for detection of water leaks in the SG by continuous monitoring of hydrogen in sodium and in cover gas and by monitoring the increase in the cover gas pressure of the surge tank.

Sizing of the sodium piping has been decided so as to allow a maximum sodium velocity of 10 m/s, reduced stress at component nozzles, flexibility, and compact piping layout. For secondary sodium main piping, the sizes chosen are 400, 550 and 800 mm dia for different sections. The total inventory of sodium in both the secondary loops is 410 t.

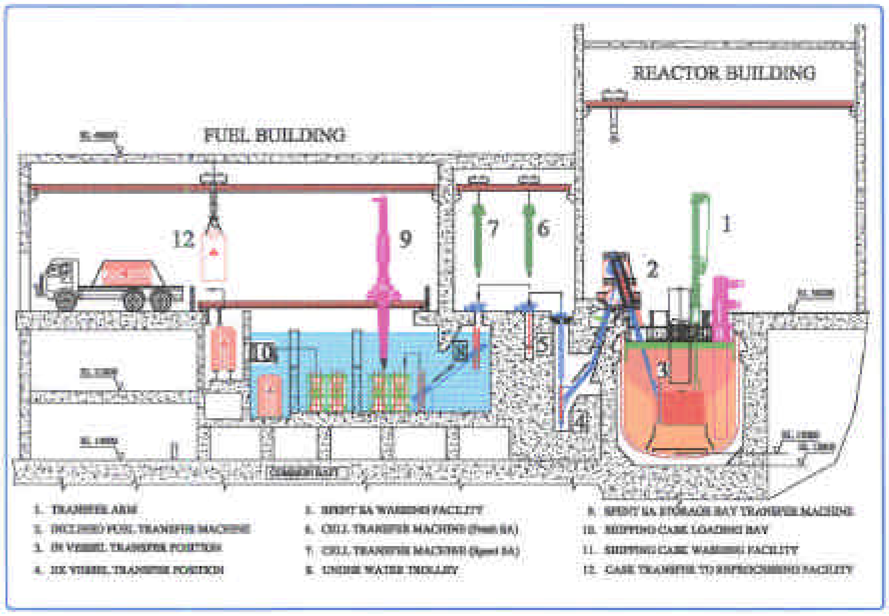


Fig4: Spent Fuel Handling System

Component Handling Systems

The core subassemblies (fuel, absorber, blanket, reflector and shielding SA) are handled at 473 K with the reactor in shutdown condition. Refueling is done at an interval of 185 effective full power days. The handling system (Fig.4) has been divided into two parts i.e. in-vessel handling and ex-vessel handling. In-vessel handling is done by a transfer arm machine, positioned with the help of two rotatable plugs. For ex-vessel fuel handling, an inclined fuel transfer machine with a rotatable shield has been chosen from economic considerations.

The spent fuel subassemblies are cooled in the in-vessel storage locations for 185 effective full power days before being shifted to spent fuel storage bay. The storage is in a pool of demineralized water, which has a capacity to store 710 subassemblies. This includes subassemblies discharged over three campaigns and an emergency discharge of the full core. Sodium sticking to the subassemblies is washed in the fuel transfer cell before depositing them in the pool water.

Special handling of compounds is done by using leak tight, shielded flasks. The 'PI' flask is used to handle large components like primary pump and IHX –and the 'ARDM' flask is used for slender components like the CSRDM and DSRDM. A decontamination facility is provided inside the RCB for the removal of sodium and radioactive products from the primary circuit components before taking them out for maintenance.

Materials of construction for NSSS

Main design consideration is compatibility with liquid sodium and operation at high temperature. For core components additionally resistance to neutron irradiation becomes important. The summary of materials selected is given in Table 2.

Balance of Plant

Steam-Water System

The steam-water system adopts a reheat and regenerative cycle using live steam for reheating. The operating parameters of superheated steam at turbine inlet are 16.7 MPa, 763 K and 1,805 t/h and of reheated steam are 3 MPa, 613 K and 1,706 t/h.

The regenerative feed water heat-

Table 2:Components and Materials of Construction		
Sl. No.	Components	Materials of construction
1	Fuel clad & subassembly wrapper	20% CW D9
2	Main vessel, Safety vessel, Inner vessel, Core support structure, Grid plate, Intermediate Heat Exchanger, Secondary sodium main piping	SS 316 LN
3	Roof slab, Large Rotatable plug, Small Rotatable plug	A48P2 (C.S.)
4	Primary sodium pumps, Secondary sodium pumps, tanks and auxiliary piping	SS 304 LN
5	Steam Generator	Modified 9Cr-1Mo (Grade 91)

ing is done in six stages consisting of three low pressure heaters of surface type, one deaerator of direct contact type and two high pressure heaters of surface type. The turbine exhaust steam is condensed in a surface type condenser by sea water at 305 K.

Turbine bypass capacity of 60 % is provided to facilitate the start-up and the shutdown of the turbine and reloading after a turbine trip.

An operation grade decay heat removal (OGDHR) system consisting of recirculation pumps, and four air-cooled condensers is provided to remove decay heat when off-site power is available.

Condenser circulating water system is used as a once through system for dissipating the heat load of 755 MW into the sea. The system is designed for a temperature rise of 10 K at full power operation.

Electrical Power Systems

Electrical Power system is the source of power for the reactor coolant pumps and other auxiliaries during normal operation and for the protection system and engineered safety systems during normal and accident conditions.

Both Off-site and On-site electrical power supply systems are provided. For the single 500 MWe plant, 220 kV is the transmission voltage. The plant is connected to the Tamil Nadu/Southern Regional Grid to transmit the power generated and also to provide off-site power supply to the station. A 220 kV substation with five transmission lines and double circuit ties to MAPS 220 kV bus are provided.

The standard BHEL make Terbine Generator (TG) with the following specifications is chosen: 500 MWe, 21kV, 3Ph., 50 Hz, 0.85pf, 3000 rpm, water cooled stator winding, hydrogen cooled rotor winding with brushless excitation. The station load being 30 MWe, the net power exported is 470 MWe.

Class IV normal power supply system is the source of power to all the station electrical loads.

Class-III emergency power supply is provided by 4 Diesel Generator (DG) sets each rated for 50% of the total emergency load providing a redundancy of 100%. Each DG is rated for 2.4 MW, 6.6 kV, 3 Ph., 50 Hz with a starting time of 10s. Instrumentation and control loads are fed from Class II 240V, AC, 1Ph., 50 Hz UPS, Class I 220 V DC, 48 DC power supply systems provided. The class I and II power supply systems are backed up by adequately sized lead acid battery banks.

Instrumentation and Control

The important functions of the Instrumentation & Control system are to monitor various plant parameters to guide the operator in all states of the reactor, control them within specified limits and take automatic safety action as necessary. A three level control scheme is adopted viz. in the field, at the local control centres and from the control room. Safety and safety related systems are monitored and operated from control room and they are designed in fail-safe mode. Two diverse hardwired logic systems are proposed for systems connected with reactor safety. Microprocessor based distributed digital control system is envisaged for non-safety related systems.

The reactor is designed to operate as a base load station and the power is controlled manually.

Ten scram parameters from core monitoring systems and heat transport systems have been connected to plant protection system to automatically shutdown the reactor in case of any abnormal event. Neutron detectors monitor the flux and provide signals for safety action on neutron power, period and reactivity. Flow delivered by the sodium pump is measured and safety action is taken based on power to flow ratio. Outlet temperature of the central fuel subassembly is monitored by three fast response chromel-alumel thermocouples in the central canal plug. Two chromel-alumel thermocouples provided at the outlet of each of the other fuel subassemblies monitor local overheating due to plugging of the subassembly, errors in fuel subassembly manufacture or fuel handling errors.

These provisions ensure that there are at least two diverse safety parameters to shutdown the reactor safely for each design basis event which has the potential in crossing its design safety limit.

Failed fuel is detected by monitoring fission product activity in cover gas and delayed neutron counts in the primary coolant.

For early detection of sodium leaks, single walled pipes are provided with wire type leak detectors, double-walled pipes and vessels with spark plug / mutual induction type leak detectors and closed cells and cabins with sodium ionization detectors.

The back-up control room has provisions to monitor systems required for core cooling and bring the reactor to safe shutdown state in the event of the main control room becoming uninhabitable.

Plant Layout

The centerline of Reactor Containment Building (RCB) is approximately 500 m from the RCB centerline of MAPS. A compact layout (Fig.6) with rectangular RCB has been selected based on economics and the ease of construction. The various important safety issues concerning plant and personnel that have been taken into account in finalizing the layout are: provision of a back-up control room, physical separation of the two SG buildings and the four circuits of SGDHR, location of turbine building considering turbine missiles, access for installation and maintenance of components and equipment, zoning and access control regulations required for radiological protection, provision of a common mat from seismic considerations and choice of ground elevation of nuclear island based on Design Basis Flood level.

Reactor Safety

A defence–in-depth philosophy, consisting of three levels of safety, viz., design with adequate safety margin, early detection of abnormal events to prevent accidents and mitigation of consequences of accidents, if any, is adopted. All safety critical systems are designed with adequate redundancy, diversity and independence.

Fig6 : Plant Layout

The engineered safety features include two diverse reactor shutdown systems, decay heat removal system with passive features of natural circulation of intermediate sodium and of air, and diversity in design of sodium to sodium and sodium to air heat exchangers. Core catcher and the reactor containment building are provided as defence in depth for Beyond Design Basis Events (BDBE).

Selection of design features, detailed design analysis and requirements specified for manufacture minimize the risk of sodium leaks from sodium components and piping, and leaks leading to sodium-water reaction in SG. In addition, the design provides for in-service inspection of the main and safety vessels, secondary sodium piping and the steam generators.

Nevertheless, provisions have been made for early detection of sodium leaks from sodium circuits and sodium-water reaction in SG and safety actions to minimize the consequences of the leaks.

Design Approach

Identifying Design Basis Events (DBE) and categorizing them is the first step in carrying out design with adequate margins. Based on the operating experience of FBR and other reactors and following Probabilistic Safety Assessment(PSA) and Deterministic approaches, events are classified. All events with probability of occurrence > 10^{-6}/ Reactor Year (RY) are classified as DBE. Those with probability of occurrence <10^{-6}/ RY are termed as Beyond Design Basis Events (BDBE) and they need be considered only for assessment of

Table 3: Classification of Design Basis Events

Category	Annual Frequency	Examples
1	>1	Normal operation, Planned start up and shutdown
2	$10^{-2}<f<1$	Off-site power failure, pump trip
3	$10^{-4}<f<10^{-2}$	Station blackout, OBE, Pump seizure
4	$10^{-6}<f<10^{-4}$	Primary pipe rupture, SSE

consequences. The DBE are further classified into four categories as shown in Table 3.

Safety classification and seismic categorization of components and systems have been made considering the safety functions they perform. The safety classification is linked to appropriate grades of design and fabrication. This design practice ensures appropriate reliabilities of performance of components and systems. Seismic categorization specifies the level of seismic loads to which components and buildings have to be designed. Category 1 components are designed for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) while category 2 components are designed for OBE only. These practices help in concentrating efforts during design and fabrication on components important to safety while having the advantage of cost effective design.

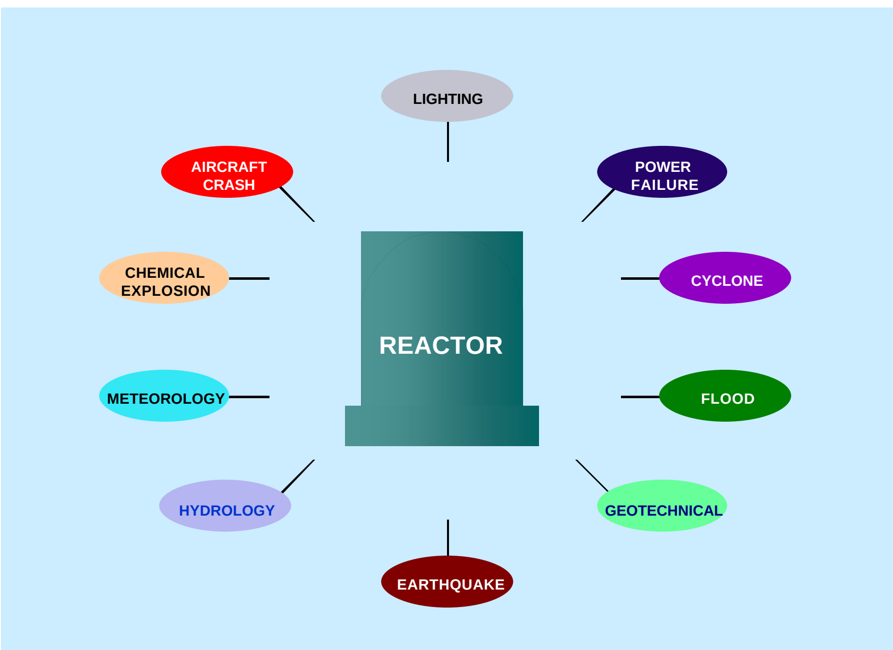
Reactor Core Sodium Void Coefficient

Core void coefficient of reactivity is an important parameter with regard to safety. The whole core net sodium void coefficient is positive and estimated to be 3.7 \$. Detailed calculations show that void coefficient is negative at top regions of core where boiling is likely to commence under temperature transients. Voiding of core is highly improbable because of multiple coolant entries preventing blockage in fuel subassemblies and avoidance of gas entrainment in sodium. Gas release from failed pin is insignificant from void reactivity considerations. Whole core sodium void coefficient is of concern only in the case of hypothetical core disruptive accident (CDA). The magnitude of void coefficient has a bearing on the magnitude of energy release during CDA. Studies have shown that the energy release is not sensitive if void coefficient is kept below 5\$. In view of this, a positive void coefficient of 3.7\$ in the case of PFBR is considered admissible.

Events Analysis

The safety has been evaluated by analyzing the plant response to various external and internal events that may confront the plant during its life.

Fig. 7: External Events



External Events

A large number of external events are considered (Fig.7) including cyclones, floods, earthquakes, lightning and various man-made events. Flooding due to rivers or dam burst does not apply to Kalpakkam site. Similarly aircraft crash on the plant does not apply as the nearest airport is 50 km away and the site does not fall in the air traffic route. Chemical explosions need not be considered as there is no allied industrial activity near the site. Missile impact from turbine is eliminated by proper location of the turbine building with respect to RCB.

The site is situated in a seismically stable region. The peak ground acceleration values for SSE and OBE have been evaluated and these are applied to building and component designs, taking into account the seismic categorization.

The eastern coast of India, on which the Kalpakkam site is situated, is prone to cyclones. Consequently, a detailed analysis has been carried out to arrive at Design Basis Flood Level (DBF). DBF level being 6.45 m above main sea level (MSL), the ground floor level of nuclear island is raised by 2 m to keep it above DBF level.

Internal Events

Internal events are classified as design basis events (DBE) and beyond design basis events (BDBE). The design basis events govern the design of components and systems in order to meet the limits prescribed from radiological safety and structural integrity considerations. The categorization of DBE has been explained earlier. Forty nine events of category 2, 24 events of category 3 and 8 events of category 4 have been considered. Bounding events in each category have been analysed in detail and the compliance with design safety limits is ensured.

Reactor Shutdown System

Two independent, fast acting diverse shut down systems are provided. The diversity is ensured over the entire system covering sensors, logics, drive mechanisms and absorber rods. Each system is designed to limit the failure probability to < 10^{-3}/RY(reactor year). The control and safety rod system is used for reactivity compensation, power control and shut down, while diverse safety rod system is used only for shut down. The control and safety rod system consists of nine rods and their associated mechanisms. The diverse safety rod system consists of three rods and their associated mechanisms. The minimum total reactivity worth of the CSR and of the DSR are 8,000 and 3,000 pcm respectively. The shut down margin provided in the fuel handling state is 5,000 pcm. Any one of the two systems is capable of bringing the reactor to a cold shutdown state, even with one of the absorber rods in the system failing to drop. The systems are designed to effectively shut down the reactor in less than one second.

Decay Heat Removal System

Even after the reactor is shut down, the fission product decay heat needs to be removed to keep the coolant temperatures within limits.

When off-site power is available, the decay heat is removed by the normal heat transport path of primary and secondary sodium and operational grade decay heat removal system incorporated in the steam-water circuit. In case of non-availability of this path, the decay heat is removed by the class I safety grade decay heat removal system, put into operation by opening of dampers on the air side. This passive system consists of four independent circuits of 8 MWt capacity each. Diversity is provided in the design of sodium-to-sodium heat exchanger and sodium-to-air heat exchanger.

Core Catcher

The whole core melt down is a highly improbable event, because of continuous monitoring of power, flow, and temperature. Local melting at the fuel subassembly level is also a highly improbable event, because of the provision of multiple radial entries for the coolant for each subassembly. However, in line with the defence-in-depth philosophy, a core catcher is provided below the grid plate to collect, disperse and safely cool the molten fuel debris from seven subassemblies, in the event of total instantaneous blockage of a single fuel subassembly.

Reactor Containment Building

As a defence-in-depth, a rectangular leak tight reactor containment building of 1 m thick RCC is provided as a final barrier against release of radioactivity in a hypothetical core disruptive accident. It is estimated that the mechanical energy release in the CDA is 20 MJ. However, an energy release of 100 MJ is assumed for evaluating the integrity of the main vessel and the top shield and estimating primary sodium leak to reactor containment building (RCB). For the estimated sodium leak of 350 kg, the consequent sodium fire gives rise to a maximum overpressure of 14 kPa. However, the RCB is designed for 25 kPa overpressure. The estimated dose at the site boundary in this scenario is 29 mSv from cloud γ-dose, 6 mSv from inhalation route and 0.7 mSv from contaminated ground surfaces. The total radiation exposure of 36 mSv is well below the allowable limit of 100 mSv for accident situations.