# Advanced Design Features of MOX Fuelled Future Indian SFRs

P. Puthiyavinayagam, K. Devan, S.R.Aithal, S.Athmalingam, R. Vijayashree, S.Raghupathy, K.Velusamy, N. Theivarajan, S. Usha and A.K. Bhaduri

Indira Gandhi Centre for Atomic Research, Kalpakkam - 603102, India

*E-mail contact of main author: vinayaga@igcar.gov.in*

**Abstract**. India has been operating a Fast Breeder Test Reactor (FBTR) successfully since 1985. Currently, a 500 MWe MOX fuelled pool type Sodium cooled Fast Reactor called Prototype Fast Breeder Reactor (PFBR) is under advanced stage of commissioning. The design, R&D, safety review, construction and commissioning experience from PFBR has motivated the commercial exploitation of MOX fuelled Sodium cooled Fast Reactors (SFR) with closed fuel cycle. Accordingly, six Fast Breeder Reactors (FBR) are planned in which, the first two units (FBR 1&2) will be located at Kalpakkam. These reactors are incorporated with advanced design features towards improved economy and enhanced safety. FBR 1&2 will be of MOX fuelled to be deployed ahead of metal fuelled reactors in order to capitalize on the experience gained in all the domains of SFR technology and to sustain the program. These future reactors need to have improved economy, enhanced safety and possible higher performance parameters. Economy is achieved by design optimization, reduction of material quantities, adoption of twin unit concept with sharing of facilities, design enabling integrated manufacture and erection leading to reduced construction time. Based on the detailed studies, reactor power is enhanced with a slightly larger core and by way of design optimization and exploiting the improved manufacturing technologies, the sizes of major large size components are kept close to the industrial capacity that have been built in the country. This approach has led to raising of reactor power to 600 MWe leading to economic gains. With regard to safety, the important aspects taken into consideration are the internationally evolving Gen-IV safety criteria especially after Fukushima. The enhanced safety level seeks to prevent severe core damage and large radioactivity release to the public and practical elimination of severe accident scenarios involving energy release and public evacuation. The major safety enhancements envisaged are: (i) improved core inherent safety characteristics with sodium void coefficient less than 1 $, (ii) passive shutdown features and additional shutdown systems employing alternative working principles to prevent events leading to accident situations and (iii) passive & augmented decay heat removal capacity. This paper presents the advanced design features envisaged, towards enhancing safety and improving economy in the future MOX fuelled Indian SFRs and other plant system design features.

**Key Words**: Indian FBR program, Advanced design features, FBR1&2, MOX fuelled Indian SFR

## 1. Introduction

India's available energy resources warrant nuclear power as a major part of the energy mix in the long term for meeting the ever increasing energy demands and also due to environmental concerns. Accordingly, Department of Atomic Energy (DAE) has evolved a plan for a challenging nuclear capacity addition. The plan is based on a three stage nuclear power program harnessing the domestic nuclear resources. The emphasis has been on the indigenous technology. Obviously, the target and the role envisaged for nuclear power in India especially through Fast Breeder Reactors (FBR), in the second stage, is large and challenging considering the perspective on growth scenarios and sustainability [1,2].

Indira Gandhi Centre for Atomic Research (IGCAR), instituted in 1971 at Kalpakkam, is involved in the mission of developing the technology of FBR. Multi-disciplinary laboratories are established in the centre around the central facility of the 40 MWt Fast Breeder Test Reactor (FBTR), which has been operational since 1985 and has given valuable experience in the operation of sodium systems including steam generators, and has served as a test bed for various experiments and fuel irradiation program [2,3]. Presently, the indigenously designed MOX fuelled 500 MWe Prototype Fast Breeder Reactor (PFBR) is in an advanced stage of commissioning [3,4].

As per the roadmap of the FBR program, beyond PFBR, six Sodium cooled Fast Reactors (SFR) are planned [3]. These will be MOX fuelled and of 600 MWe capacity which would be deployed ahead of metal fuelled reactors. This is because metal fuel reactors would need lead time for R&D and maturity on fuel cycle technologies and hence they will be deployed subsequent to MOX based FBRs, in the long run. These future reactors need to have improved economy, enhanced safety and possible higher performance parameters. With respect to safety, the general trend is towards enhancing the safety level which seeks to prevent severe core damage and large radioactivity release to the public and practical elimination of severe accident scenarios involving energy release and public evacuation. Hence, safety level is sought to be enhanced to higher levels. This paper presents the advanced design features envisaged in the future MOX fuelled Indian SFRs and other plant system design features.

## 2. Design Objectives

Of the six FBRs planned, two units will be located at Kalpakkam (FBR 1&2) and four units at other site to be identified. These future FBRs will be in the commercial domain. Hence, an important design consideration should be to have improved economy. Further, considering the time frame of the deployment, these future FBRs should possess higher levels of safety in line with the internationally emerging safety standards after Fukushima and Gen-IV safety criteria. In the design evolution, rich experience obtained through the design, R&D, safety review, construction and commissioning activities of PFBR is to be factored into towards deriving the maximum possible improvements. At the same time, it is desirable that the design should not warrant large manufacturing technology development so as to exploit the industrial capability & capacity and other infrastructure that have been established for the PFBR project, both within the department and the country. Hence, it is preferred to continue with the MOX fuel for these six reactors in order to capitalize on the experience gained in all the domains of FBR technology and to sustain the program and to commercially exploit the matured MOX fuelled reactor technology in the initial period. In a nutshell, the approach behind the initial choice of MOX fuel is to establish the FBR technology and to develop the matching industrial and associated capabilities and sustain them. In parallel, metal fuel R&D would be taken up including test irradiation during which technologies will be developed related to metal fuel front end and back end technologies including the pyro reprocessing. This would enable switching over to the metal fuel reactors in an industrial scale smoothly. Another essential design parameter in the Indian context is the need for possible higher breeding ratio and lower doubling time. Since metal fuelled reactors are envisaged for the future, it is preferred to get as high a breeding ratio as possible in the initial MOX fuelled reactors itself.

The major design objectives for FBR1&2 are: (i) Improved economy and higher power output with nearly same reactor assembly size (ii) Enhanced safety aiming Gen-III+ or possible higher safety level (iii) Sodium Void Reactivity < 1 $ : enhanced safety demonstration to practically eliminate severe energetic accidents (iv) Breeding Ratio as high as possible aiming up to 1.2 and optimum fuel inventory (v) Possible higher operating temperatures towards higher efficiency (vi) Optimum number of heat transport systems & components (vii) Maximum utilization of the manufacturing technology established for PFBR (viii) Reduction of specific capital cost & construction time (ix) Incorporation of inherent and/or passive safety features to terminate Severe Accidents and (xi) elimination of the need for offsite public evacuation.

## 3. Design Approach

The approach adopted in the design to achieve the objectives are: (i) Retaining the standard design options which have been validated in the design of PFBR (ii) Optimization & simplification of component design leading to reduction in specific capital cost (iii) Optimum number of primary heat transport systems (iv) Improvement in the design based on current state of art manufacturing technologies (v) Advanced design concepts and features towards economy & safety (vi) Highly reliable engineered safety systems (vii) Incorporation of inherent and/or passive features to terminate Severe Accidents (viii) Use of alternative materials for high performance and economy (ix) Increased burnup (x) Twin units layout with sharing of facilities without compromising safety (xi) Higher thermodynamic efficiency through higher plant operating temperatures (xii) Design for higher plant life (xiii) Design features facilitating reduced construction time and parallel construction (xiv) Reduced fuel cycle cost through higher burnup and lower throughput.

One important choice in the design exercise is that the reactor vessel should not be very much larger than that of PFBR which facilitates reduction in lower specific capital cost through higher power capacity. This will help in avoiding any major manufacturing technology development program for FBR1&2. Another important fact is that the objectives of 'higher breeding' and 'lower sodium void coefficient' are conflicting in nature. Hence, the target sodium void reactivity was chosen as < 1 $ and the breeding ratio to be as high as achievable. Also, it has been reported that with the sum of all reactivity coefficients less than 1 $, the core safety can be sufficiently ensured [5,6] and it is in line with the international approach of having sodium void reactivity as either 'zero' or 'small positive'. Hence, in order to be able to achieve both sodium void coefficient and breeding ratio to a reasonable level, heterogeneous core concept was also kept as a design option in addition to the homogeneous concept.

## 4. Design Features

### 4.1. Reactor Core

Three reference core designs are worked out: homogeneous, radial and axial heterogeneous core [7]. The schematic of the three core options is depicted in Fig.1 and the major design parameters are compared in Table I. Based on detailed assessment of all three concepts, the homogeneous core option is retained by considering the factors such as technology maturity, operational aspects, rich experience etc. Although the breeding ratio is less for this option, the doubling time is only marginally higher.

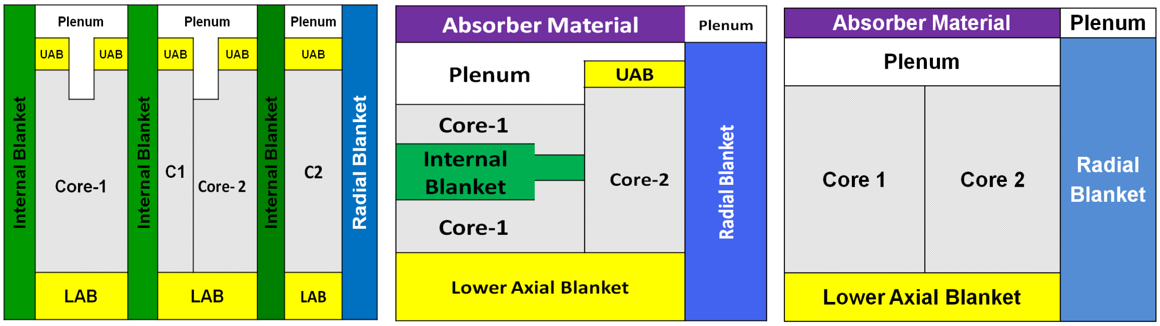

*FIG. 1. (a) Radial Heterogeneous core (b) Axial Heterogeneous core (c) Homogeneous core*

TABLE I: COMPARATIVE DESIGN PARAMETERS FOR THREE CORE DESIGN CONCEPTS

| **Parameter** | | **Radial Heterogeneous** | **Axial Heterogeneous** | **Homogeneous** |
|---|---|---|---|---|
| Core power, MWt | | 1500 | 1500 | 1500 |
| No. of enrichment zones | | 2 | 1 | 2 |
| Pu enrichment (wt %) | | 29/32 | 27.7 | 19.9/26.5 |
| Active core height, mm | | 900/800 | C1:650/700;<br>C2:1000 | 1000 |
| Fuel pin / Spacer wire dia (mm) | | 6.6/1.6 | C1:6.6/1.6; C2: 6.8/1.4 | 6.6/1.6 |
| Axial blankets within fuel pin (mm) | Upper | 100 | C1: 0.0; C2: 100 | No |
| | Lower | 300 | 400 | 400 |
| | Internal | No | C1: 250; C2: 200 | No |
| Peak linear power (W/cm) | | 450 | 410 | 450 |
| Relative fissile inventory | | 1.24X | 1.16 X | X |
| Sodium void coefficient, $ | | 0.9 | 0.9 | 0.85 |
| Breeding ratio | | 1.18 | 1.17 | 1.11 |
| Reactor doubling time, years | | 22 | 22 | 24 |

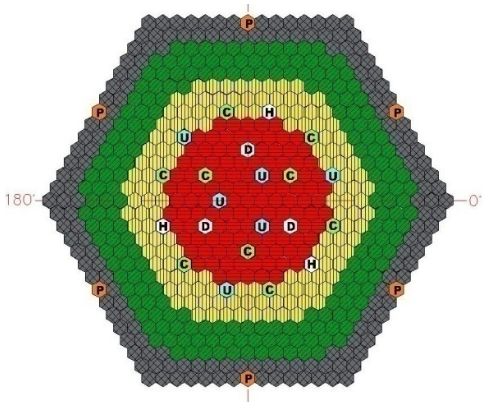

*FIG.2. Core Layout*

The proposed homogeneous FBR1&2 core configuration is shown in Fig.2. It has two enrichment zones of mixed oxide ($PuO_2$-$UO_2$) fuel, followed by blanket and reflector. After in-vessel storage, Ferro-Boron (Fe-B) sub-assemblies are provided as the outer shield. Based on detailed studies, Fe-B is selected as an improved shield material [8,9]. Fe-B has good compatibility with sodium, good neutron shielding properties and lower density compared to steel, resulting in smaller core size and reduced load on core support structure. A weight reduction by 50 % on the outer rows of in-vessel shields is possible with this choice thereby the shielding cost reducing by a factor 5. Optimization studies show that a reduction by one radial row is possible with Fe-B compared to the option with Stainless Steel and $B_4C$ shields. With this, the core diameter up to the radial shielding is worked out to be 6615 mm. The sodium void coefficient is estimated to be 0.9$. Reduction of sodium void coefficient, in comparison with PFBR, is made possible by increasing the neutron leakage and by reducing the power density in the inner core. To enhance the neutron leakage from the core in axial direction, the upper axial blanket is replaced by a fission gas plenum. Breeding ratio is maintained by increased radial blanket thickness and increased axial lower blanket length. In addition, core-2 enrichment is also optimized to improve the breeding potential of radial blanket. Each fuel assembly (FSA) has 217 integrated annular pins (helium bonded) of 6.6 mm diameter (Fig. 3). Increased burnup of about 150 GWd/t is possible based on an analysis by using ferritic steel as the wrapper material from the consideration of controlling the irradiation induced deformations [10].

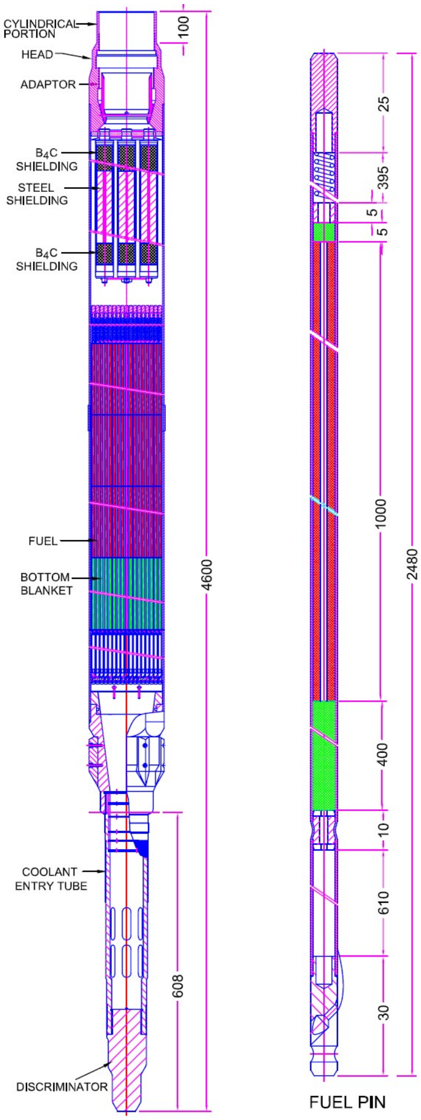

*FIG.3. Fuel Assembly & Pin*

### 4.2. Reactor Assembly

The reactor has three main heat transport circuits, namely primary sodium, secondary sodium and steam-water system. Reactor assembly houses the primary sodium circuit along with core. All the reactor internals are contained in a single vessel called Main Vessel and it is closed with top shield. Based on the experience gained during design, manufacture and erection of PFBR components, detailed and exhaustive design optimisation exercises were initiated. As an outcome of this exercise, several design features and improvements have been considered for incorporation in the design of FBR1&2. For example, some of the design improvements proposed in the reactor assembly design are: (i) Welded grid plate with smaller plenum to accommodate only those sleeves which support core subassemblies through which sodium flows. Support spikes are provided for peripheral shielding subassemblies Study on economic benefits of adopting welded grid plate in place of bolted construction has indicated a net weight reduction of ~55% based on the estimates carried out for a 500 MWe reactor [11] (ii) Inner vessel having toroidal redan with uniform thickness which provides enhanced buckling strength (iii) Dome shaped structure in stainless steel for roof slab which avoids dissimilar metal weld in the critical location and reduces differential thermal movement (radial) between components supported over roof slab and their corresponding standpipe in inner vessel (iv) Redundant support for reactor assembly and core support structure (v) Increased number of primary pipes (from 4 to 6): Increased number of primary sodium pumps and pipes make the consequences of one pump seizure and primary pipe rupture events benign giving rise to the possibility of elimination of a few signals from the list of SCRAM parameters [12]. Notably, the core flow under pipe rupture conditions improves to 42% in comparison with 30 % in the case of PFBR (vi) Simplified fuel handling scheme with elimination of inclined fuel transfer machine and sharing of components in twin units which leads to 44% cost savings (component handling aspects are not covered in this paper as it is covered in a companion paper [13]) (vii) Incorporation of Post Accident Heat Removal System along with the safety vessel as an additional passive safety feature to enhance the decay heat removal capability (with a maximum heat removal capacity in

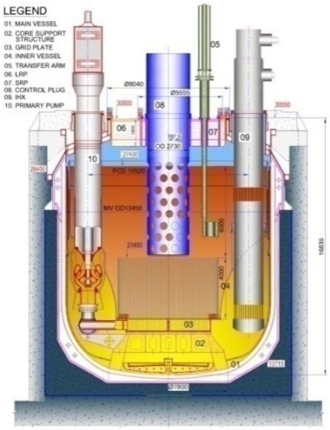

*FIG. 4. Reactor Assembly*

the order of 10 MW) under an unlikely event of weakened SGDHR system following postulated HCDA (viii) Core catcher designed to hold the core debris resulting from meltdown of full core and having the provision of sacrificial layer.

Further, it is observed that vessel dimensions contribute to the overall capital cost. With this as an objective, the sizing of reactor assembly has been optimised by adopting following measures: (i) Machined penetrations in roof slab which will reduce the annular gaps between penetrations and the corresponding component passing through it (ii) Innovative clamping arrangement for IHX which results in reduced support flange width for supporting the component (iii) Optimizing the width of IHX-IV seal (iv) Dimension control of annular gap between inner vessel-inner baffle, inner baffle-outer baffle and outer baffle-main vessel which contributes in reducing the annular gap width. For the proposed homogenous core configuration (Fig. 2) the overall diameter of reactor vessel is worked out to be ~13.5 m after implementing the above listed sizing optimization measures and the reduction in specific raw material requirements is expected to be in the order of 11% for reactor assembly [10]. The reactor assembly vertical sectional view and the top shield layout are shown in Fig.4 and Fig.5 respectively.

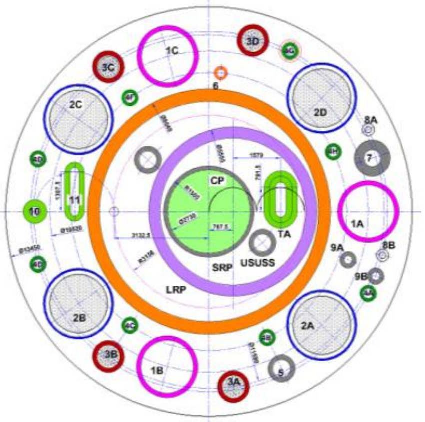

FIG. 5. Reactor Top Layout

## 4.3. Sodium Heat Transport Systems

The main components in the primary sodium circuit are primary sodium pump and intermediate heat exchangers. The main components in the secondary sodium circuit are secondary sodium pump, steam generator and associated piping. Considering the economy, safety and experience gained from design, manufacturing and testing of sodium circuit components of PFBR, the configuration of sodium heat transport system for FBR1&2 is arrived at. Primary sodium system is designed with 3 primary sodium pumps (as compared with 2 pumps in PFBR) and 4 Intermediate heat exchangers. The selection of 3 Primary sodium pumps is based on manufacturing feedback from PFBR. Secondary sodium system consists of 2 loops and each loop consist of 1 secondary sodium pump, 1 surge tank, 2 intermediate heat exchangers and 3 Steam Generators. The steam generator is conceptualised with 30 m long tubes in order to enhance the equipment safety by reducing the total number of tube - tube sheet joints by 2188 from 8752 and also to reduce the manufacturing schedule (Fig.6). With this, there is 25 % reduction in the rate of weld failure [10].

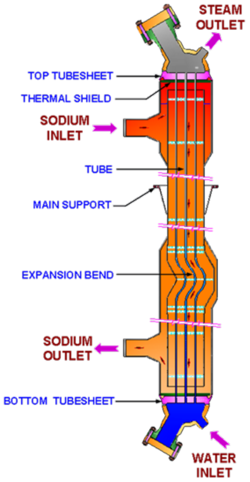

FIG. 6. Steam Generator

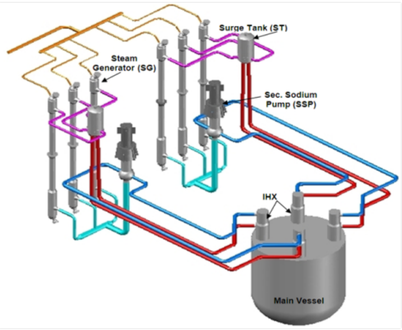

FIG. 7. Secondary Sodium Main Circuit

The pipeline connecting the secondary sodium pump and each intermediate heat exchanger and also the pipe connecting intermediate heat exchanger and surge tank are maintained same as that in PFBR (550 mm diameter), pipe line connecting surge tank and each steam generator and pipe connecting each steam generator outlet and header are sized as 500 mm diameter (as compared with 400mm in PFBR) and the header and header to pump are maintained same as that in PFBR (800 mm diameter). The pipes are sized considering the limit on sodium velocity in sodium pipes, economy and minimum sodium inventory. The pipe thickness is arrived based on the design conditions and the nearest plate thickness is used for rolling to the required pipe size instead of choosing from standard pipe schedule. This helps in reducing the steel consumption for piping. The sodium purification systems are maintained similar to that in PFBR. The components like primary sodium pump, intermediate heat exchanger, steam generator and secondary sodium pump are conceptualised considering the economy, experience gained in PFBR and industrial manufacturing capability. Secondary sodium main circuit is shown in Fig.7.

### 4.4. Shutdown Systems

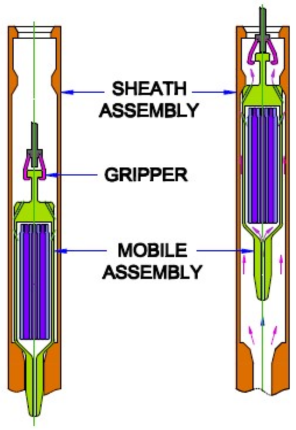

*FIG. 8. Hydraulic Suspended Absorber Rod*

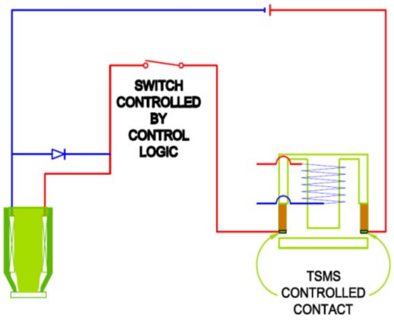

*FIG. 9. Temp sensitive magnetic switch in series with DSRDM EM*

Safety design criteria are evolving continuously internationally. Some of them are IAEA SSR-2/1 [14], WENRA [15] and GEN IV [16]. Accordingly safety design criteria for FBR 1&2 has been evolved [17]. For new reactor designs, several scenarios that are considered as beyond design basis for previous generation reactors needs to be addressed under Design Extension Condition (DEC). The postulated DECs for SFR are the three major Anticipated Transients Without Scram (ATWS) events namely Unprotected Transient Over Power (UTOP), Unprotected Loss of Flow (ULOF), and Unprotected Loss of Heat Sink (ULOHS). The probability for an ATWS events is reported to be in the range of $10^{-7}$-$10^{-8}$ per reactor year [18]. The following are the important decisions related to shut down systems to facilitate practical elimination of core degradation due to ATWS (i) Strengthen the first two shutdown systems by addition of passive/active features and (ii) Introduce an additional shutdown system which is completely diverse, independent, passive & confined within core subassembly. This shall come into action on failure of first two systems.

The first shutdown system (SDS-1) comprises of nine Control and Safety Rods (CSR) and three Hydraulically Suspended Absorber Rods (HSARs). CSRs are used for both power control as well as shutdown whereas HSARs are used exclusively for shutdown. The Scram signals of first shutdown system trigger both CSR and HSAR. In addition, HSARs get triggered by passive means due to flow reduction through core and ensure shutdown of the reactor (Fig.8). A Stroke Limiting Device is provided in CSRDM to prevent inadvertent withdrawal of CSR [19]. Studies carried out for PFBR show that by limiting the stroke length of CSR by 25 mm, the consequences of inadvertent withdrawal of CSR is contained well within the design safety limits specified for category 2 events even with the failure of other safety actions. The second shutdown system (SDS-2) has three Diverse Safety Rods (DSRs). DSR have the function of reactor shutdown only. The temperature sensitive magnetic switch based on Curie point is introduced in series with the power supply circuit of electromagnet of Diverse Shutdown Rod Drive Mechanism as an additional passive safety feature (Fig.9) [20].

Apart from this, an ultimate shutdown system which is fully passive and gets activated due to temperature rise of coolant inside the core is also envisaged. It may be noted that provision of ultimate shutdown system even with a failure probability of less than $10^{-2}$ per reactor year will ensure a failure probability $< 10^{-7}$ for shutdown per reactor year. The ultimate shutdown system would be either $B_4C$ granule based or liquid absorber based. These design features envisaged for the shutdown system render the UTOP, ULOF and ULOHS events into the DEC category.

### 4.5. Steam Water System

FBR-1&2 is planned as a 2x600 MWe plant from considerations of availability of proven turbo-generator sets and lower specific cost (cost per MWe). With the 600 MWe option, the specific cost gets reduced by about 11% in comparison with PFBR. The steam parameters at the turbine inlet are expected to be 170 $kg/cm^2$ (a) and 763 K, the same as in PFBR. The steam cycle is similar to that of PFBR. Turbine for 600 MWe is typically a 4 cylinder machine i.e. 1 HP, 1 IP and 2 LP turbines connected to 2 condensers, unlike a single LP and a condenser in case of PFBR. The sea water intake, outfall, sea water pump house, DM plant and auxiliary boiler are shared for the twin units.

### 4.6. Electrical Power Systems

The power systems in the plant are arranged into Offsite and Onsite power systems. Except the switchyard, power systems are independent for each unit. Details of onsite power systems are briefly presented here.

***Onsite Class IV and Class III Power Systems:*** The onsite power system is arranged into non safety related Class IV power system and safety related Class III and Class I& II systems. Class IV power system receive power supply from the grid through Unit Auxiliary Transformers (UAT) / Station Transformers (ST). Two independent Class III systems each arranged in two sections with its own Emergency Diesel generators (EDGs) are provided with overall 4 EDGs. To meet the plant monitoring of safe shutdown conditions, two SBO DGs are provided. They will be located at elevation higher than EDGs. They will sustain select loads beyond SBO duration. To enhance Class III power supply reliability, alternate AC power supply through a combination of an additional grid connection to the plant and a Gas Turbine Generator is proposed.

**On *Site Class I & II No Break Power Systems:*** A 4 Train 48V DC System to meet I & C Safety Class -1 loads is proposed. Each train will have independent chargers and batteries. Standby charger common for two trains is also provided with battery. For other I&C loads, a 4 Train integrated 220V DC and 415V AC Systems without compromising safety requirement is proposed. Parallel redundant inverters in each UPS Train is followed. Standby charger common for two trains is also provided with battery.

### 4.7. Plant Temperatures

The major plant temperatures are finalised through detailed studies. Though the studies indicated the feasibility of raising the core outlet temperature and temperature rise across the core and also higher steam temperatures were possible with the available turbo generators, it is decided to retain the same temperatures as in PFBR. Based on the experience, the parameters could be revised later on. The major parameters are presented in Table II. Thermo mechanical studies have indicated that average core outlet temperature of 830 K with associated higher steam temperature is a possible option [10]. However, at this temperature, the life of IHX gets limited to 30 years due to Creep-Fatigue damage of top tube-sheet. Though IHX is a replaceable component, considering the involved efforts, economic penalty and impacts on availability of plant in addition to radioactive waste inventory to be stored, core outlet temperature is retained as 820 K.

TABLE II : PLANT TEMPEARTURES

| Parameter | Value |
|---|---|
| Core inlet ($T_{RI}$) | 670 K |
| Core outlet ($T_{RO}$) | 820 K |
| Primary sodium inlet to IHX | 817 K |
| Primary sodium outlet from IHX | 667 K |
| Secondary sodium inlet to IHX | 628 K |
| Secondary sodium outlet from IHX | 798 K |
| Feedwater inlet to SG ($T_W$) | 509 K |
| Steam outlet from SG ($T_S$) | 766 K |
| Steam temperature at turbine inlet | 763 K |
| Steam pressure at SG outlet | 178.5 kgf/cm$^2$ |
| Steam pressure at turbine inlet | 170 kgf/cm$^2$ |

### 4.8. Safety Measures

Various initiating events that can result in an accident are identified, classified as design basis and beyond design basis events and analyzed to evaluate their consequences and to provide suitable safety measures. Beyond design basis events are further classified as design extension conditions (DEC) and practically eliminated conditions (PEC). Radiation release to public domain is eliminated under DEC. Probability of occurrence of PEC are achieved to be very low by stringent design measures. Nevertheless, severe accident management guidelines are developed for handling PEC situations. Thus, all the aspects of safety are taken care in the design. Safety design measures adopted for DEC and PEC conditions are listed in Table III.

## 5. Summary

India is developing the design of a Sodium cooled Fast Reactor with advanced design features towards achieving economy and higher levels of safety. It encompasses the rich experience gained from the design, safety review, component manufacture, erection and commissioning exercise from PFBR. The power is upgraded to 600 MWe leading to reduction in specific material weight, which is expected to be in the order of 7-8% for NSSS in total. It would be deployed as a twin unit resulting in economic advantages. The design envisages to meet the safety criteria being evolved internationally after Fukushima, while at the same time improving the breeding potential, which is an essential design requirement for

India from growth consideration. This design will be adopted as the standard for the future Indian FBRs.

TABLE III : SAFETY DESIGN MEASURES

| Event | Systems | Remarks |
|---|---|---|
| ***ATWS*** | | |
| UTOP | Stroke limiting device, Temperature Sensitive Magnetic Switch | Passive feature |
| ULOF | Hydraulically suspended absorber rod | Passive feature / Passive system |
| ULOHS | Ultimate shutdown system | Passive feature |
| Core Support | Redundant shell/ISI, inclination monitoring for core/grid plate | Design measures and ISI |
| ***LOHRS*** | | |
| LORL | Small inter-vessel space, DHX inlet window length and level optimization, SV design to withstand mechanical loads from earthquakes whilst retaining leaked sodium for a long time, Separate support structures for RV and SV to the extent practicable,<br>Sufficient margins against earthquakes. | Design measures and ISI |
| LOHS | Additional DHR system incorporated in the safety vessel with air as ultimate heat sink | Natural convection based circuit |
| *Energetic CDA* | Ultimate shutdown system based on $B_4C$ granules. Reduced positive coolant void coefficient. | Avoiding energetic CDA |
| *SAM* | Core catcher with features for prevention of recriticality for credible core debris and PAHR, decay heat removal directly from vessel, containment to limit the radioactivity release. | Avoiding re-criticality and ensuring PAHR |

## 6. References

[1] ANIL KAKODKAR, “Evolving Indian Nuclear Programme – Rationale and Perspective”, Public lecture at Indian Academy of Science, Bangalore, India, July 4, 2008.

[2] PUTHIYAVINAYAGAM, P., Indian Nuclear Power Program with Focus on Fast Breeder Reactors, FJOH Summer School on Nuclear Reactor Physics, Fuels and Systems- Tomorrow's Liquid Metal Fast Reactors: Towards Improved Safety and Performance, Aix-en-Provence, France, Aug 24 -Sep 2, 2016.

[3] PUTHIYAVINAYAGAM, P., 2015, Progress in Fast Reactor Programme of India: April 2014-March 2015, 48th Annual Meeting of TWGFR, IAEA, held in IPPE, Obninsk, Russian Federation, May 25-29, 2015.

[4] VASUDEVA RAO, P.R., SATYA MURTY, S.A.V., PUTHIYAVINAYAGAM, P. and RAVISANKAR, A., Current Status of the Indian Programme on Fast Reactors with Closed Fuel Cycle, ANUP 2016, October 24-27, 2016, Sendai, Japan.

[5] PASCAL ANZIEU, et al., Practical elimination, DEC and sodium voiding safety criteria for the ASTRID project, Joint IAEA-GIF SFR Safety Workshop, Vienna, 10-11 JUNE 2014.

[6] YASUSHI OKANO, Reactor Core, Reactivity Characteristics and Sodium Void Effect in relation to SFR Safety, Joint IAEA-GIF SFR Safety Workshop, Vienna, 10-11 JUNE 2014.

[7] ABHITAB BACHCHAN, et al., Conceptual Design of Future Fast Reactor Core with Enhanced Safety and High Breeding, Proceedings of the 20th National Symposium on Radiation Physics (NSRP-20), RPNI-2, 535, Mangalore University, Mangalore, India.

[8] SUNIL KUMAR, D., A Feasibility Study of Ferro-boron as in-core Shield Material in Fast Breeder Reactors, Nuclear Engineering and Design, 240, 2972, 2010.

[9] KESHAVAMURTHY R S et al., Experimental and theoretical investigations of Ferro-Boron as in-vessel shield material in FBRs, FR-13 Paris, 2013.

[10] Conceptual Design Report of 600 MWe Commercial Breeder Reactor, Internal Document: No. CBR/CDR/Rev-A/February 2016.

[11] CHELLAPANDI P. et al., Design Concepts for Reactor Assembly Components of 500 MWe future SFRs, Nuclear Engineering and Design, 240, 2948-2956 (2010).

[12] AMIT K. CHAUHAN et al., Safety benefits of increasing number of primary coolant pumps and pipes in future FBR, 6th International and 43rd National Conference on Fluid Mechanics and Fluid Power, 2016, MNNITA, Allahabad, India.

[13] RAGHUPATHY, S., et al., Component handling system: PFBR and beyond, FR-17 (IAEA-CN-245-314), Yekaterinburg, Russian Federation, 26 – 29 June 2017.

[14] Safety of Nuclear Power Plants: Design Specific Safety Requirements, “IAEA safety standards series No. SSR-2/1”, 2007.

[15] Western Union Nuclear regulators Association, Report on Safety of new NPP designs, Study by Reactor Harmonization Working Group RHWG, March 2013.

[16] IAEA-GIF, SDC-TF/2013/01: Safety design criteria for Gen IV sodium cooled fast reactor, May 2013.

[17] Safety Criteria and Guidelines for Design of Fast Reactors, IGCAR Internal Document, 2015.

[18] LUCIANO BURGAZZI, “Analysis of solutions of passively activated safety shutdown devices for SFR”, Nuclear Engineering Design 260 (2013) 47-53.

[19] SURESH, L., et al., Design and Development of Stroke Limiting Device for Control & Safety Rod Drive Mechanisms (CSRDMs) of future FBRs Paper No: IAEA- CN-245-327, FR17, Yekaterinburg, Russian Federation, 26 – 29 June 2017.

[20] S. PRASANTH KUMAR, et al., Convective thermal analysis of a temperature sensitive magnetic switch, XVII National & VI ISHMT/ASME Heat and Mass transfer Conference, IGCAR, Kalpakkam, 2004, pp.381-385.