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Fast reactor programme in India

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Abstract. Role of fast breeder reactor (FBR) in the Indian context has been discussed with appropriate justification. The FBR programme since 1985 till 2030 is highlighted focussing on the current status and future direction of fast breeder test reactor (FBTR), prototype fast breeder reactor (PFBR) and FBR-1 and 2. Design and technological challenges of PFBR and design and safety targets with means to achieve the same are the major highlights of this paper.

Keywords. Sodium fast reactor; design challenges; construction challenges; emerging safety criteria; passive shutdown and decay heat removal systems; fast breeder reactors in India.

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1. Introduction

India has limited uranium and abundant thorium resources. The uranium resources reasonably assured plus inferred in India is about 160,000 t (<2% of the world resource). However, the thorium resource in the country is 225,000 t (second largest reserve in the world), which has an energy potential 155,000 GWe-y. To ensure energy security with the available resources, Department of Atomic Energy (DAE) follows the 'Three Stage Nuclear Power Programme', formulated by Dr Homi Bhabha. This programme involves water reactors, mainly pressurized heavy water reactors (PHWRs) to extract ∼10 GWe capacity for about 50 years in the first stage; fast breeder reactors (FBR) to extract 275 GWe for about 550 years in the second stage; and thorium-based reactors beyond 2050 in the third stage. The current share from water reactors is 4780 MWe from 20 reactors. The nuclear share including the imported light water reactors (LWRs) under safeguards up to 40 GWe, would be 63 GWe by 2032, ∼220 GWe in 2052 and ∼600 GWe in 2090 (figure 1).

FBRs can extract more than 80 times the thermal energy from the same quantity of uranium and generate electricity with higher thermal efficiency (40%), the available uranium can be utilized very effectively. The thorium-based reactors require a large quantity of

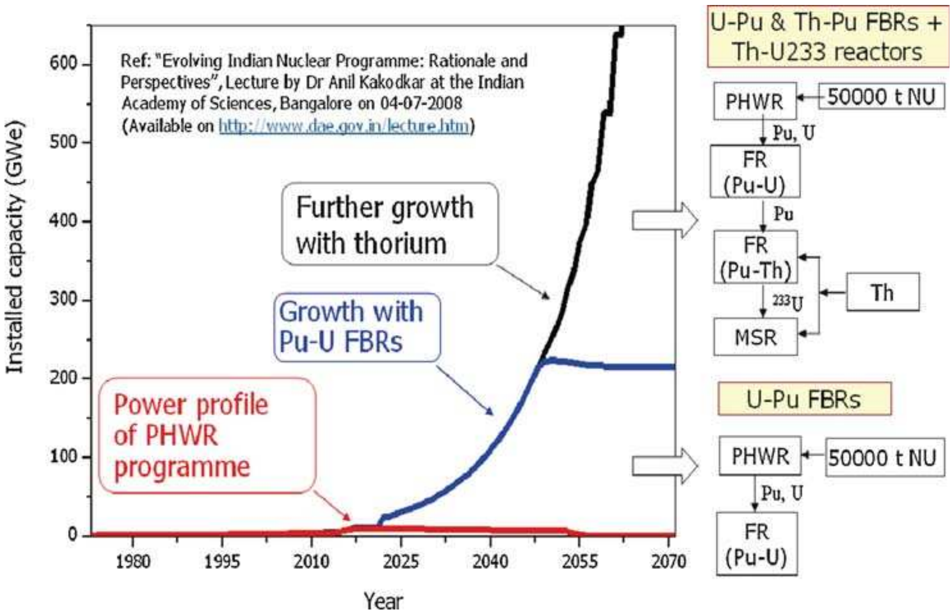

Figure 1. The Indian nuclear power growth scenario.

fissile materials for the initial feed as well as for a few subsequent cycles, which has to come from FBRs. Hence, a large component of nuclear power capacity expansion has to be based on FBRs with closed fuel cycle.

FBRs can burn the long-lived radioactive elements or nuclides by converting them to short-lived elements. FBRs built beyond 2050 would have the capability of burning minor actinides so that the radiotoxicity of the waste would be insignificant (natural background level) within about 300 years. Thus, FBRs can play important roles even in the third stage to realize the abundant energy potential of thorium. Hence, the second stage through FBRs is considered as 'essential' for the Indian Nuclear Power Programme. This paper highlights the Indian Fast Reactor Programme upto 2030 focussing on the status of FBTR, PFBR and FBR-1 and 2.

2. Indian fast reactor programme: Highlights

India started the FBR programme by constructing a 40 MWt/13.5 MWe loop-type fast breeder test reactor (FBTR) at Kalpakkam. FBTR is a sodium-cooled experimental reactor and was commissioned in 1985 with a unique plutonium-rich carbide fuel. This was followed by the design and development of 500 MWe capacity prototype fast breeder reactor (PFBR). The main objective of PFBR is to demonstrate the techno-economic viability of sodium fast reactor (SFR) for the commercial deployment in series. Further, demonstration of comprehensive closed fuel cycle technologies such as fuel fabrication, reprocessing, waste management and waste immobilization can also be achieved through the successful operation of PFBR. Apart from these, PFBR and associated fuel cycle facilities will provide validation of several first-of-a-kind design concepts and standardization for the adoption in future reactors, obtaining experience on operation and maintenance of mechanisms, components and sensors for the integrated operation in sodium at relevant reactor temperatures (820 K at steady state). The design and development activities for PFBR were carried out at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, a mission-oriented centre dedicated to perform comprehensive R&D on fast reactor technology. Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI), a government company, formed in 2003 is implementing the project. The construction of the reactor was started in 2003. The safety clearance for the design and construction was accorded by the Atomic Energy Regulatory Board (AERB). PFBR will attain its first criticality in the second half of 2015.

Roadmap of SFR development beyond PFBR has been formulated in the first phase up to 2030 (figure 2) and it is under discussion at various levels. In this roadmap, two FBRs have been conceived to be commissioned by 2023–24 with several features to achieve economic competitiveness (with respect to 700 MWe-capacity Indian PHWR) and enhanced safety (similar to GEN-IV targets) towards exploiting the matured MOX fuelled SFRs with closed fuel cycle. Economic optimization study prefers to opt for 600 MWe for both FBR-1 and FBR-2. These two FBRs will be built as one twin unit, i.e., 2×600 MWe plants sharing several non-safety related facilities.

In view of their high breeding potential, metallic fuelled FBRs are the focus in the development of FBRs beyond 2030. To realize such scenarios, a metallic fuelled test

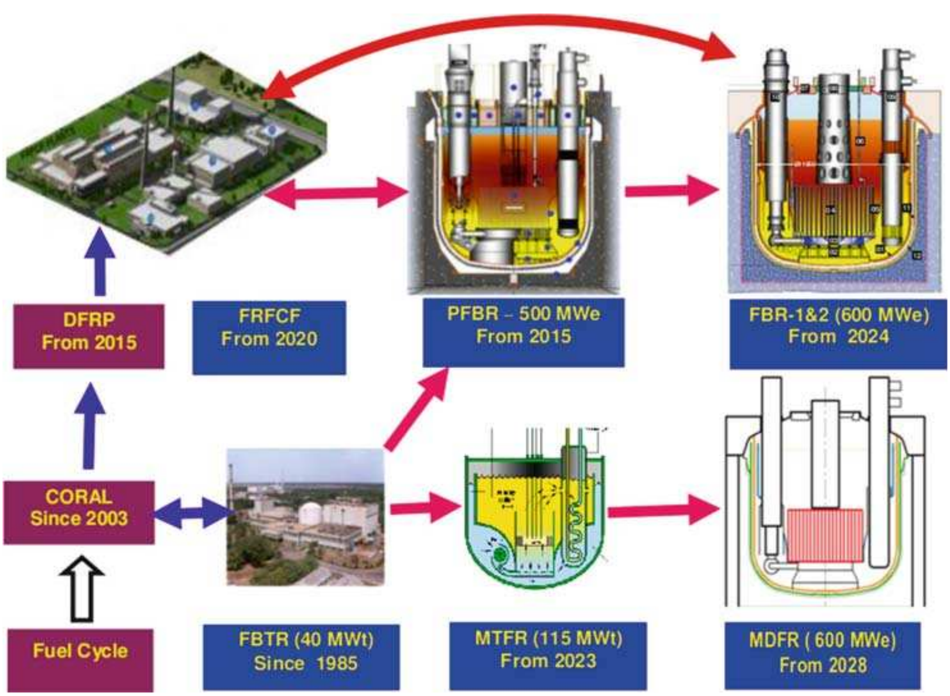

Figure 2. Fast reactor and associated fuel cycle programme in India 2030.

reactor (MFTR) of about 115 MWt capacity is planned to demonstrate the associated fuel cycle technologies through pyroprocessing route. After attaining adequate confidence on the performance of MFTR, a demonstration plant of 600 MWe capacity called metallic demonstration fast reactor (MDFR), would be constructed and commissioned.

In the domain of fast reactor fuel cycle, reprocessing of high burn-up carbide fuel has been well demonstrated with CORAL facility. For reprocessing FBTR fuel on a regular basis and also for reprocessing the annual discharge of MOX fuel from PFBR, a demonstration fast reactor fuel reprocessing plant (DFRP) was conceived several years ago and now the plant is getting ready for commissioning by 2015. In parallel, a fast reactor fuel cycle facility (FRFCF) was planned to close the fuel cycle of PFBR. The financial sanction was obtained in 2013 and the plant is under construction, near the PFBR site. The target date of commissioning is sometime during 2019. FRFCF was conceived to cater to the needs of three FBRs: PFBR and FBR-1 and 2.

3. FBTR: Current status and future programme

FBTR is a sodium-cooled, loop-type fast reactor, fuelled with a unique high Pu mixed carbide fuel. It has two primary and two secondary sodium loops. The first criticality was achieved in October 1985 with a small core of 22 fuel subassemblies (SA) of MK-I composition (70% PuC–30% UC), with a design power of 10.6 MWt and peak linear heat rate (LHR) of 250 W/cm. Progressively, the core was expanded by adding SAs at peripheral locations. Carbide fuel of MK-II composition (55% PuC – 45% UC) was inducted in the peripheral locations in 1996. Turbogenerator (TG) was synchronized to the grid for the first time in July 1997. Linear heat rate (LHR) of MK-I fuel was increased to 400 W/cm in 2002. Eight high Pu MOX fuel SA (44% PuO_2) represented for PFBR were loaded in the core periphery in 2006. The reactor has so far been operated up to a power level of ∼25 MWt. Twenty two irradiation campaigns have been completed till date.

The unique carbide fuel has set an international record in burn-up (165 GWd/t). The fuel discharged at 155,000 MWd/t has been successfully reprocessed and refabricated. This is the first time in the world that the plutonium-rich carbide fuel has attained such a high burn-up and has been reprocessed successfully. The test capsules with MOX (53% PuO_2) represented for PFBR sphere-pac pins were irradiated for 300 h with a peak linear power ranging from 205 to 260 W/ cm and the burn-up achieved is 1300 MWd/t. Towards designing and building future metallic fuelled test reactors, irradiation of metallic fuel pins is in progress. Nine metallic fuel pins with natural U + 6 wt% Zr, enriched U (15% U-235)−6 wt% Zr and natural U−19 wt% Pu−6 wt% Zr are being irradiated currently. The claddings of these pins have 6.6 mm outer diameter, 0.45 mm thickness and 531.5 mm length. This apart, FBTR is utilized for radioisotope production. Notably, a programme to produce Sr-89 from yttria is in progress.

In the near future, it is planned to irradiate mechanically bonded fuels with a composition of LEU–15 wt% Pu in a double cladded tube of 9Cr–1Mo/Zr–4 as well as three sodium bonded fuel pins in a capsule with ternary fuel (19.75% U-235 + 23.5 wt% Pu + 6 wt% Zr). For continuing the irradiation programme for a long period, the reactor life has been extended for further operation of about seven full power years. With this, the reactor is expected to operate at least up to 2025.

4. Prototype fast breeder reactor

4.1 *Technical details*

PFBR is a 500 MWe capacity sodium-cooled pool-type reactor. The overall flow diagram comprising a primary circuit housed in reactor assembly, secondary sodium circuit and energy conversion system is shown in figure 3. The nuclear heat generated in the core (1250 MWt) is removed by circulating sodium from the cold pool at 670 K to the hot pool at 820 K. The sodium from the hot pool after transporting its heat to four intermediate heat exchangers (IHX) mixes with the cold pool. The circulation of sodium from cold pool to hot pool is maintained by two primary sodium pumps and the flow of sodium through IHX is driven by a level difference (1.5 m of sodium) between the hot and cold pools. The heat from IHX is in turn transported to eight steam generators (SG) by sodium flowing in the secondary circuit. Steam produced in SG is supplied to the turbogenerator to produce electric power of 500 MWe. The main vessel houses the entire primary sodium circuit including the core. The inner vessel separates the hot and cold sodium pools. The reactor core consists of about 1757 sub assemblies including 181 fuel sub assemblies. The control plug housing 12 absorber rod drive mechanisms is supported on the top shield. The top shield also supports the primary sodium pumps, IHX and fuel handling systems. PFBR uses mixed oxide with natural uranium and approximately 35% PuO_2 as fuel. For the core components, 20% cold worked D9 material (15% Cr–15% Ni with Ti and Mo) is used to provide better irradiation resistance. Austenitic stainless steel of type 316 LN is the main structural material for the out-of-core components and modified 9Cr–1Mo (grade 91) is chosen for SG. PFBR is designed for a life of 40 years with a load factor of 75%.

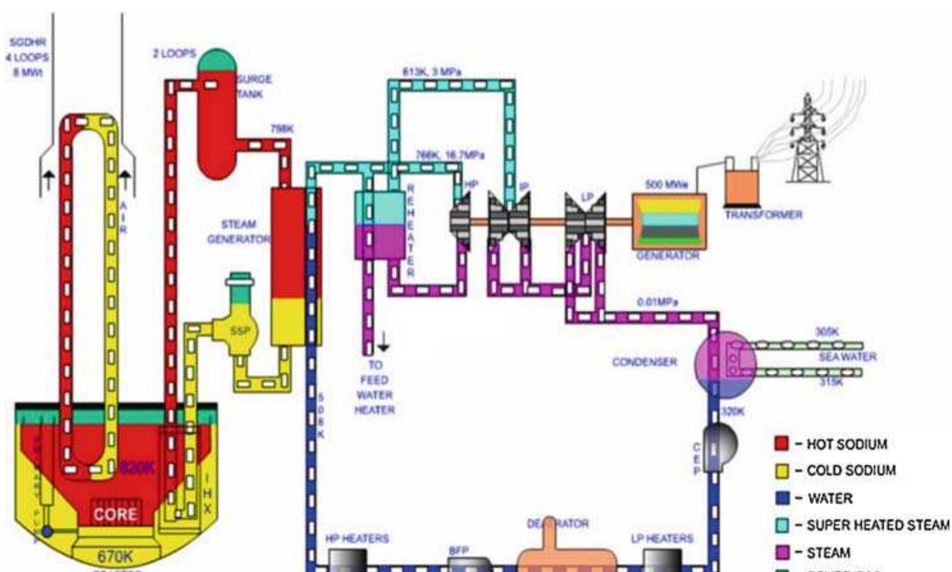

Figure 3. Heat transport circuit of PFBR.

4.2 *Design challenges*

Design of SFR requires a complete understanding of the unique behaviour of the fuel and structural materials under high temperature, sodium and irradiation environments as well as science and technology aspects in the domain of sodium chemistry, aerosol behaviour, sodium fire and sodium water reactions, special sensors for sodium applications (detection of water leaks in steam generator, sodium leaks, purity measurements, level detectors), thermal hydraulics and structural mechanics (turbulences, instabilities, gas entrainments, thermal striping, stratifications, racketing, etc.). Various failure modes are identified comprehensively and analysed in detail by employing validated analytical, numerical and experimental techniques.

4.3 *Technological challenges*

4.3.1 *Manufacturing.* Realization of SFR technology calls for the enhanced capability of industries to manufacture large dimensioned thin-walled welded shell structures made of austenitic stainless steel petals with close dimensional tolerances (main vessel, safety vessel, inner vessel), machining of large dimensioned and tall slender components with stringent tolerances (grid plate, absorber rod drive and component handling systems), fabrication of large size box structures with controlled distortions, hard facing technology with special materials (grid plate), development of inflatable seals, large size bearings and high-temperature fission chambers. Development of these components was possible through detailed technology development exercises.

For the manufacture of components, robust construction codes, standards and methodologies were employed. The most important aspect is the specification of manufacturing tolerances, which should be employed based on the rational basis of functional requirements and structural integrity considerations and they should truly reflect the earlier experiences and industrial capability. For those items, which the industry is attempting for the first time, appropriate and novel mock-up trials are essential. The assembly sequences should be well understood and efficient handling schemes are to be designed and validated. It is essential that the components should be manufactured practically with insignificant repairs.

Manufacturing of large-diameter thin shell structures involves several challenges: basic plates should not have any defects such as laminations (high-quality control is essential), large lengths of welds while integration of individual petals, stringent control on the manufacturing deviations, such as form tolerances ($< 1/2$ thickness), verticality and horizontality ($<\pm 2$ mm), high-quality welds and low residual stress to be achieved without any heat treatment. To sensitize the Indian industries and assess manufacturing tolerances that can be achieved by them, elaborate manufacturing technology development works were undertaken prior to the start of construction. Figure 4 provides the details of form tolerances achieved in the fabricated thin shells, viz., main vessel, safety vessel and inner vessel.

4.3.2 *Challenges in civil construction.* The concept of interconnected buildings has been adopted for the nuclear island of PFBR. The nuclear island extends over 100×92 m^2 area with very tall buildings; the highest among them is the RCB which is about

Component	ASME	Form Tolerance on Radius (mm)		
	($ID_{Max} - ID_{Min}$)	RCC-MR	PFBR	
			Specified	Achieved
Main Vessel	± 70	± 50	± 12	< ± 12 (during fit up) < ± 18 (at isolated locations)
Safety Vessel	± 70	± 50	± 12	± 12 (at majority locations) ± 18 (at isolated locations)
Inner Vessel	± 67	± 50	± 12	< ± 8 (during fit up) < ± 20 (at isolated locations)

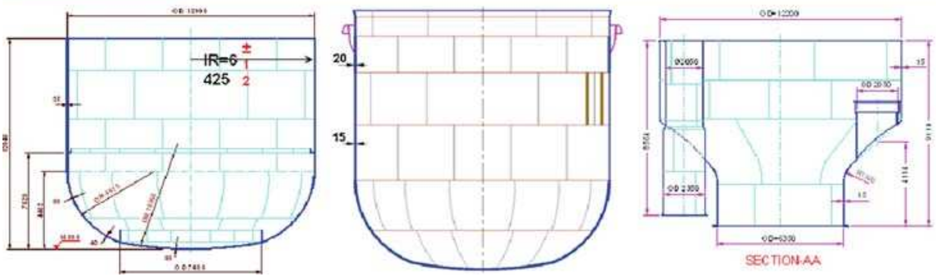

Figure 4. Form tolerances achieved in the thin shells of PFBR.

72 m tall. With the base raft thickness of 6 m, the civil construction of nuclear island interconnected buildings (NCIB) requires 35,000 m^3 of concrete. To complete the civil activities in time, it was required to carry out civil construction and equipment erection in parallel, involving state-of-art erection equipments and construction methodologies and highly optimized construction sequences.

4.3.3 *Erection challenges.* The installation of individual permanent components manufactured and assembled at the industries/site assembly shop has to be completed precisely meeting the specified erection tolerances towards meeting the functional requirements. Figure 5 depicts the erection sequences of the reactor assembly components. Among several erection tolerances, the most important ones are those for: (1) smooth insertability of absorber rods, (2) accurate positioning of core monitoring thermocouples at the top of each fuel subassemblies through accurate erection of control plug, (3) smooth fuel handling components, smooth movements of ISI equipment in the intervessel space between the main vessel and safety vessel, and (4) ensuring uniform coolant flow over the Weir flow system in the main vessel cooling circuit. To achieve this challenging task, elegant methodologies were conceived, validated and realized. Due considerations were given to time schedule and economy. Transportation of thin shell structures from the site assembly to the support locations is another challenging activity in the construction, executed in an innovative way to achieve economy without affecting safety. Erection of very large dimensioned and slender components with very stringent dimensional accuracies (a typical tolerance achieved on horizontality over 15 m is less than ±1 mm) is the most challenging task completed for the first time in the country with systematically planned mock-up trials. Application of modern software has helped to resolve many of the assembly and construction problems.

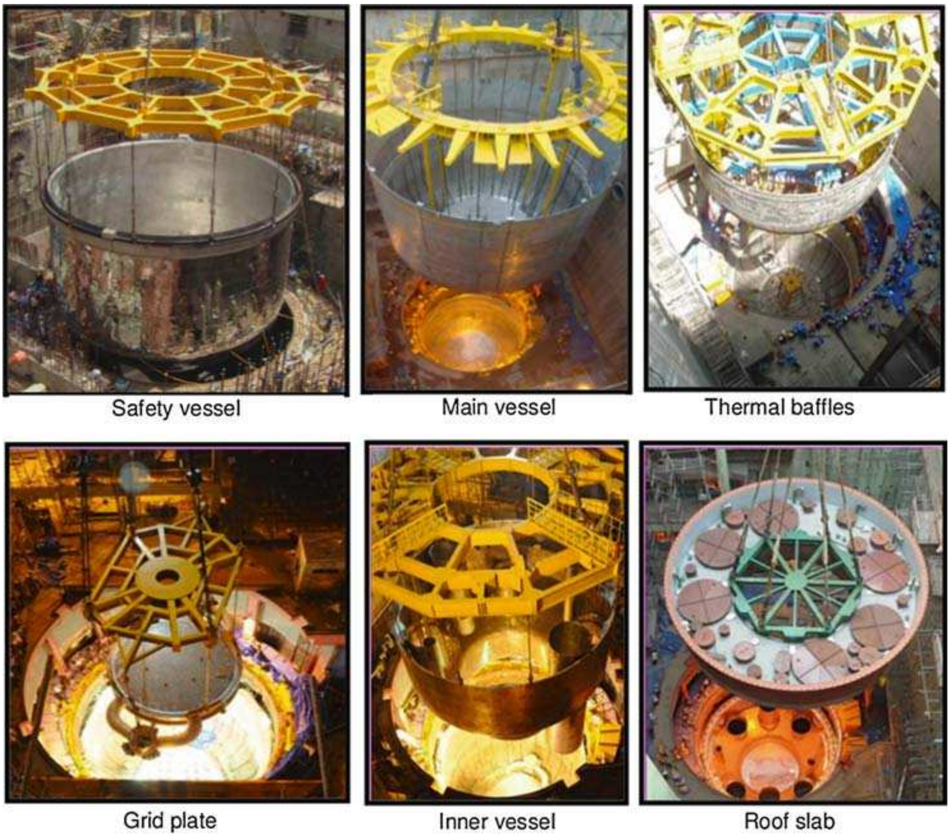

Figure 5. Erection stages of various components in PFBR construction.

4.4 *Current status of the PFBR project*

Erection of all reactor components has been completed successfully. Pictures of the status of the components erected at the site are depicted in figure 6. Project is already in the commissioning phase and most of the supporting systems have been commissioned and being operated on regular shift basis. Integrated commissioning work has commenced and the project is marching ahead to achieve the remaining milestones. Site has started vacuum pulling activities in sodium piping loops and argon filling, flushing and purging activities. All preparatory tasks for sodium filling are currently in progress. A systematic procedure has been established for accomplishing the first criticality of PFBR smoothly. It is expected that the reactor will attain its first criticality in 2015 and power will be raised subsequently after completing the specified reactor physics experiments.

5. FBR-1&2

The design of PFBR was re-looked in view of high demand on improving safety and economic aspects. Elaborate optimization studies by taking into account the PFBR expe-

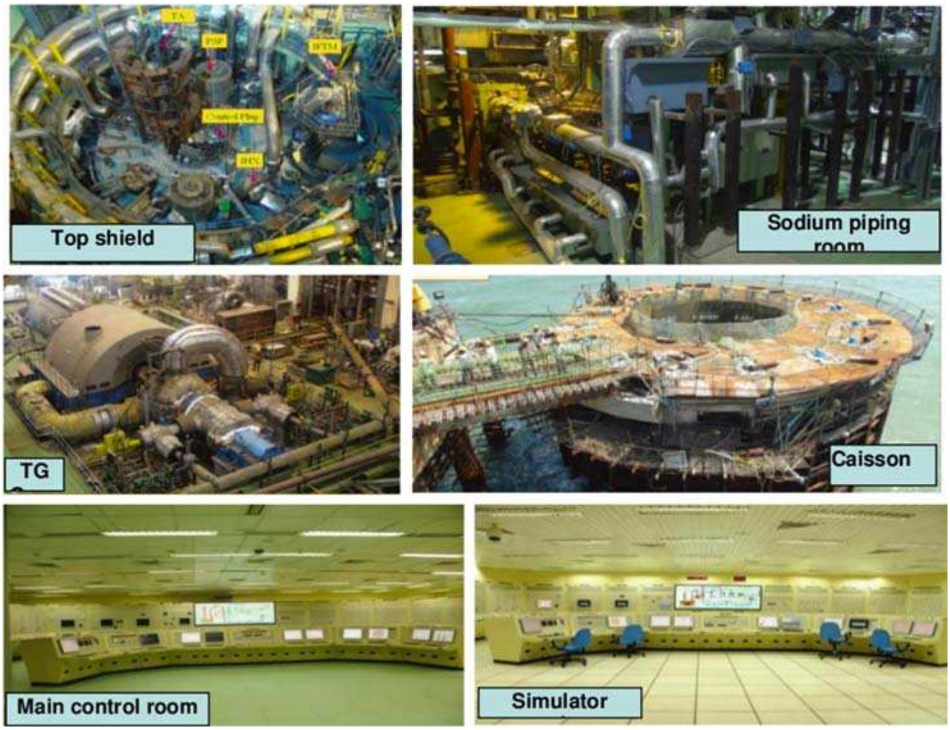

Figure 6. Current status of PFBR: major systems erected at the site.

riences, evolving safety criteria internationally, capacity and capability of Indian R&D institutions including DAE units as well as industries, have been completed for finalizing the conceptual design of FBR-1&2. The design revisions arrived at the optimization studies have been introduced in a calibrated manner in FBR-1&2. The main objective of such revisions is that the FBR-1&2 can be commercially exploited beyond 2030 towards realizing significant contributions targeted from nuclear option.

5.1 *Salient design features*

For FBR-1&2, MOX fuel and pool type with two sodium loops are retained. The major change is in the core design, to meet the safety criteria emerging after the Fukushima accident. Among several aspects, the most important one is to limit the sodium void reactivity to less than 1$. This value is 2.7$ for PFBR, which is the lowest among those reported for other international reactors designed before the Fukushima accident. Based on detailed optimization studies, it is concluded that a heterogeneous core is the most preferred solution with reference to core size, fuel inventory, available knowledge, matured analysis capability and international trend. Among a few potential options, introducing depleted uranium within pins of a few subassemblies occupying the core central zone and/or introducing radial blankets in the central zones provide attractive solutions to derive higher breeding ratio while restricting the sodium void reactivity. The

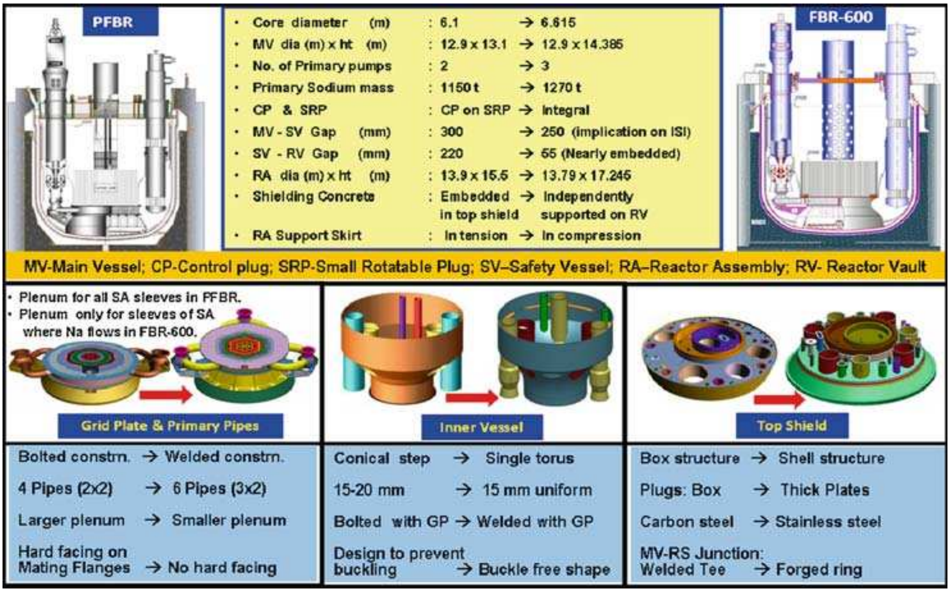

Figure 7. Evolution of reactor assembly design for FBR-600.

preliminary analysis has indicated that breeding ratio of ∼1.2 with the sodium void reactivity not exceeding 1$ is also possible.

Apart from core design, a few modifications have also been done in a calibrated manner. The major revisions resulted from the optimization studies are: rise of the reactor power from 500 to 600 MWe, twin unit concept (2 × 600 MWe units), use of 2 1/4 Cr–1 Mo in place of 304 LN for cold leg components and piping, three SG modules/loop with increased tube length of 30 m (PFBR has four modules/loop of 23 m length), 85% load factor, 60 years design life, reduced construction time (6 yr) and enhanced burn-up (up to 200 GWd/t to be achieved in stages). Further, significant improvements have been introduced in the reactor assembly design (figure 7): (i) welded grid plate with smaller plenum to accommodate only those sleeves which support core subassemblies through which sodium flows, (ii) inner vessel having single curved redan with uniform thickness, (iii) thick plate rotatable plugs, (iv) control plug integrated with small rotatable plug, (v) torus shaped thick plate roof slab, (vi) support skirt for reactor assembly kept under compression, (vii) safety vessel made of carbon steel embedded with reactor vault and (viii) simplified fuel handling scheme excluding the inclined fuel transfer machine.

The improved design concepts have indicated significant economic advantages: reduced material inventory (∼25%), simplified fuel handling scheme and reduced manufacture time as well as enhanced safety. These innovations are being tested through systemic research, technology development, testing and evaluation, etc.

5.2 *Enhanced safety features*

In view of the evolving trends of safety features, in particular after the Fukushima accident, there is need to give due considerations to enhance the core safety. Specifically, two aspects are to be considered in FBR-1&2: (1) analysis of all beyond-design basis events (DBE) including prolonged station blackout conditions resulting in severe core damage and large radioactivity release to the public and (2) practical elimination of severe accident scenarios. The DBEs have been split into three subcategories, viz., design extension condition-1 (DEC-1), design extension condition-2 (DEC-2) and practically eliminated condition (PEC). For the events coming under both DEC-1 and DEC-2, the site boundary dose is limited to 20 mSv. Further, for the events coming under DEC-2, consequences should be limited to within some distance and time duration (being evolved internationally). Events involving overheating of fuel pins due to inadequate cooling of the core (prolonged station blackout condition, for example) come under DEC-2. The events that could release a large quantity of radioactivity materials to the public are categorized under PEC. A few events under this category are: (1) failure of structures lying along the core support path (roof slab, main vessel, core support structure and grid plate), (2) simultaneous failure of main vessel and safety vessel, (3) CDA and (4) recriticality of the core.

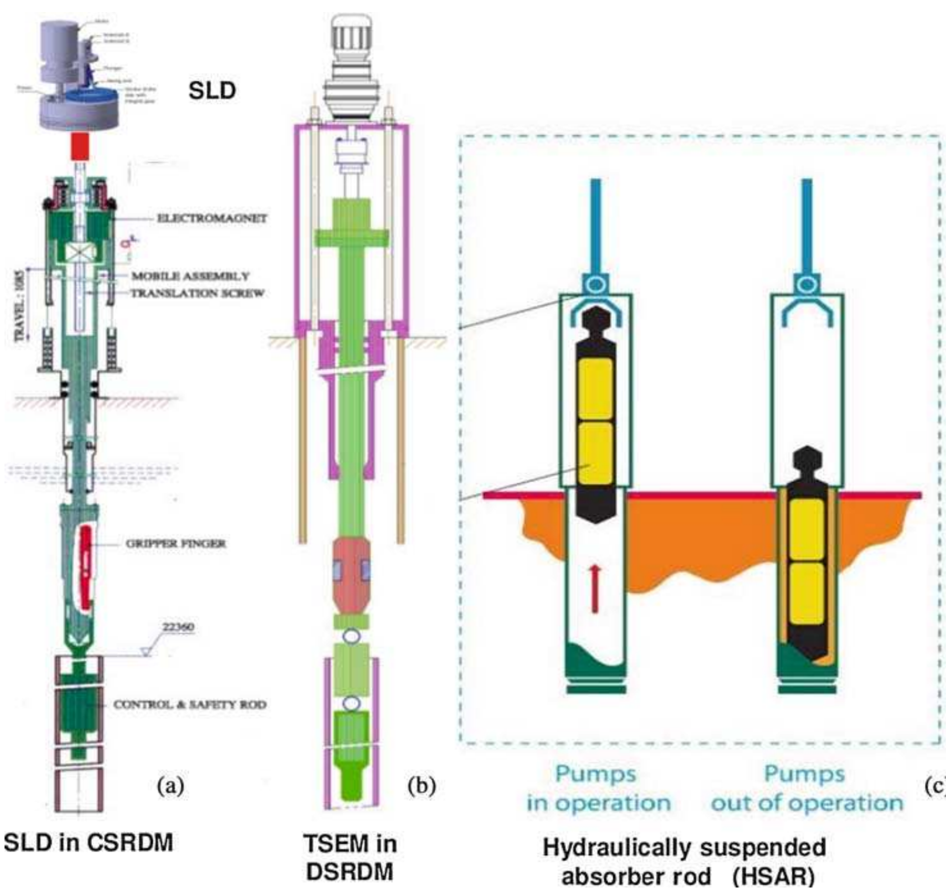

Figure 8. CSRDM and DSRDM with passive safety features for FBR-600.

5.2.1 *Improved reactor shutdown system.* Major improvements under consideration are: (1) enhancing the reliability of CSRDMs and DSRDMs (as in PFBR) with the introduction of passive safety features and (2) adequately addressing the recriticality issue. Towards further improving the reliability of shutdown systems (at least by one order with respect to PFBR), passive safety features are introduced: stroke limiting device to limit the uncontrolled withdrawal of control and safety rods in their drive mechanisms (figure 8a) and temperature-sensitive magnet/magnetic switch (Curie point magnet) in the diverse shutdown rod drive mechanisms (figure 8b). These apart, introduction of hydraulically suspended absorber rods that would be dropped immediately once the primary sodium flow reduction is caused by the initiating events such as rupture of more than one primary pipes, seizure of all primary pumps, etc. is under consideration (figure 8c). To avoid recriticality, adequate number of ultimate shutdown system (USD) that work on dropping of either liquid (Li-6) or granules (enriched B4C powder) will be introduced. The recriticality issue and concept of USD are explained schematically in figure 9. Finally, the scheme of shutdown systems (type, number and location) will be decided based on deterministic approach with due considerations on probabilistic approach. However, R&D activities on the systems mentioned will be continued and adequate knowledge and data will be accumulated. R&D includes introduction of such systems in FBTR itself to increase their confidence under actual environment (sodium, irradiation and high temperature).

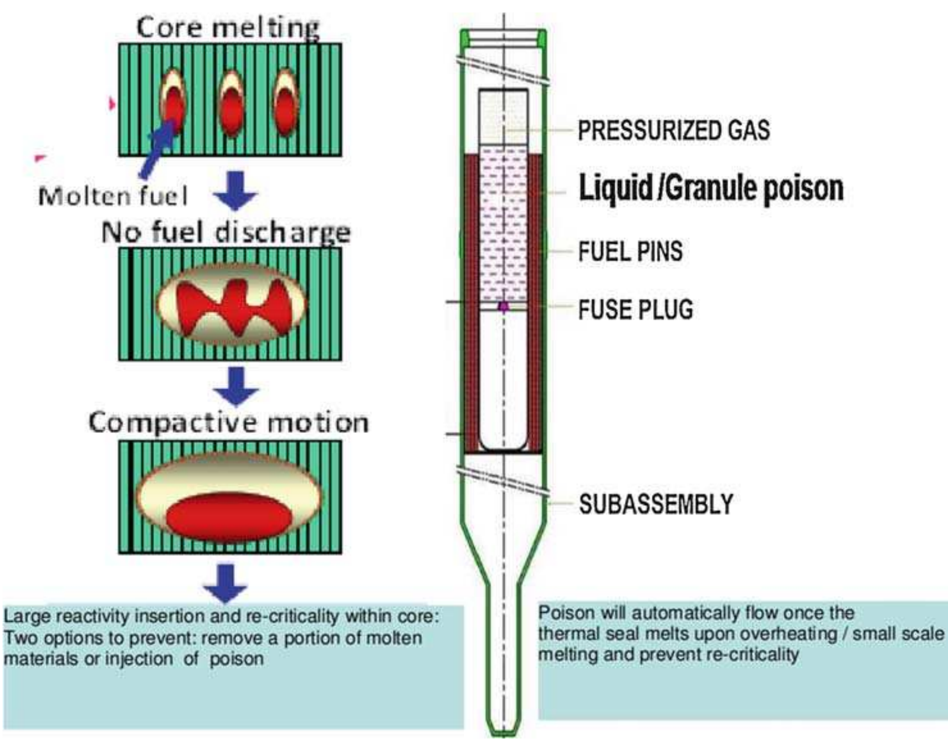

Figure 9. Recriticality scenario and option to prevent in FBR-600.

5.2.2 *Improved decay heat removal (DHR) systems.* The availability of DHR systems with reliability has to be ensured under five situations: (1) fuel handling, (2) in-service inspection (ISI), (3) DBEs, (4) DECs and (5) post-accident conditions. High emphasis is given to address the prolonged station blackout condition. For the first three situations (figure 10), dedicated DHR systems (4×10 MWt) will be introduced in the secondary sodium circuits (called SSDHR), instead of the operating grade decay heat removal system (OGDHR) introduced in steam water system in PFBR. This decision is taken after detailed assessment of design and operational simplicity and availability (a marginal cost increase needs to be tolerated). For taking care of DHR during DEC (situation 4), safety grade decay heat removal system (SGDHR) introduced in PFBR will be retained. SGDHR can be made operational by appropriate opening of the dampers in case of any DBEs resulting in loss of power to the secondary sodium pumps. However, design studies are in progress to make the SSDHR operational even during the loss of power to pump, to ensure high reliability of DHR requirement. Finally, to meet the DHR requirement during post-accident situations, the current features incorporated in PFBR will be retained,

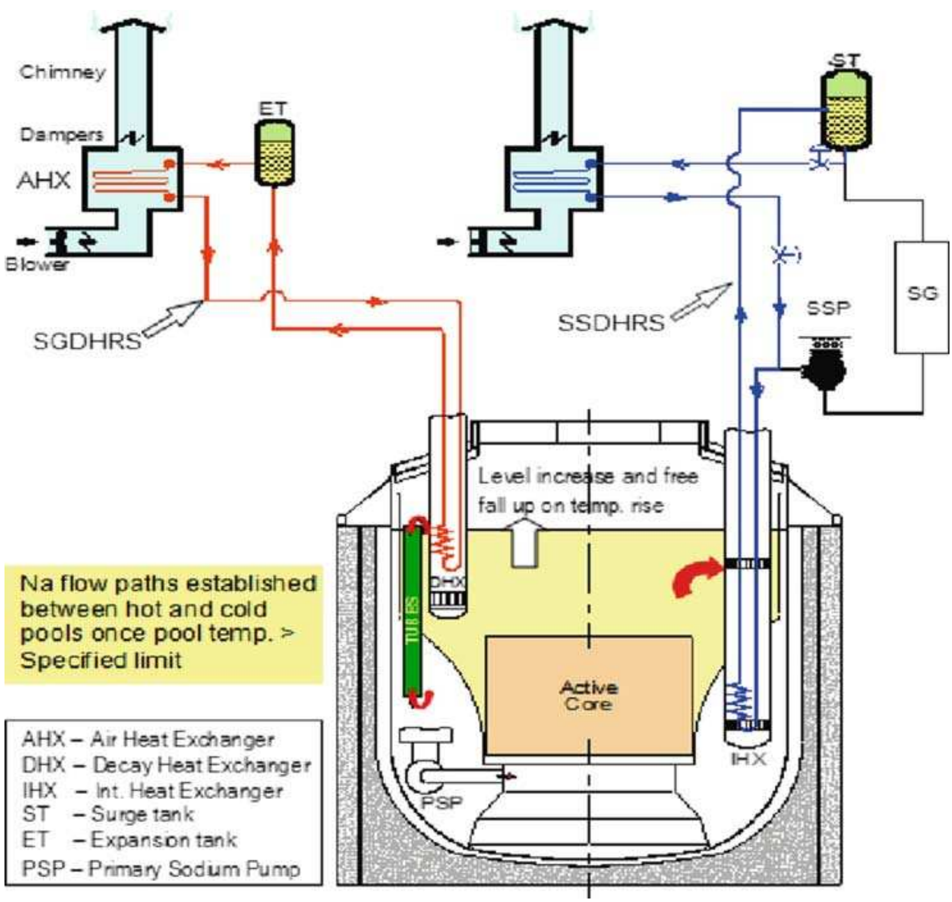

Figure 10. Judicious combination of active and passive DHR systems for FBR-600.

i.e., ensuring heat removal capacity of SGDHR after CDA and core catcher to support the core debris resulted from the CDA. Further improvements required are: ensuring the heat removal capacity features in core debris resulting in whole core melt down. Though it has been shown by CFD analysis that large perforation created by the molten fuel, while melting through the grid plate and CSS facilitate adequate natural circulation of sodium to remove the heat from debris settled on the core catcher and transport to SGDHR inlet windows through natural convection mode, considerations are being given to incorporate a few pipes penetrating through the inner vessel for providing alternate/additional passages for the sodium flow once the mean temperature exceeds a certain value. R&D on validation of this concept is in progress in the PATH facility commissioned for this purpose.

5.3 *Current status*

Conceptual design documents have been prepared and reviewed by experts. R&D activities identified for FBR-600 are being pursued at IGCAR, BARC and BHAVINI. Further, a technology development activity to manufacture the integrated reactor assembly with permanent components has been taken up jointly by BHAVINI and IGCAR . This reactor assembly will be manufactured as per the applicable FBR specifications. Detailed project report (DPR) will be submitted by BHAVINI shortly. Project safety analysis report (PSAR) will be prepared jointly by IGCAR and BHAVINI within two years. It is expected that the construction of FBR-1&2 will start in 2017 to meet the target date of commissioning in 2024–25.

6. Epilogue

Successful operation of FBTR since 1985, successful commissioning of PFBR and full power operation in the forthcoming years (from 2015), construction and commissioning of FBR-1&2, which have potential for commercial exploitation with high public acceptance, robust roadmap for metallic reactor programme for the realization of rapid growth rate of nuclear energy contributions from FBRs, availability of high quality scientists and engineers in the department as well as in other collaborating institutes in the country and enhanced international cooperation in a calibrated manner will certainly demonstrate that India will be the world leader of SFR technology at least by 2025.

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