500 MWe PFBR : Concept to Realisation and Approach for Future 500 MWe FBRs

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500 MWe Prototype Fast Breeder Reactor

- Economics
- Basic Features
- Safety and Reliability
- Resource Utilisation
- R & D
- Manufacturing Technology
- Licensing Experience
- Construction
- Fuel Cycle

Evolutionary Design + A few innovative features in comparison to FBRs built or then under design abroad

Indian Thermal SMRs

Future 500 MWe FBRs

Enhanced Safety and Improved Economics
A few innovative features over PFBR
Three Stage Nuclear Power Programme

World class performance

Global Advanced Technology

Globally Unique

Stage – I

PHWRs + LWRs

Stage – II

- 40 MWt Fast Breeder Test Reactor in operation
- 500 MWe PFBR - under advanced stage of construction

Stage – III

300 MWe AHWR

$\text{U}^{233}$ – Th Reactors

- 6 X 500 MWe Twin Unit MOX
- 120 MWe Experimental Metal Fuel Reactor
- 1000 MWe Metal Fuel Reactors
SMALL & MEDIUM SIZE INDIAN PHWRs
220 MWe and 540 MWe small and medium size PHWR reactors – Optimum solution for the countries where medium size electricity grids are in operation, and are keen on expanding their power base.

15 x 220 MWe and 2 x 540 MWe PHWRs in operation. PHWR plants have undergone peer review by WANO.

Availability factor : 82 to 91%

Uninterrupted plant operation : As high as 529 days from one of the 220 MWe unit and 590 days from one of the 540 MWe unit

Excellent safety record.

Indian industries capable of manufacturing reactor components.

540 MWe units commissioned within 5 years from first pour of concrete

Capital cost much lower than other reactors in the international market.
Currently, India has 20 operating Nuclear Power Plants

- Oldest operating Boiling Water Reactor (BWR) in the world.
- Unit-2 of Tarapur Atomic Power Station operated uninterruptedly for 590 days.

First to be built in EPC mode – construction in less than 60 months

- TAPS-3&4 540 MWe PHWR
- Economy Of Scale
  - Further up scaling to 700 MWe
  - Four projects launched

TAPS-1&2 160 MWe BWR

1970s Technology Demonstration
1980s Indigenisation
1980s Standardisation
1990s Consolidation
2000s Commercialisation

RAPS-1&2
MAPS-1&2
NAPS-1&2
KAPS-1&2
KGS-1&2
RAPP-5&6

220 MWe PHWRs
• Tarapur- 3 & 4 were completed within 85% of estimated gestation period.

• First criticality of Tarapur-4, 540 MWe PHWR was achieved in less than 5 years
Performance – Average dose from NPPs at 1.6 km (2003-2009)

Radiation Dose to Occupational Workers and Public is insignificantly small fraction of natural background and the regulatory limit.
India graduated from technology user to technology developer and is in a unique position to understand needs from the point of view of technology user.
India’s capabilities

- Technologies
- Site Evaluation
- Design & Engineering
- Procurement
- Commissioning
- HRD & Training
- Quality Assurance
- Regulatory Support
- Construction
- Manufacturing
- Operation & Maintenance
- Services
Industrial Infrastructure

- Industry capable of manufacturing critical nuclear components / equipment to exacting standards.
- Large engineering industry capable of taking up EPC contracts.
- Construction industry capable of executing large projects.
- Manufacturing industry base of cement, steel and other materials including special materials.
- Facilities within Department of Atomic Energy for production of special nuclear materials like Heavy Water, Zirconium etc.

Main Nuclear Equipment/Components
- Calandria/ reactor pressure vessel
- Radiation Shield Assemblies
- Steam Generators
- Nuclear grade Pumps
- Main Air Lock Assemblies
- Turbine Generators and associated steam and water system equipment
- Cooling water systems for disposing heat to environment

Calandria and end-shield at manufacturer shop
Erection of major equipment for PHWRs

Steam Generator Erection

Erection of Turbo Generator

End Shield Erection

Calandria Erection
AHWR: Some Major Design Features

AHWR is a 300 MWe vertical pressure tube type, boiling light water cooled and heavy water moderated reactor.

- Flexible fuel cycle option
  - Pu-Th fuelled
  - Fuel option with LEU + Thorium
- Capability to remove decay heat safely through passive system
- Two shutdown systems
- Design life of 100 years
- Easily replaceable coolant channels

Design validation through extensive experimental programme.

Pre-licensing safety appraisal by AERB

Site selection in progress.
Design Objectives of PFBR

* Commercial demonstration reactor.
* Indigenous design and manufacture.
* Lessons learnt from Fast Breeder Reactor experience to be incorporated in design.
* Cost reduction efforts of (Superphenix → European Fast Reactor) foreign Fast Breeder Reactors to be well studied.
* Simplification in design. Reduction of number of components with due consideration for availability / capacity factor.
* Design to meet applicable Atomic Energy Regulatory Board safety guides for Pressurised Heavy Water Reactors.
* In-service Inspection to be incorporated at design stage.
* Marginal breeder
* Utilisation of limited Plutonium
* Co-located fuel cycle facility
Plant Size

* Size of national fossil fired power plants of 500 MWe had reached maturity.
* Construction of 540 MWe Pressurised Heavy Water Reactor was under progress at the time of decision (2 units now in operation).
* Economy of scale.
* 250 MWe unit energy cost ~ 20% higher than 500 MWe.
* Rapsodie → Phenix Joyo → Monju BR-5 → BN350 R&D needed for design will not vary significantly whether unit is say 250 MWe or 500 MWe.
* Manufacturing technology and associated aspects in scaling from Fast Breeder Test Reactor in one step to Prototype Fast Breeder Reactor recognised. Main consideration to standardise the basic design in short time that will be the forerunner of a series of units to follow.
Incorporates lessons learnt from 400 r-y of operational experience (Design aspects specific to sodium systems, material selection, manufacturing aspects etc.)

- Pool type reactor (no penetration in main vessel)
- 2 independent shut down systems. One system not available + 1 rod not available – cold shut down.
- Safety grade decay heat removal system based on 4 independent circuits in 8 MWt. Diversity in design of HXs. Natural circulation on sodium and air side.
- All sodium piping and components designed for OBE and SSE, irrespective of safety classifications.
- No risk of primary sodium leak to air for DBE (N₂ inerted).
- State-of-art single wall steam generator with butt welded tube to tube sheet joints. Leak detection, protection against small, medium and large leaks.
- Whole core accident as Cat. 4 DBE for site boundary dose, integrity of main vessel and decay heat removal system.
- In-service inspection for main vessel and steam generator tubes.
Fuel Subassembly

- MOX
- Relatively Small Pin diameter of 6.6 mm to reduce fissile inventory
- Annular Fuel Design
- 2 Enrichment Zones
PFBR 37 Pin Simulated fuel subassembly 112 GWd/t

Experience in Regulatory review in design, operation and life extension

Cradle for human resources
Training of O & M Personnel for PFBR

25 years of successful operation in sodium technology. Confidence in operation of PFBR
Behaviour of SS 321 in High Temperature Sodium Circuits

- Cracks in SS 321 tee junction in Phenix (Sodium Leak).
- Cracks in circumference welds in PFR reheater vessels manufactured in SS 321, welded with SS 347 (Sodium Leak).
- Cracks in SS 321 Phenix steam generator (inspection; Life extension programme).
- SS 321 operating experience in fast reactors not satisfactory. Phenix secondary sodium piping and steam generator in SS 321 replaced by SS 316 LN.
- Failure due to delayed reheat cracking or stress-relief cracking.
- SS 321 and SS 347 prohibited for PFBR
PFR Steam Generators

- Operating Experience
  - One reheater in Austenitic Stainless Steel suffered extensive damage due to caustic stress corrosion cracking.
  - Highest load factor in first 10 years - 12%.

PFBR
- Material: Mod 9Cr-1Mo
- Tube to tube sheet weld
  - Raised spigot type
  - Preheated and PWHT
Number of Components / Loops

- **Economics**
  - Operating experience on sodium components, in particular sodium pumps and sodium heated steam generator.
- **Capacity Factor**
  - Minimum number of components other than steam generator consistent with safety.
- **Decay heat removal aspects**
  - 2 Primary Pumps
    (Operation not foreseen with 1 Pump)
  - 2 Secondary Pumps
  - 1 Turbine
  - 2 IHX / Primary Pump.
    (Economics of pool type reactor)
  - 4 Steam Generator / Loop
    (Based on (N-1) operation of affected loop)
- 90% power operation in case of one SG not available
- **Overall optimization**
### Steam Generator

**Particulars**

- **Thermal Power/SG**: MW 158
- **Number of SG**: 8
- **Sodium Inlet/Outlet temp**: °C 525 / 355
- **Water Inlet/Steam Outlet temp**: °C 235 / 493
- **Steam Pressure at SG outlet**: MPa 17.2
- **Materials**: Mod. 9Cr-1Mo
- **Number of tubes**: 547
- **OD of tube**: mm 17.2
- **Thickness of tube**: mm 2.3
- **OD of SG shell**: mm 1237
- **Overall length**: m 25
- **Weight of module**: t 42
Absorber Rod Worths

- Two diverse shutdown systems
- CSR + DSR minimum 5000 pcm – Shutdown margin
  CSR –1 cold shut down by margin of 1$
  DSR –1 cold shut down by margin of 1$
- CSR + DSR with loading error
  1 FSA replacing absorber SA – 1850 pcm
  2 Maximum worth absorber SA missing – 3530 pcm
- CSR - 9 rods of 65% enriched $B_4C$ in
  Central portion + 20 cm top and bottom
  natural $B_4C$
- DSR - 3 rods of 65% enriched $B_4C$
  9 CSR + 3 DSR worth - 13300 pcm

CSR - Control and safety rod
DSR - Diverse safety rod
- 4 independent circuits of 8 MWt each.
- Diversity in design of heat exchangers.
- Natural circulation on both intermediate sodium and air side.
- Redundancy for dampers.
- Automatic operation with no credit for 30 minutes.
SAMRAT Model
(1:4 scale water model of PFBR primary circuit)

- Gas Entrainment studies
- Free Level Fluctuation Studies
- Flow distribution in hot plenum
- Validation of numerical codes

Subassembly hydraulic test rig

- Pressure drop studies in fuel subassembly
- Optimization of Pressure drop devices
- Flow Induced Vibration studies in fuel subassembly
Testing of Shutdown Systems of PFBR

 CSR & CSRDM
Upper & Lower Parts of CSRDM

 No. of Scram cycles tested in Na
at 803 K : 500
at 823 K : 1093

Drop time of CSR : 610 ms < 1 s

DSR & DSRDM
Upper & Lower Parts of DSRDM

No. of cycles tested in Na
(Scram) at 773 K : 186
(Translation) at 670 K : 986
Scheme for Authorization for PFBR Construction

**AERB**

**PROJECT DESIGN SAFETY COMMITTEE**

**CIVIL ENGINEERING DESIGN SAFETY COMMITTEE**

**INTERNAL SAFETY COMMITTEE WORKING GROUPS SPECIALIST GROUPS**

**APPLICANT**

**AERB BOARD**

**CONSENT FOR CONSTRUCTION**

- Stage wise clearance for construction
- Review of construction methodology
- Safety review has helped to make PFBR design and construction robust
- First nuclear plant to undergo public hearing by State Government
PFBR Construction Highlights

Erection of Safety Vessel

Primary pipe with Grid Plate

Roof Slab: Fabrication at Site Assembly Shop

Erection of Main Vessel

Secondary Sodium Pump

Steam Generator

Primary Sodium Pump
Plans to build 6 FBRs of 500 MWe each by 2023. Two FBRs at Kalpakkam to make use of co-located Fast Reactor Fuel Cycle Facility to reduce Fuel Cycle Cost. Sanction granted for pre-project activities at Kalpakkam

Cost Reduction consistent with enhanced safety will be main objectives

CFBR will incorporate lessons learnt from construction of PFBR, in particular, manufacturing specifications, material procurement, means to reduce manufacturing time and plant layout

CFBR will have changes in design and safety requirements to reflect experience gained through regulatory review

Revised Safety Criteria under final stage of review by safety committee

Reduction in construction time

Well defined R&D tasks
CFBR : Shutdown Systems

Reliability Target $10^{-7}$/reactor year
CFBR: Decay Heat Removal

**CFBR:**
3 SGDHR circuits with forced cooling (2/3 of heat removal under natural convection)

&

3 SGDHR circuits with natural convection cooling each with a power removal capacity of 6 MWt

**PFBR:**
4 independent SGDHR loop each with 8 MW heat removal capacity
The SGDHRS is completely passive except for the dampers at the inlet and outlet of Air Heat Exchangers
Reliability target – $10^{-7}$/Ry

6 MW each on forced circulation.
~ 4 MW on natural circulation
CFBR: Improvements Proposed For Reactor Assembly

1. Welded grid plate with reduced height
2. Eight primary pipes
3. Optimization of main vessel thickness
4. Integrated liner and safety vessel with thermal insulation arrangement
5. Thick plate Top shield
6. Dome shaped roof slab
7. Conical shell for reactor assembly support
8. Inner vessel with single toroidal shell (redan) directly connecting grid plate with the upper cylindrical shell

Specific Steel Consumption is reduced by 25%
Steam Generator

Objective: To reduce the number of butt welds of tube to tube sheet raised spigot. This reduces manufacturing time. Favourable impact on reactor schedule and enhances safety.

- Tube length increased from 23 to 30 m.
- Number of steam generators reduced from 8 to 6 (3 SG/loop for future reactors)
- Operation flexibility to run with 3 + 2 SG of affected loop

PFBR SG
- 547 tubes [L-23m]
- 17.2 OD X 2.3 WT
- 40 yrs. Life
- 8 SG/plant
- Matl: Gr. 91
- No of tube – T/S joints
  547 x 16 = 8752

CFBR SG
- 433 tubes [L-30m]
- 17.4 OD X 2.4 WT
- 60 yrs. Life
- 6 SG/plant
- Matl: Gr. 91
- No of tube – T/S joints
  433 x 12 = 5196 (~41% Reduction)
India has reached technical maturity and competitiveness in PHWRs 220 and 540 MWe. SMRs are available for export at economical price. 700 MWe PHWRs are under construction.

India has demonstrated indigenous design and construction of 500 MWe sodium cooled fast reactor-SFR (FOAK). Lessons learnt to result in enhanced economics and improved safety for future 500 MWe SFRs.
Thank you