Fast Reactor Program in India

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Plenary Talk
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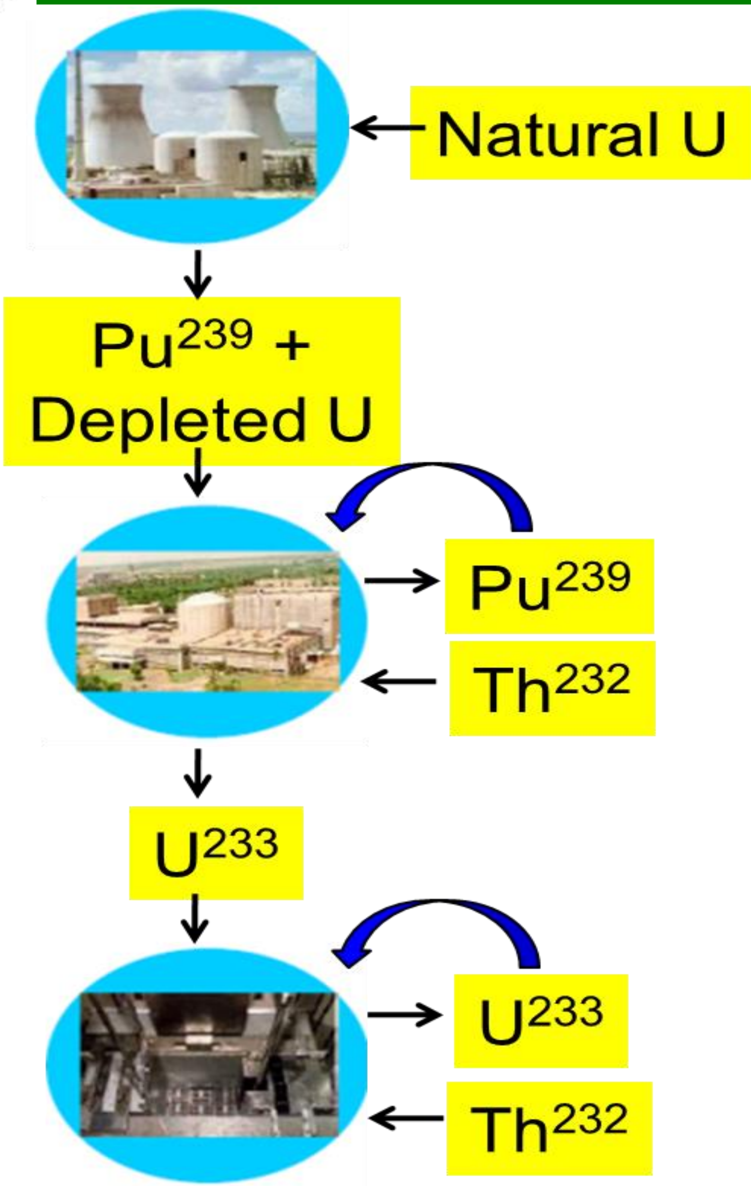
Structure of the Talk

- Role and advantages of Fast Reactors – Indian Context
- Fast Breeder Test Reactor
  - Genesis
  - Status
  - Lessons learnt over the last 36 years
  - Role in fast reactor fuel, structural material and human resource development
  - Life Extension Program
- Post Irradiation Examination of Fuels – Challenges and Significant Results
- Reprocessing of Fuels – Challenges
- Indian Fast Reactor Program – Way forward

Fast Reactors : Catalytic Linkage



- ✓ Provides perfect link covering natural nuclear resources of India
- ✓ Effective and optimal utilization of uranium
  - Better resource management
- ✓ Long term energy supply
- ✓ Higher growth rate with breeding
- ✓ Waste management
  - Incineration of radioactive waste from spent fuel
  - Reduction in long-term storage requirements
- ✓ Enhanced performance parameters
  - High temperature of operation
  - Higher thermodynamic efficiency
- ✓ Closed fuel cycle program is essential

Indira Gandhi Centre for Atomic Research Kalpakkam

Kalpakkam : Unique complex internationally with reactor systems utilizing all the three fissile isotopes – U235, Pu-239 and U-233

Fast Breeder Test Reactor:
Pu-239 (Operation)

Madras Atomic Power Station:
U-235 (Operation)

Prototype Fast Breeder Reactor:
Pu-239 (Comissioning)

KAMINI:
U-233 (Operation)

Fast Breeder Test Reactor (FBTR) – Genesis/Status

- 1968 - Decision to initiate fast reactor programme, beginning of FBTR in Fast Reactor Section of Reactor Engineering Division, BARC.
- 1969-Agreement signed between CEA and DAE to prepare project report to set up a fast breeder test reactor
- 1971 - Dedicated Centre setup at Kalpakkam -Reactor Research Centre – renamed Indira Gandhi Centre for Atomic Research (1985).
- 1972 - Ground breaking for the Fast Breeder Test Reactor. Parallely other facilities initiated.
- ~1978 – Decision to use Carbide Fuel
- Oct. 18, 1985 – First Criticality
- July 2006 - MK-I fuel reaches burnup of 155 GWD/t without failure
- Dec. 2009 – PFBR test fuel achieves target BU of 112 GWD/t at LHR of 450 W/cm
- **March 07, 2022 FBTR attains rated capacity - 40 MWt / 10 MWe**

FBTR today...

Unique U-Pu mixed carbide fuel. Record burn-up of 165 GWd/t – international milestone.

36 years of successful operation without any major incidence

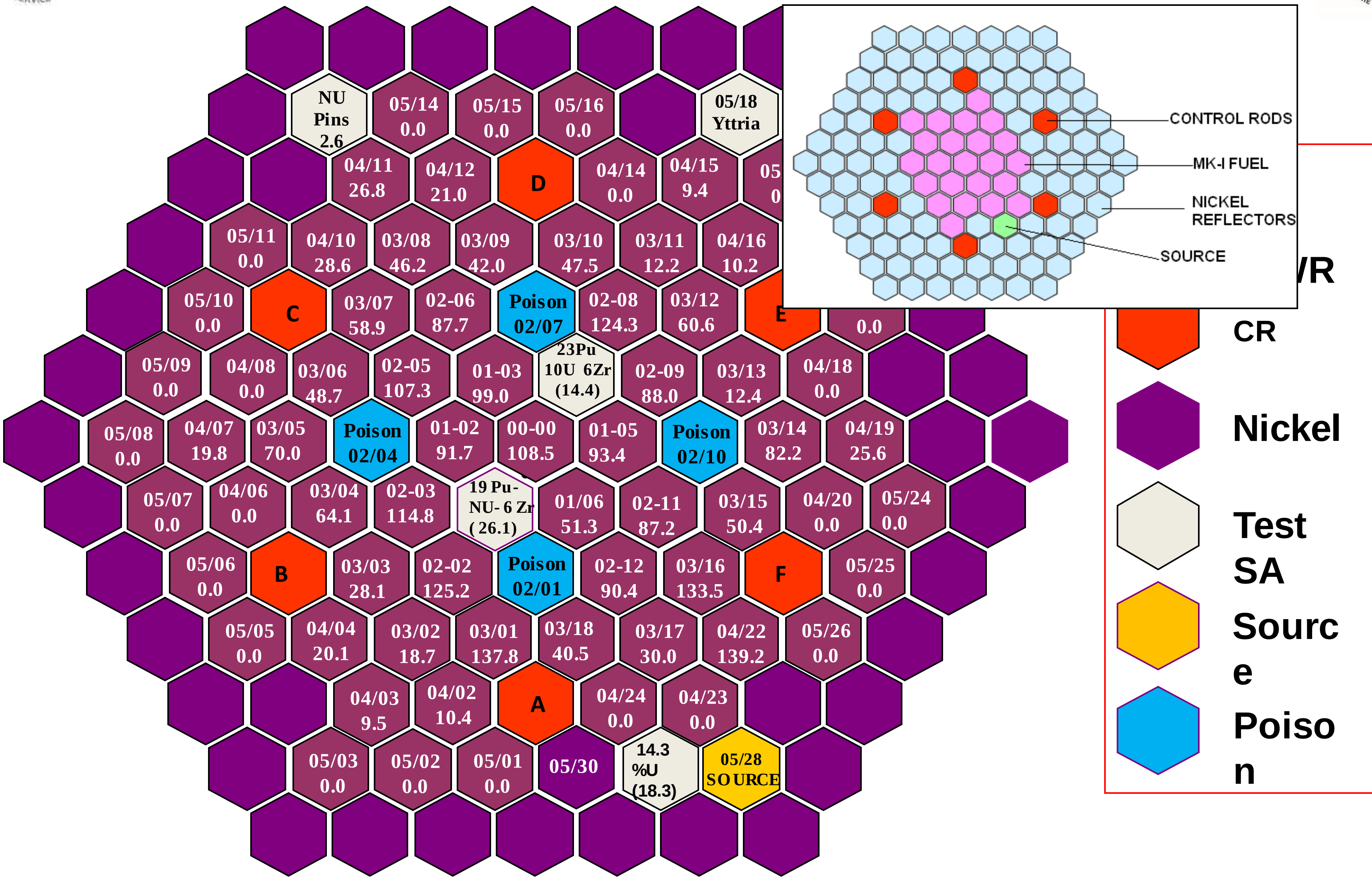
Indigenously developed sodium pumps login > 8,70,000 trouble-free cumulative service.

Operated at power level of 40 MWt;

Elecrical Power generated 10.0MWe

Cradle for Human Resource Development in Fast Reactors

Irradiation of yttria yielded ^{89}Sr, for the first time in India, used as a palliative for cancer patients – Societal Application



30^{th} Irradiation Campaign 68 SA Core

FBTR: Cradle of Human Resources

Grade	Number of personnel Trained
Engineers	57 (FBTR) 38 (BHAVINI) 6 Licences for FBTR operation
Scientific Assistants	29
Technicians	55

O&M Personnel of PFBR undergoing training in FBTR

Irradiation of various fuels

VARIOUS FUELS

MK-I PINS/SA 70%PUC-30%UC (155GWd/te)

MK-II PINS/SA 55%PUC-45%UC (100GWd/te)

MOX SA 44 % PuO_2 - 56%UO_2

PFBR MOX TEST PIN/SA 29 % PuO_2 & 53 % enriched U(112 GWd/te)

METALLIC FUEL PINS Natural U-6%Zr sodium bonded

METALLIC FUEL PINS Enriched 14.8%U -6%Zr sodium bonded

METALLIC FUEL PINS Natural U-19% Pu-6%Zr sodium bonded

TRISO coated particle/ Sphere pac fuel

Irradiation of various structural materials

STRUCTURAL MATERIAL

- Zr-Nb capsules for PHWRs & Glass Fibre Reinforced Plastic (GFRP) for IPR
- D-9 materials
- Grid plate material for life assessment & WC pellet for life extension
- SS316L Materials & Ferritic – Martensitic steel
- Ferro-boron shielding material
- Low dose irradiation SS304LN Materials
- Low dose irradiation SS316LN Materials
- CHTR Structural Materials

Life extension of FBTR

- FBTR has completed 36 years of safe & successful operation.
- Various studies carried out for extending the life of FBTR.
- The operational life of FBTR limited by the neutron damage to the grid plate which is a non-replaceable component.
- Limiting residual ductility of 10% uniform elongation is attained at 6.3 dpa for FBTR grid plate.
- At the end of 29th irradiation campaign, grid plate has accumulated 2.35 dpa
- Remaining residual life corresponding to 6.3 dpa: ~ 8 EFPY
- WC pellets planned to be introduced as lower axial shield in FSAs to reduce the fluence on the grid plate.
- By this, Life of FBTR extends by ~ 33% i.e. to limiting dpa - 8.19.
- FBTR expected to operate upto 2034/2035.

Challenges faced successfully during operation of FBTR

- In the last 36 years of operation, FBTR has faced several challenges. This includes
  - Fuel handling incident
  - Sodium leak from bellows sealed sodium service valves
  - Choking of hot argon communication lines between primary capacities
  - Leak from embedded biological shield concrete cooling coils
  - Fuel clad failure
  - Steam generator tube leak and its replacement

Challenges in PIE of FR Materials

- Characteristics of FR Core
  - High power density
  - Compact core
  - Efficient coolant is essential (sodium)
  - High burn-up
  - High fluence

Reactor →	Thermal	Fast Breeder
Fissile enrichment	0 – 3 ^{235}U	10 – 30 ^{239}Pu
Ave. $_0n^1$ energy	0.025 eV	100 keV
Burn-up (GWd/t)	~ 30	~ 100
$_0n^1$ flux n/cm^2.s	10^{14}	$5 – 10 \times 10^{15}$
$_0n^1$ fluence	10^{22}	$2 – 10 \times 10^{23}$
Ave. core power density, W/cm^3	~ 100	~ 300 - 400

Clement, 2020

Main factors that influence burn-up: sodium inlet temperature and LHR

Parameters for stage-wise increase of LHR and burn-up

- Post-irradiation Examination
  - Fuel swelling rate | Internal porosity within fuel | Fission gas release | Cladding and wrapper swelling and creep | Cladding - Pellet gap | Clad strain | Wrapper dilation | Fuel bundle wrapper mechanical interaction
- Reactor Operations
  - SA extraction force
- Thermomechanical modelling
  - Flow reduction through SA | Cumulative damage fraction on clad

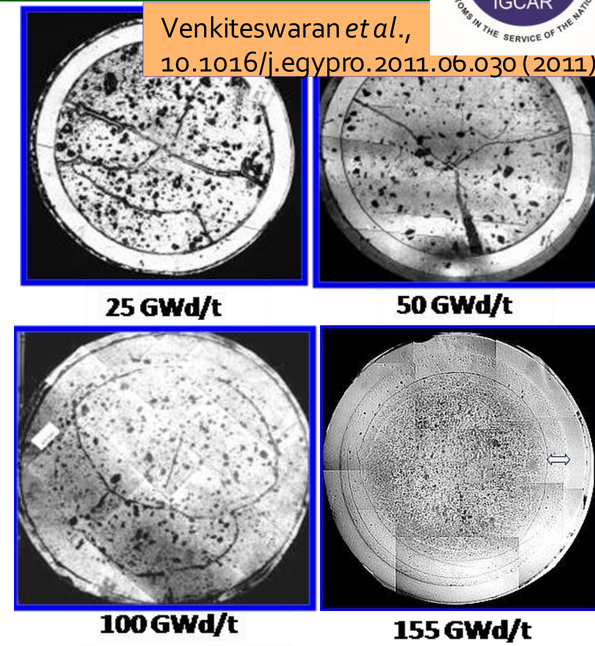
Criteria for burn-up limit

- **FG induced creep rupture**
- **Creep rupture due to FCMI**
- **Porosity exhaustion**
- **Hex-can sheath deformation**
- **Pin – duct interaction**
- **Residual ductility of cladding and duct**

High burn-up FR fuels and other materials involve handling of αβγ – high active fuel and potentially α-active structural materials, requiring specialised hot-cells, shielding and handling

PIE of irradiated FR Fuels – burn-up limits

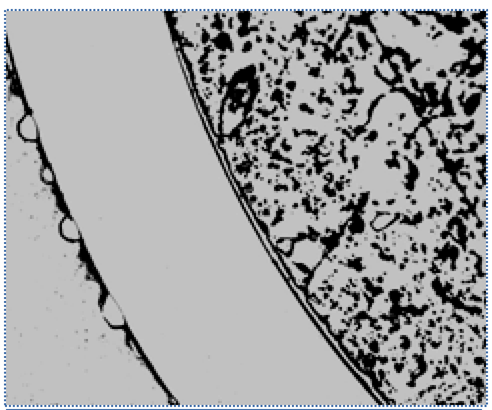
Venkiteswaran *et al.*, 10.1016/j.egypro.2011.06.030 (2011)



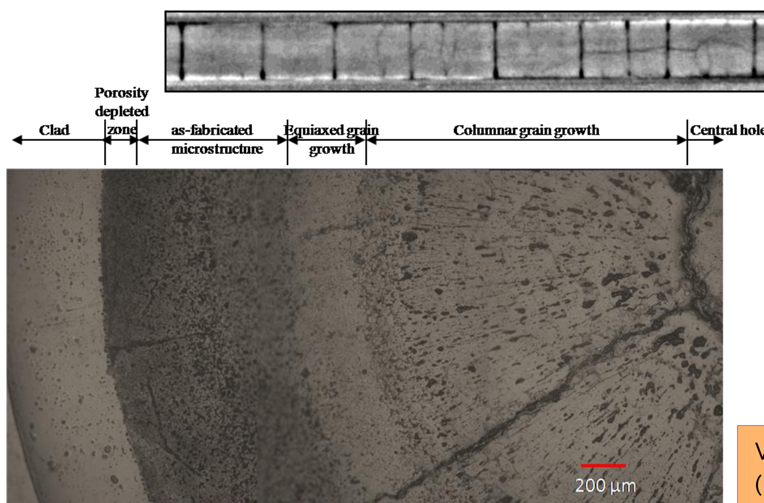
Carbide Fuel 25-155 GWd/t

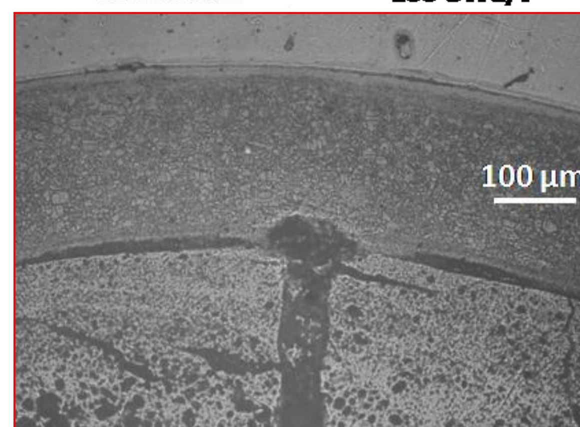
- Fuel pin cross-sections of 25 and 50 GWd/t burn-up fuel pins indicate radial cracks and progressive reduction in the fuel-clad gap
  - free fuel swelling rate estimated ~1 to 1.2% per atom% burn-up
  - lower than expected value (fuel design)
- At 100 and 155 GWd/t burn-up, fuel-clad gap is closed along the entire length of the fuel column
  - circumferential cracks indicate restrained swelling
  - porosity free outer rim at 155 GWd/t fuel pin cross-section due to creep of the fuel due under FCMI stress

MOX Fuel 112 GWd/t



Fuel-Clad gap reduction from 90-110 µm to 13 µm in 13 EFPD – BoL low LHR duration for fresh fuel

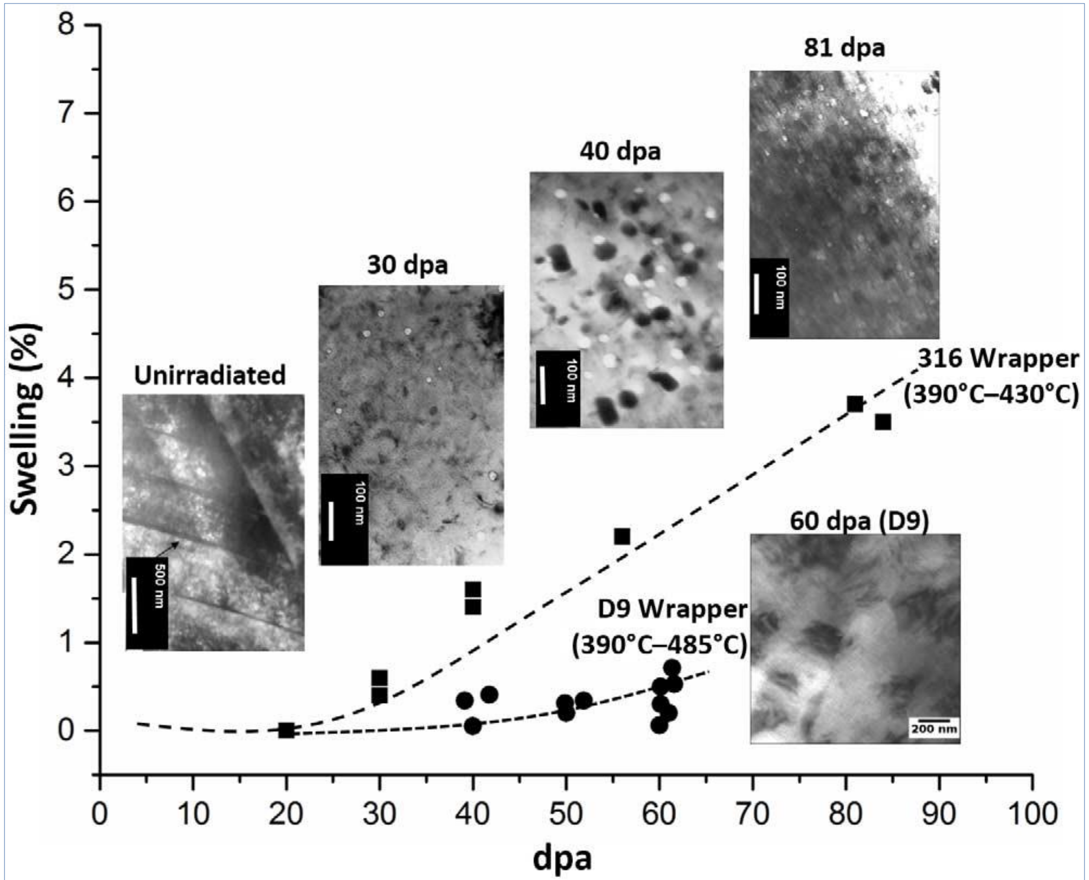




Venkiteswaran *et al.*, 10.1016/j.jnucmat.2014.01.045 (2014), Jayaraj *et al.*, 10.1016/j.jnucmat.2018.06.001 (2018)

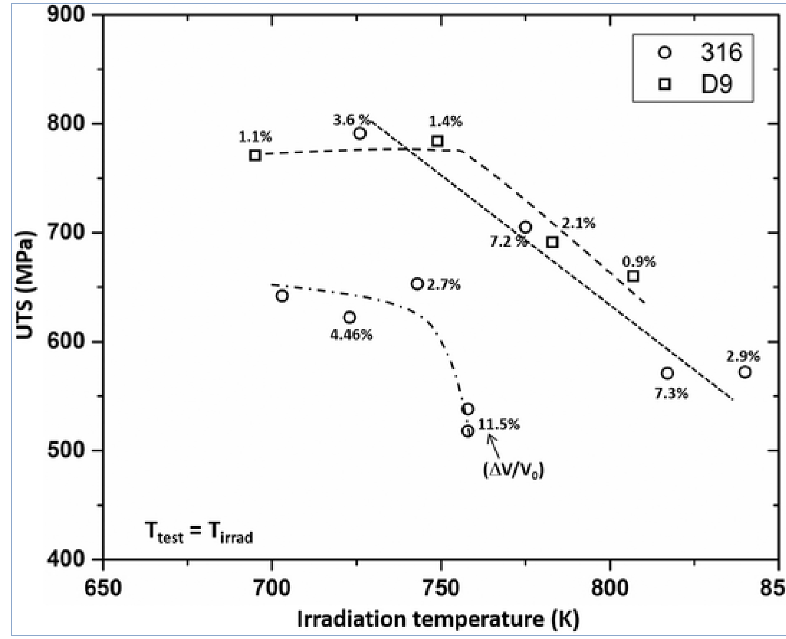
Comprehensive PIE of PFBR MOX fuel and D9 clad/wrapper indicated safe operation to rated burn-up of 100 GWd/t. FCMI and clad wastage due to FCCI could be life limiting precursors at higher burn-ups.

Fuel cladding behaviour SS316 and Alloy D9



Swelling of SS316 and Alloy D9 wrappers of FBTR at various dpa corresponding to fuel burn-up up to 155 GWd/t (SS316, carbide fuel) and 112 GWd/t (Alloy D9, MOX fuel).

Reddy *et al.*, DOI 10.1520/MPC20200208 (2022)



AISI 316 and D9 claddings irradiated in FBTR: UTS reduction at locations of high swelling

Karthik *et al.*, (2022) 10.1016/j.jnucmat.2022.153711

CHALLENGES
Thermal Vs Fast Reactor Fuel Reprocessing

Thermal Reactor fuel	Fast Reactor fuel
Low burnup (6700 MWd/ton)	**Very high** burnup (1,00,000 MWd/ton)
Long cooled (> 5 yrs)	**Short** cooled (<1yr)
Low specific activity (~300 Ci/kg)	**High** specific activity (~10,000 Ci/kg)
Low Pu content (0.3%)	**High Pu** content (>20%) 70% for FBTR carbide fuel, 44% for FBTR oxide fuel, 21% (inner core) and 28% (outer core) for PFBR fuel
Contact maintenance -> Cell entry feasible	**Remote** maintenance ->Cell entry generally **not** feasible
Dynamic containment is generally adequate	**Static containment** is required (Leak tight cells required for handling high Pu content)

Status of COmpact facility for Reprocessing of Advanced fuels in Lead cells - CORAL

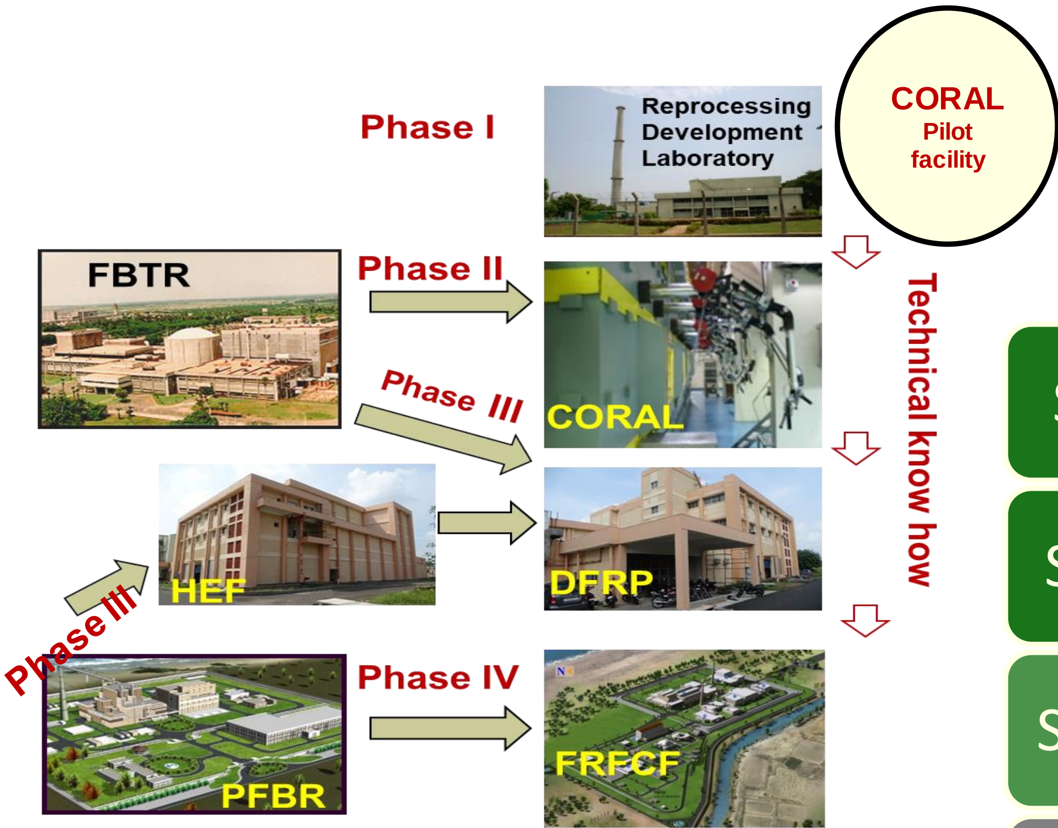
Closing the fuel cycle for FBTR

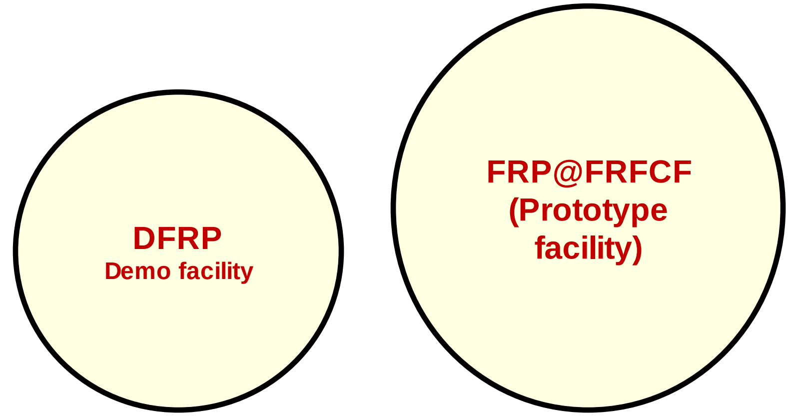
- Operational since 2003
- Objectives met
  - Development and optimisation of process for Fast reactor fuel reprocessing
  - Validation of several first-of-its-kind equipment for process, remote handling
  - Establishing analytical techniques
- Reprocessed Pu was refabricated into fuel and it is generating power in FBTR. Closure of FBR fuel cycle has been demonstrated.
- Successfully completed processing of 61 campaigns of FBTR fuel with maximum burn-up of 155 GWD/ton and continues to operate

CORAL operating area

Inside cell view of CORAL

INDIAN FAST REACTOR FUEL REPROCESSING PROGRAM





Stage-I	• Process and equipment development
Stage-II	• Pilot Plant (CORAL)
Stage-III	• Demonstration plant (DFRP)
Stage-IV	• Commercial scale plants (Starting with FRFCF)

Sodium bonded Metal Fuel Pin Fabrication Facility

- **A metal fuel fabrication Laboratory has been set-up with high purity inert atmosphere glove box train.**
-
- **Flow sheet and process equipment development including injection casting technology & fuel pin welding (T91) established.**
- **Sodium bonded metal fuel pins of U-6Zr, EU-6Zr, U-19Pu-6Zr, EU-23Pu-6Zr fuel pins fabricated and are currently under irradiation at FBTR.**

Sodium boned metal fuel pin fabrication facility

- **Slugs qualification (physical & chemical)**
- **Slug loading & settling**
- **End-plug welding & Post Weld Heat Treatment (PWHT)**
- **Sodium Bonding**
- **HLT and Pin qualification**

Pyrochemical Reprocessing

- Development of pyrochemical technique based on molten salt electrorefining is planned for reprocessing of spent fuel from metal fuel fast reactors.

- Electro-refining of Uranium (U) and U alloys small scale demonstrated.

- Electro-refining of irradiated U-Zr and U-Pu-Zr at small scale demonstrated at Hotcells.

- An engineering scale facility – few kg scale Pyro Process R&D Facility is set-up for scaled up pyroprocessing studies. Alloys of natural U containing surrogates for Pu and typical fission products will be used for electro-refining. Electro-refining of Uranium at small kg scale is under progress.

Way forward…

- Setting up of facility for fabrication & quality control of sodium bonded metal fuel pins and fabrication of qualified 1.0 m length fuel pin subassembly (37 fuel Pin) for irradiation at FBTR.

- Fabrication of subsequent test fuel pins of varying composition / enrichment to achieve target LHR.

- Establishing pyroprocessing facility for spent fuel of sub-assembly level & flow sheet for re-fabrication to close fuel cycle.

Three Phase Indian Fast Reactor Program

FBTR
- 40 MWth
- 13.5 MWe
- Loop type
- PuC – UC
- Design: CEA
- Since 1985

Phase – I
Successfully accomplished
2022

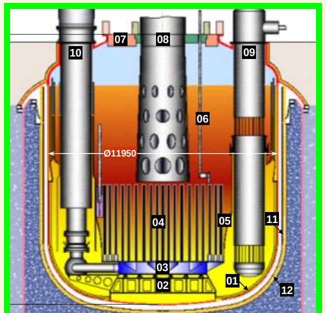
Reprocessing

PFBR
- 1250 MWth
- 500 MWe
- Pool Type
- UO_2-PuO_2
- Indigenous

Phase – II
Techno Economic Demonstration
In progress

2030

FBR-MOX
- 500 MWe
- Pool Type
- UO_2-PuO_2
- 1 twin unit at Kalpakkam
- Indigenous



Phase – III
Commercialisation

2047

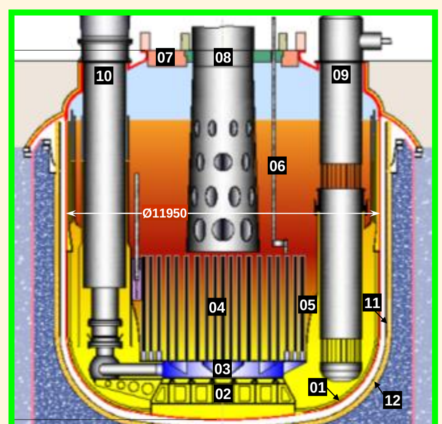
Metallic Fuel Fast Reactor Program-Way forward

Engineering Scale demonstration of
(a) sodium bonded metal fuel pin fabrication
(b) Pyrochemical processing
Pilot scale plants for both colocated at Kalpakkam

FBTR – 2 : Metallic fuel based reactor 100 /320 MWth at Kalpakkam

FBR-Metal

- 500/1000 MWe
- Pool Type
- Metallic fuel
- Serial constr.
- Indigenous



FBTR -2 by 2040 and subsequently after 2050 metal fuel based reactors

Thanks to Organising Committee – FR 22, IAEA, DAE

Thank You