India's Thorium Utilisation – Updated Plans

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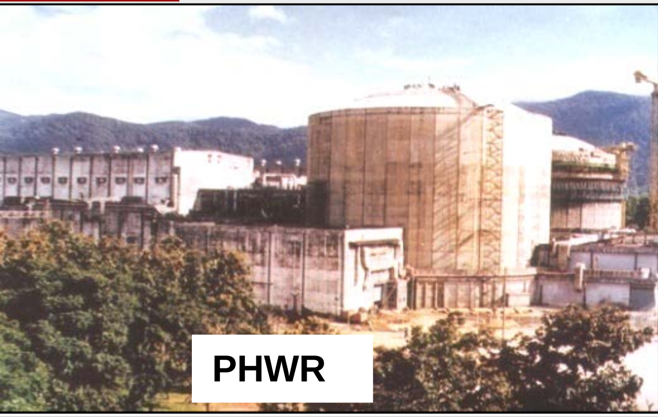
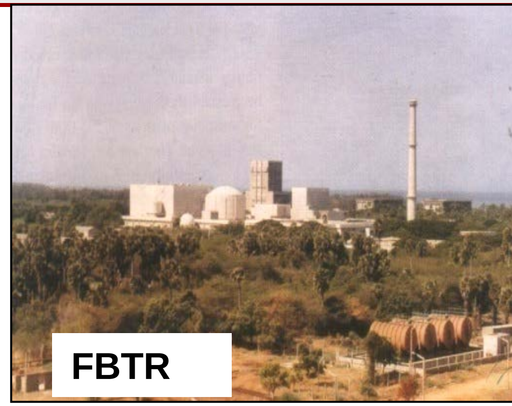
भाभा परमाणु अनुसंधान केंद्र
BHABHA ATOMIC RESEARCH CENTRE

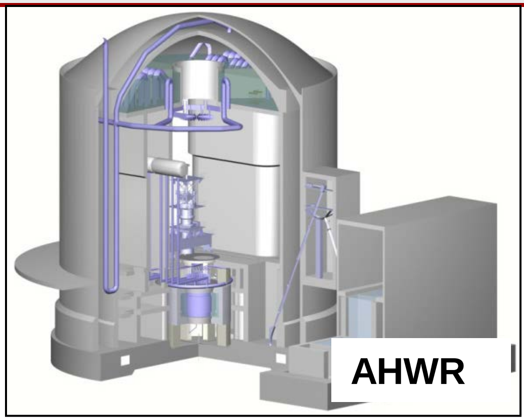
India has 11.93 million tonnes of monazite reserves containing about 1.07 million tonnes of thoria

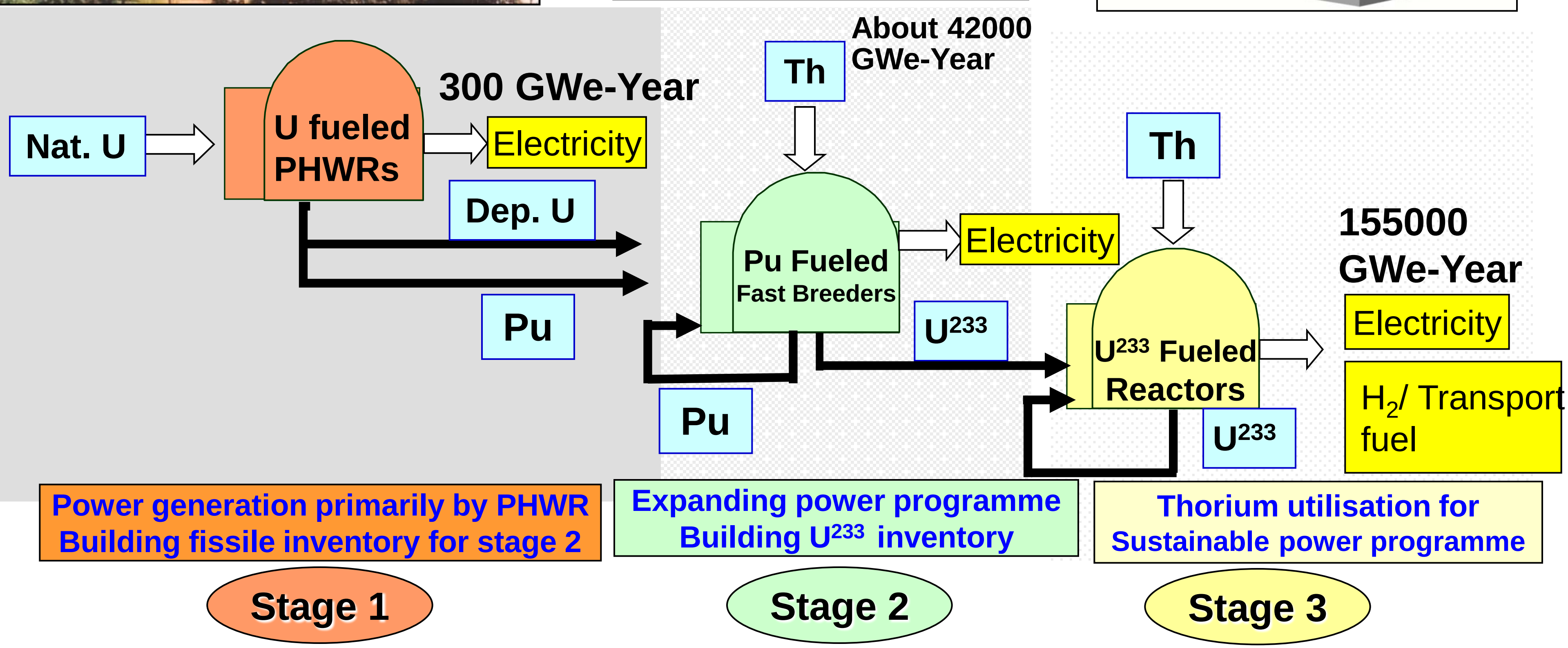
State	Monazite (Million tonnes)
Jharkhand	0.22
West Bengal	1.22
Odisha	2.41
Andhra Pradesh	3.72
Tamil Nadu	2.46
Kerala	1.90
Total	11.93

The goal of three stage Indian nuclear power programme is resource sustainability- Accordingly power generation in 3rd stage is predominantly dependent on thorium based fuel

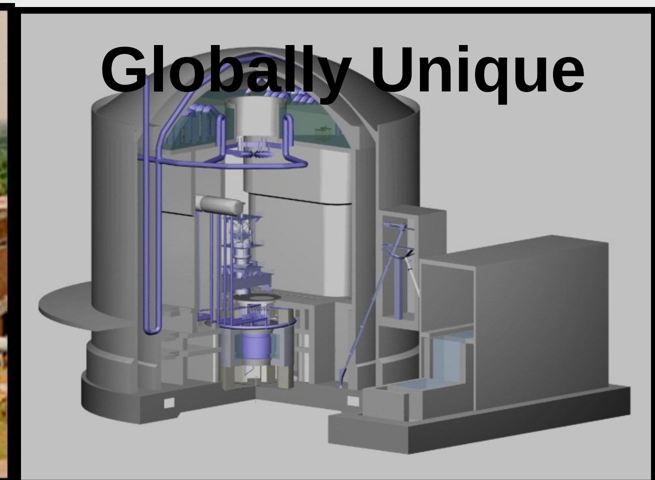








Current status of Indian three stage nuclear power programme



Stage – I PHWRs

- 18 – Operating
- 4 - Under construction
- Several others planned
- Scaling to 700 MWe
- Gestation period has been reduced
- POWER POTENTIAL ≅ 10 GWe

LWRs

- 2 BWRs Operating
- 1 VVER – operating at close to its full rated power
- 1 VVER- under commissioning

Stage - II

Fast Breeder Reactors

- 40 MWth FBTR - Operating since 1985 Technology Objectives realised
- 500 MWe PFBR- Under Construction
- TOTAL POWER POTENTIAL ≅ 530 GWe (including ≅ 300 GWe with Thorium)

Stage - III

Thorium Based Reactors

- 30 kWth KAMINI- Operating
- 300 MWe AHWR: Pre-licensing safety appraisal by AERB completed, Site selection in progress
  POWER POTENTIAL IS VERY LARGE
- MSBRs – Being evaluated as an option for large scale deployment
- Non-electric applications – HTRs for hydrogen production

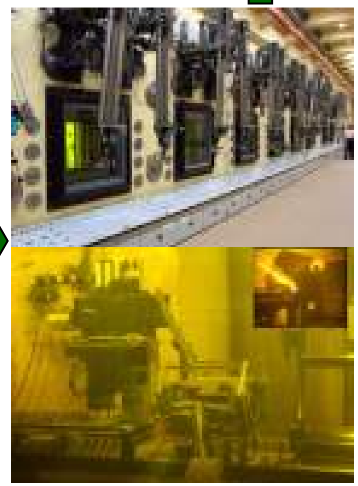
Brief history of thorium utilisation in India

Evolution of thorium fuel cycle development in India

Thorium extraction

Thoria pellets irradiation in research reactor

Use of thoria in PHWR



Dismantling of irradiated bundle and Post-irradiation examination

Irradiated fuel reprocessing and fabrication

KAMINI research reactor: ^{233}U-Al Fuel

AHWR critical facility

Thorium extraction

Beach Sand Minerals

ILMENITE & RUTILE	:	**TITANIUM (52%)**
ZIRCON	:	**ZIRCONIUM (3.2%)**
MONAZITE	:	**THORIUM** & **RARE EARTHS (1.13%)**
SILLIMANITE	:	**ALUMINIUM**

Mining of thorium

Beach sands of India contain rich deposits of monazite (thorium ore), ilmenite, zircon, etc. The total minerals established so far include about 8 million tonnes of monazite.

Composition of Monazite

Thorium as ThO_2	9%
Rare Earths as REO	58.5%
Phosphorus as P_2O_5	27%
Uranium as U_3O_8	0.35%
Calcium as CaO	0.5%
Magnesium as MgO	0.1%
Iron as Fe_2O_3	0.2%
Lead as PbO	0.18%
Insolubles	3%

Thorium Processing

Monazite, which contains about 9% ThO_2 and 0.35% U_3O_8, is a phosphate of thorium, uranium and rare earth elements. Thorium is extracted with trace uranium as by-product

Thoria Irradiation in Indian reactors

- **CIRUS**:
  - Thoria rods irradiated in the reflector region for ^{233}U production
  - Irradiations of (Th, Pu) MOX fuels in Pressurised Water Loop to burnup of 18 GWd/te.
- **PURNIMA-II (1984 -1986):** First research reactor using ^{233}U fuel.
- **PURNIMA-III (1990-93):** ^{233}U-Al dispersion plate type fuel experiments.

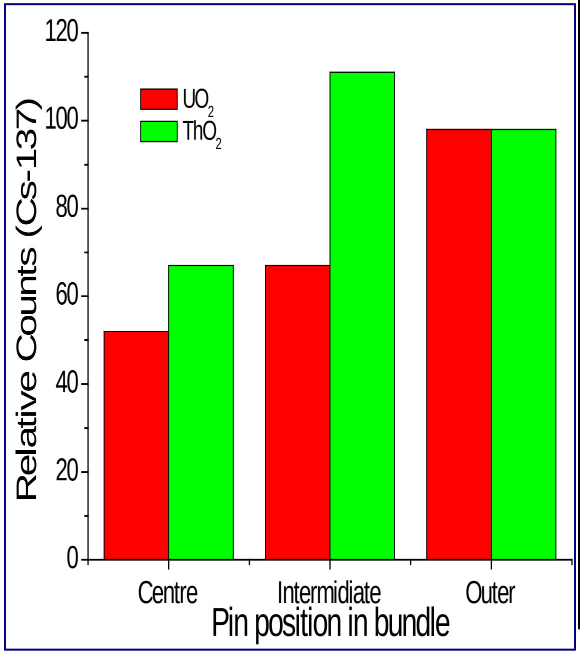
- **KAMINI**: Research reactor operating at 30 kW power, commissioned at Kalpakkam in 1996. Reactor based on ^{233}U fuel in the form of U-Al alloy, for neutron radiography.

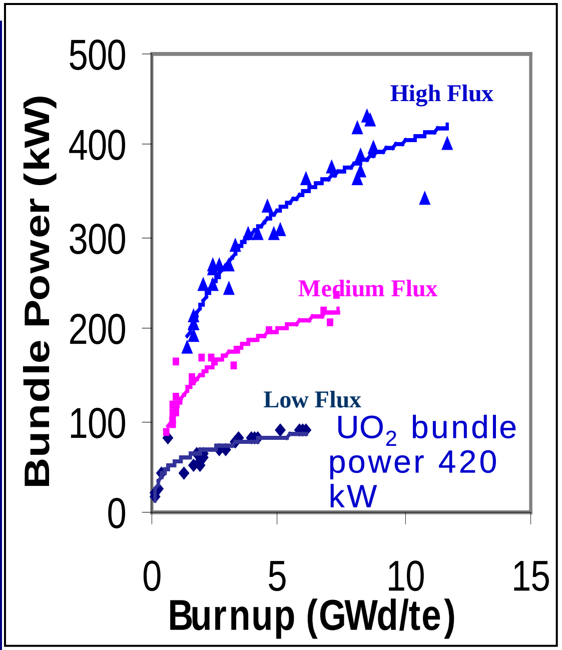
- Thoria bundles irradiated in the blanket zone of Fast Breeder Test Reactor (FBTR)
- ^{233}U-MOX fuel being irradiated in FBTR

- Three PHWR stations at Kakrapar, Kaiga and Rajasthan (units 3&4) have irradiated a total of 232 thorium bundles, to maximum discharge burnup of 14 GWd/te. The power produced by the bundle just before discharge (600 FPD) was about 400 kW.

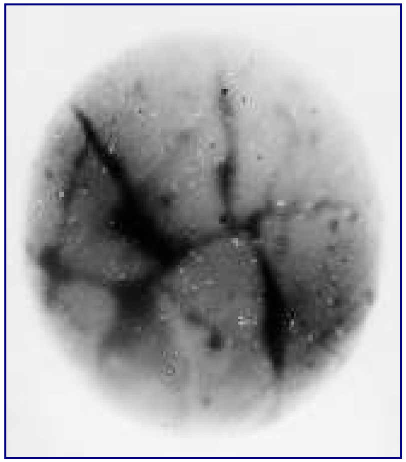
Post Irradiation Examination (PIE) of thoria fuel

- The PIE was carried out for one of the discharged bundles from Kakrapar unit-2, which had seen 508 full power days.
  - Dissolution tests done for uranium isotope composition and fission products and compared with theoretical evaluations.
  - Power distribution shows peaking at the outer pins for UO_2 bundle and at intermediate pins for thoria bundle.
  - Fission products (137Cs) migrate to thoria pellet cracks unlike upto periphery in UO_2 fuel.





α-Autoradiograph



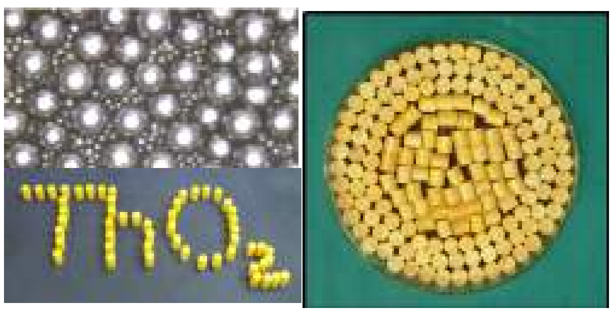
β, γ - Autoradiograph

PIE hot cell facility

Fission gas analysis set up

Fuel fabrication and reprocessing facility

- Experience with fabrication of thoria-based fuel
  - Thoria bundles for PHWRs.
  - Thoria assemblies for research reactor irradiation.
  - (Th-Pu) MOX pins for test irradiations.
- Fabrication was similar to that of UO_2 & (U-Pu) MOX



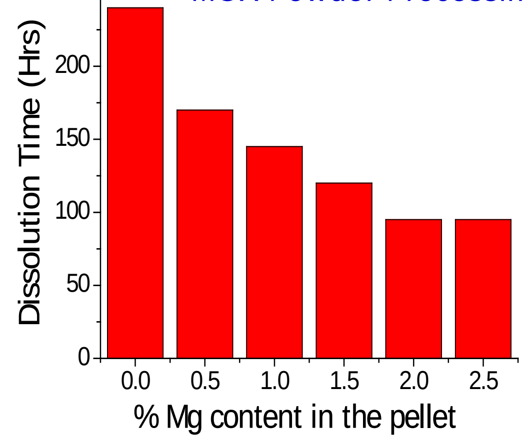
Thoria microspheres and ThO_2 Pellets fabricated for AHWR Critical Facility

Glove box and cask handling

Bundle dismantling

Impregnation setup

MOX Powder Processing



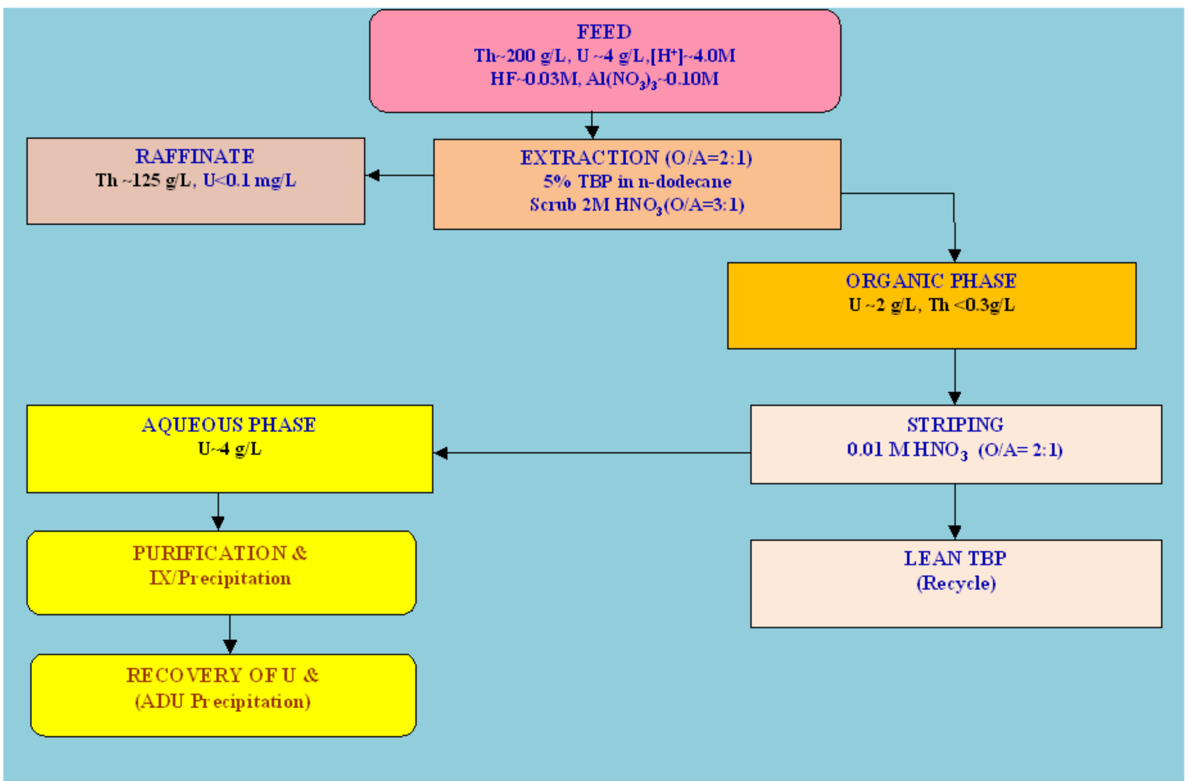
Thoria dissolution studies

Thorium fuel cycle technologies is relatively complex because of

- inert nature of thoria
- radiological aspects

Power Reactor Thoria Reprocessing Facility (PRTRF)

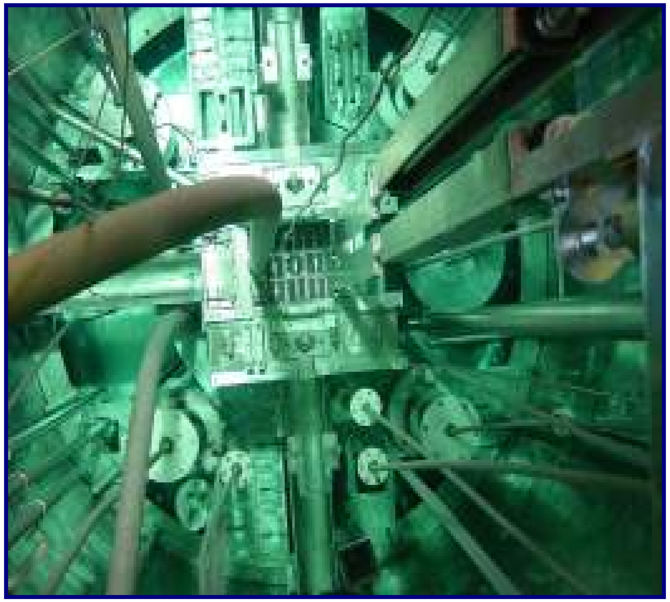
- PRTRF has been constructed for processing of irradiated zircoloy clad thoria bundles from PHWRs for separation of ^{233}U
- Several new technologies have been adopted in the flow sheet



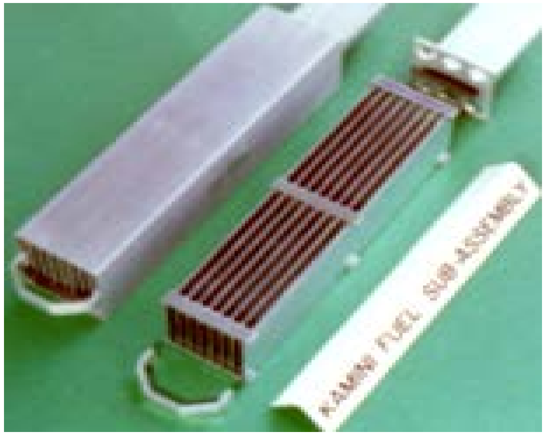
KAMINI Reactor for thorium based fuel physics studies

KAMINI (KAlpakkam MINI) is a 30 kWth, ^{233}U-Al alloy fuelled, light water moderated and cooled, special purpose research reactor. Beryllium oxide (BeO) is used as reflector and cadmium is used as absorber material in the safety control plates. The reactor functions as a neutron source with a flux of 10^{12} n/cm^2/s at the core center. Used for:

- Neutron radiography of both radioactive and non-radioactive objects (e.g. FBTR fuel pins)
- Neutron activation analysis.
- Carrying out radiation physics research,
- Irradiation of large number of samples, and
- Calibration and testing of neutron detectors.



U^{233} based KAMINI Reactor



KAMINI Fuel subassembly

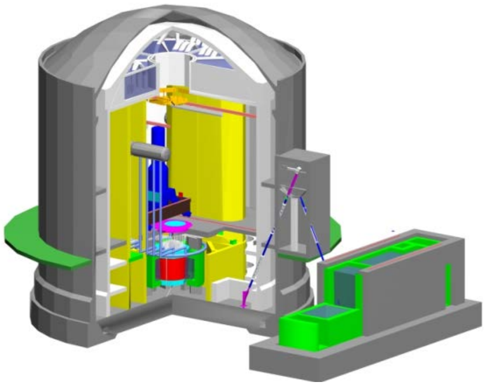
Future Thorium Utilisation: Pilot plant to Industrial Scale - Advanced Heavy Water Reactor

Indian Advanced Reactors Based on Thorium Fuel: AHWR & AHWR-LEU

- AHWR is a technology demonstration reactor designed to achieve large-scale use of thorium for power generation.
- Provides transition to 3rd stage of Indian Nuclear Power Programme.
- Addresses most issues required in advanced reactor designs
  - Enhanced safety, Proliferation resistance and sustainability
  - Minimize waste burden & Maximize resource utilisation
  - Economic competitiveness
  - Site in population centres
  - No emergency planning in public domain
- Some of the additional features for its wider acceptability are:
  - Reduced generation of Plutonium
  - Lower level of technological infrastructure should suffice
  - Low power unit
- AHWR and AHWR-LEU meets the above requirements

AHWR General Introduction

- AHWR is a 300 MWe, vertical, pressure tube type, boiling light water cooled, and heavy water moderated reactor.
- The reactor incorporates a number of passive safety features and is associated with a fuel cycle having reduced environmental impact. AHWR possesses several features, which are likely to reduce its capital and operating costs.
  - AHWR-LEU: Near term deployment
  - AHWR–Pu: for long term deployment

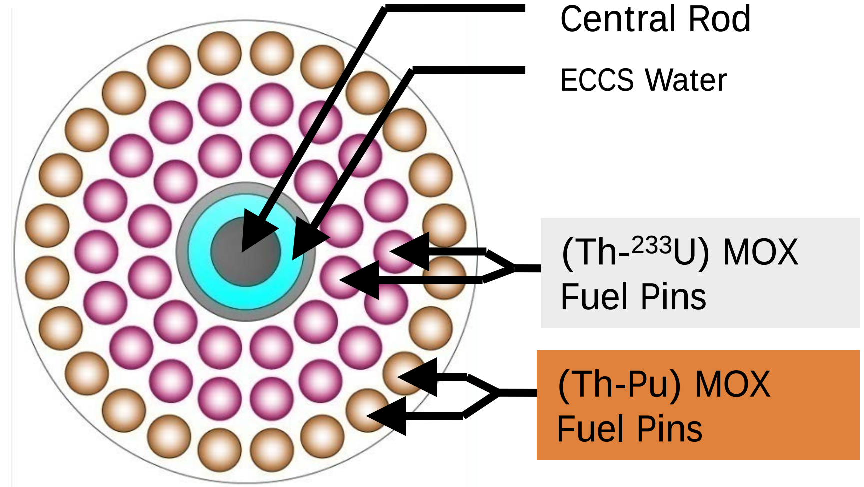


AHWR Schematic

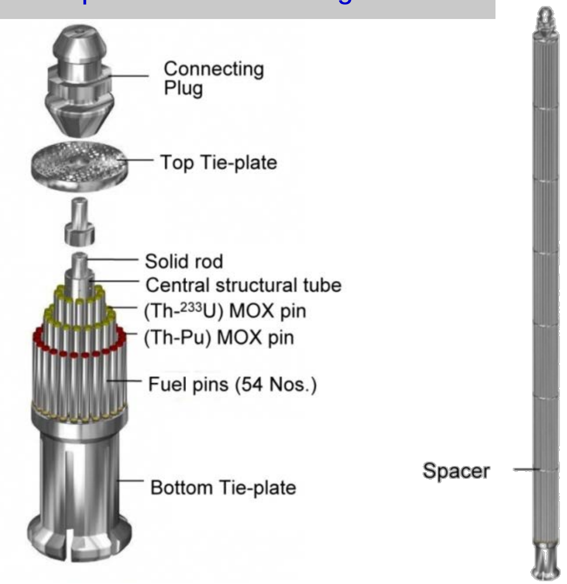
	AHWR - LEU	AHWR - Pu
Fuel Cluster configuration		
Inner circle	30% of $LEUO_2$ + ThO_2	(Th,^{233}U) MOX (3.0% ^{233}U)
Middle circle	24% of $LEUO_2$ + ThO_2	(Th,^{233}U) MOX (3.75% ^{233}U)
Outer circle	16% of $LEUO_2$ + ThO_2	(Th, Pu)MOX (3.25% Pu)

Selection of Fuel Material for AHWR (Th-^{233}U) MOX and (Th-Pu) MOX

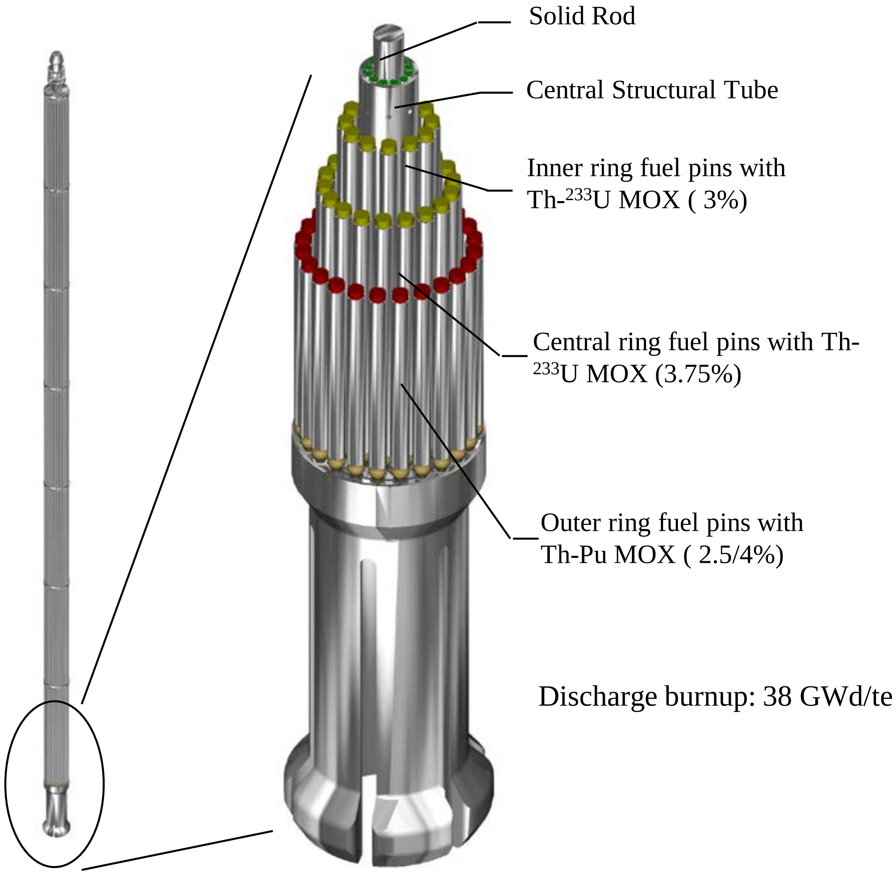
- Closed fuel cycle to maximise energy generation from thoria
- Recycling of self-generated ^{233}U and thoria
- External fissile feed of plutonium
- Initial Core
  - Fuel cluster has pins of (Th-Pu) MOX
- Equilibrium Core
  - Fuel cluster has pins of both (Th-Pu) MOX & (Th-^{233}U) MOX
- Features of fuel assembly:
  - 54 fuel pins arranged in three concentric rings
  - Outer ring has (Th-Pu)O_2 fuel
  - The inner and intermediate rings have (Th- ^{233}U)O_2 fuel



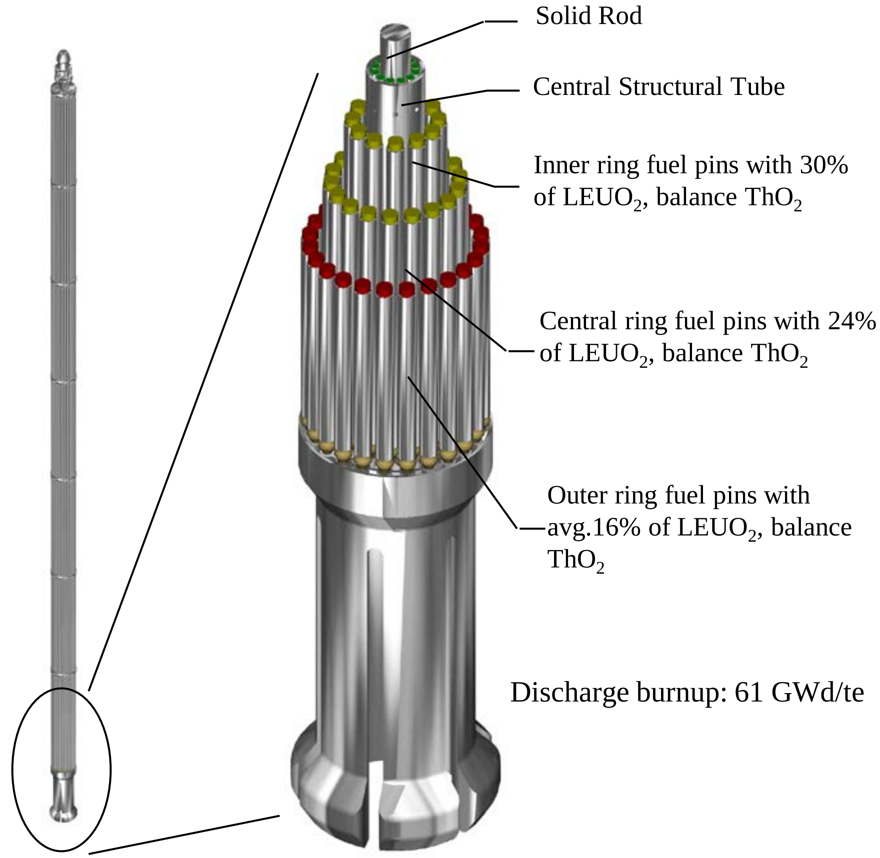
Fuel Cluster Cross Section
- 12 pins in the inner ring
- 18 pins in the intermediate ring
- 24 pins in the outer ring



AHWR-LEU: Advanced Heavy Water Reactor with LEU-Th MOX Fuel



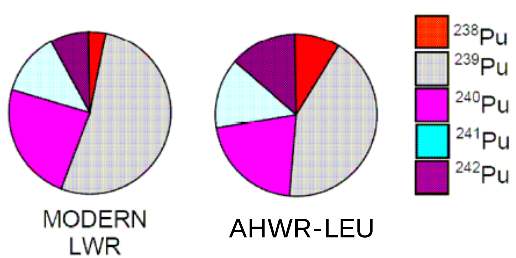
Fuel cluster of AHWR



Fuel cluster of AHWR-LEU

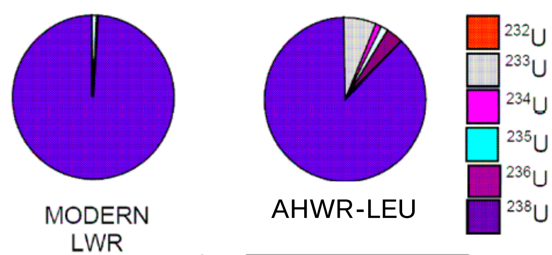
AHWR-LEU: Advanced Heavy Water Reactor with (LEU-Th) MOX Fuel

- Proliferation resistance
  - Use of LEU and thorium leads to reduced generation of Plutonium in spent fuel with lower fissile fraction and a high (~10%) fraction of ^{238}Pu
  - Fissile uranium in the spent fuel contains about 200 ppm of ^{232}U, whose daughter products produce high-energy gamma radiation
- Waste management
  - The AHWR-LEU fuel contains a significant fraction of thorium as a fertile host. Thorium being lower in the periodic table, the quantity of minor actinides is significantly reduced.



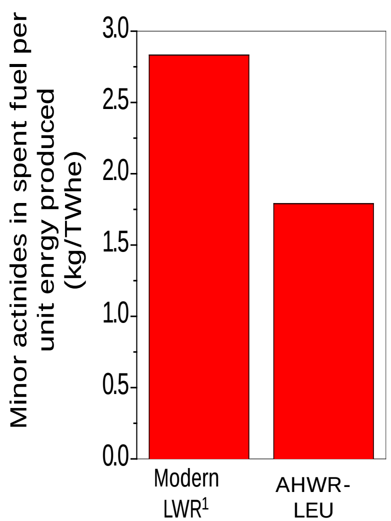
MODERN LWR		
^{238}Pu	3.50	%
^{239}Pu	51.87	%
^{240}Pu	23.81	%
^{241}Pu	12.91	%
^{242}Pu	7.91	%

AHWR-LEU		
^{238}Pu	9.54	%
^{239}Pu	41.65	%
^{240}Pu	21.14	%
^{241}Pu	13.96	%
^{242}Pu	13.70	%

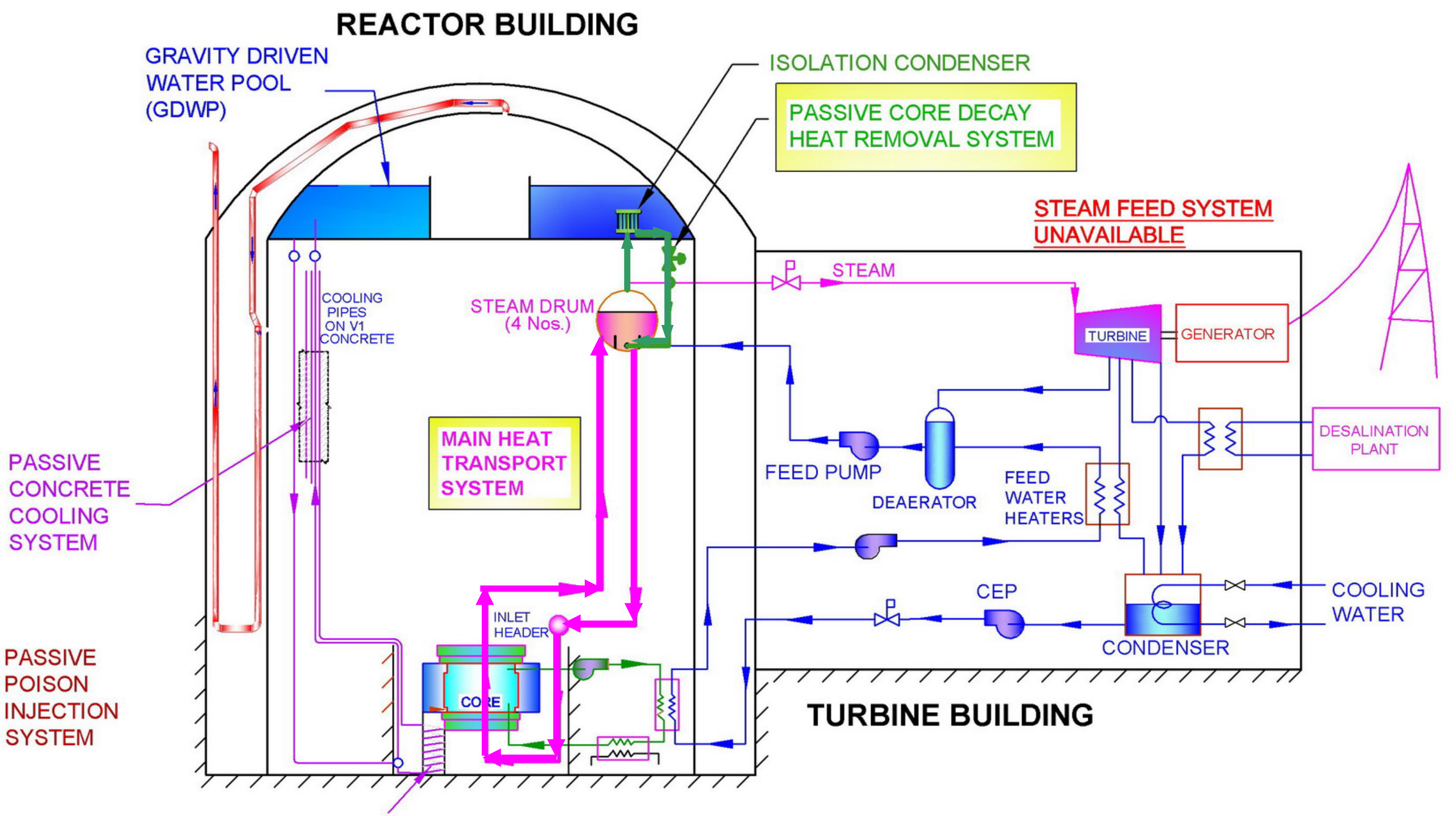


MODERN LWR		
^{232}U	0.00	%
^{233}U	0.00	%
^{234}U	0.00	%
^{235}U	0.82	%
^{236}U	0.59	%
^{238}U	98.59	%

AHWR-LEU		
^{232}U	0.02	%
^{233}U	6.51	%
^{234}U	1.24	%
^{235}U	1.62	%
^{236}U	3.27	%
^{238}U	87.35	%

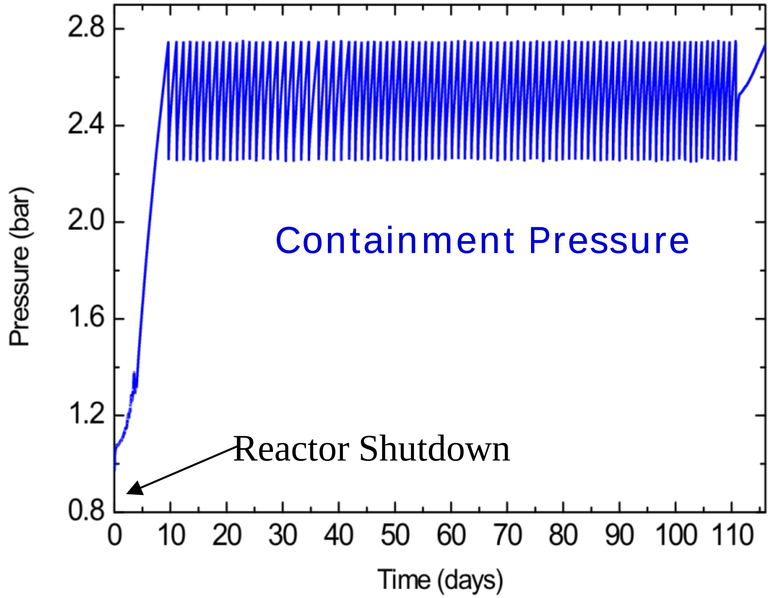


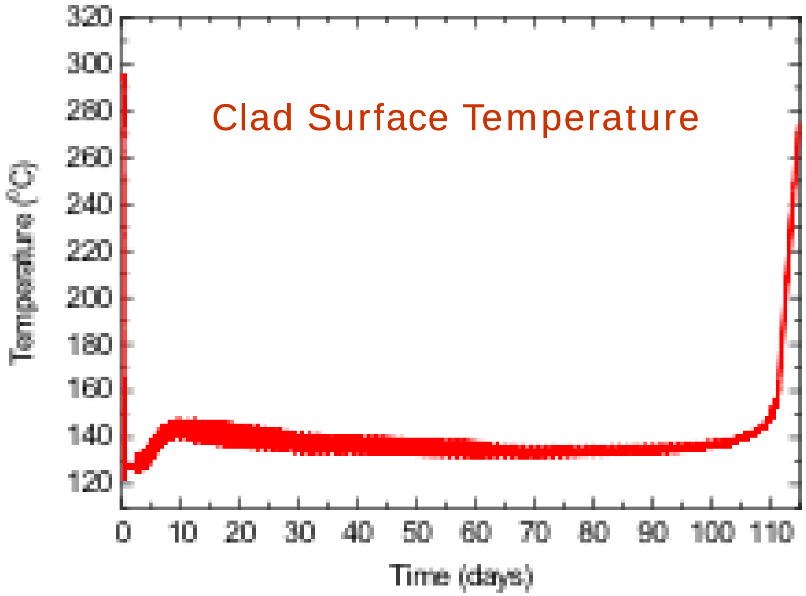
Heat removal from core under both normal full power operating condition as well as shutdown condition is by natural circulation of coolant.



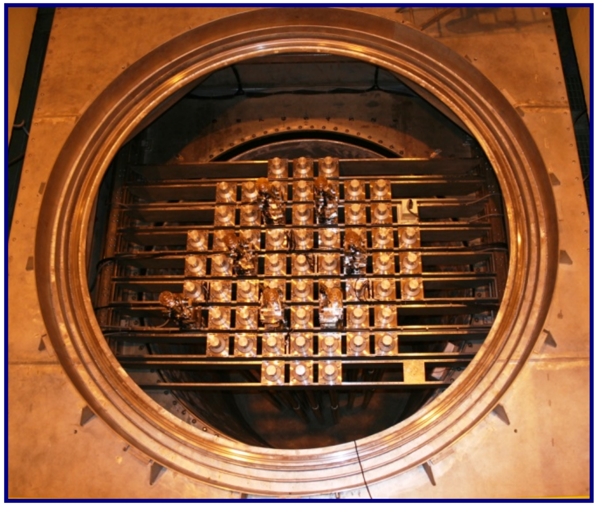
In a Fukushima type scenario, decay heat can be removed in AHWR without any electrical power, external source of water or operator action for 110 days

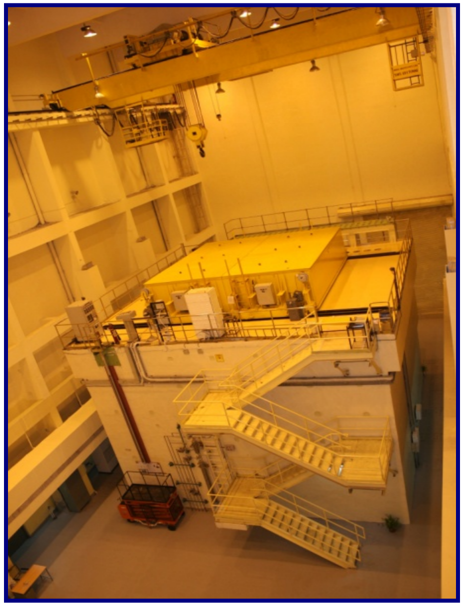
- Prolonged SBO in AHWR with Decay Heat Removal by Isolation Condenser System
- Scenario considered
  - Reactor trips on earthquake signal (t = 0)
  - GDWP water can remove decay heat ~110 days
  - Periodic venting of containment is required in this case
  - Venting starts at 2.75 bar of containment pressure and resets at 2.25 bar of containment pressure





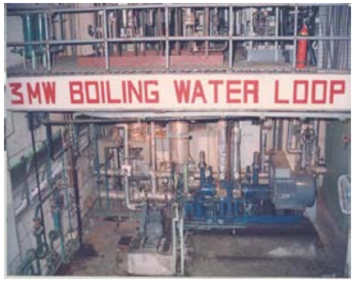
Critical Facility for AHWR



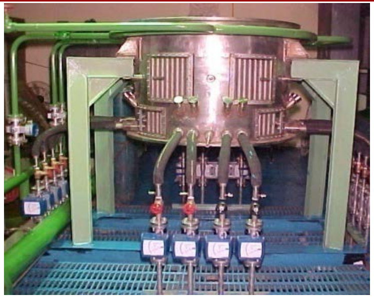


- AHWR Critical Facility has been established for conducting lattice physics experiments to validate AHWR physics calculations.
- Enough flexibility to arrange the fuel inside the core in a precise geometry at the desired pitch for facilitating study of different core lattices based on various fuel types, moderator materials and reactivity control devices.
- Criticality attained in April 2008.
- The fuel cluster of present reference core consists of 19-pin metallic natural uranium fuel with aluminum cladding.

Test Facilities for AHWR Design Validation



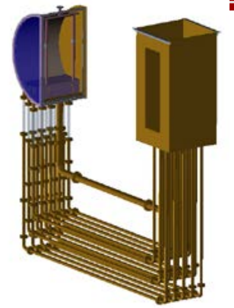
HPNCL



Moderator & liquid poison distribution

PCCTF

PCITF

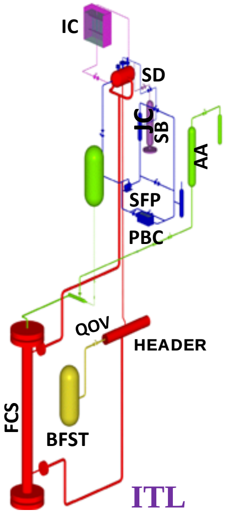


SDTF

ATTF, Tarapur

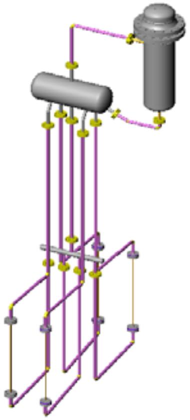
AHWR Critical Facility

Building housing ITL



ITL

FPTIL/CHIL



PCL

Several test facilities have been setup for AHWR design validation. Some of these are devoted to the study of specific phenomena. Major test facilities include 3 MW BWL, ITL and AHWR critical facility. The ATTF at R&D Centre Tarapur is the latest of these.

Long term Thorium Utilization: HTR & MSBR Programs

Major aim of the HTR programme is high efficiency nuclear hydrogen production

- Thermochemical water splitting
- High temperature steam electrolysis

In future, high temperature process heat could also be used for material processing

Indian High Temperature Reactor Programme

Compact High Temperature Reactor (CHTR)-Technology Demonstrator

- 100 kWth, 1000 °C, TRISO coated particle fuel
- Several passive systems for reactor heat removal
- Prolonged operation without refuelling

Status: Design of most of the systems worked out. Fuel and materials under development. Experimental facilities for thermal hydraulics setup. Facilities for design validation are under design.

Innovative High Temperature Reactor for Hydrogen Production (IHTR)

- 600 MWth , 1000 °C, TRISO coated particle fuel
  - Small power version for demonstration of technologies
- Active & passive systems for control & cooling
- On-line refuelling

Status: Optimisation of reactor physics and thermal hydraulics design, selection of salt and structural materials in progress. Experimental facilities for molten salt based thermal hydraulics and material compatibility studies set-up.

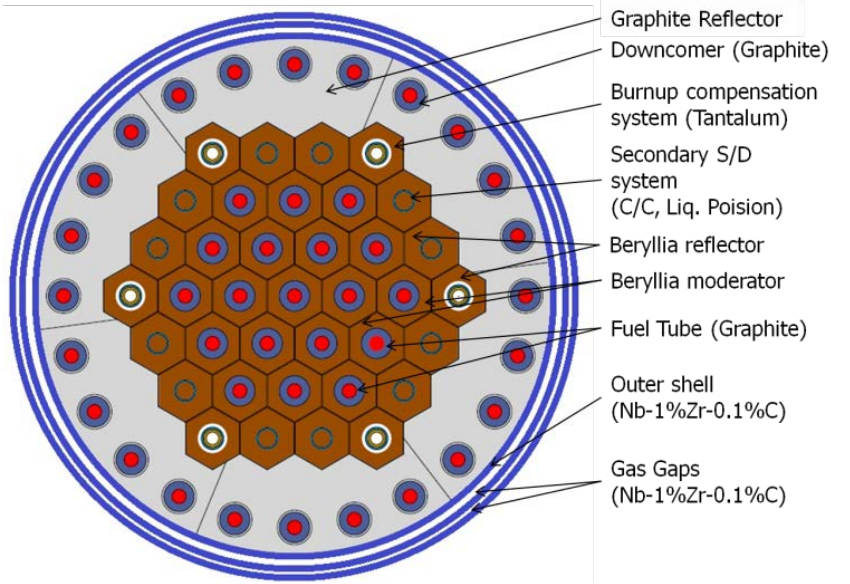
Indian Molten Salt Breeder Reactor (MSBR)

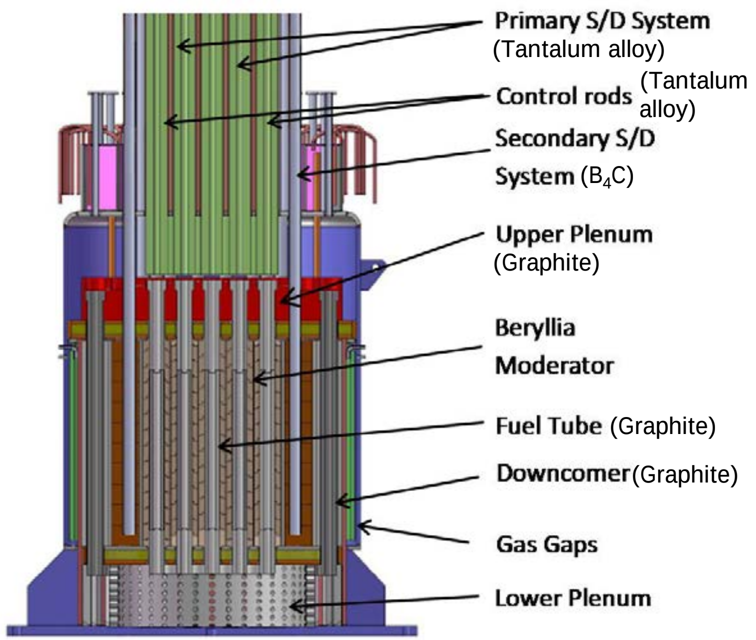
- Large power, moderate temperature, and based on ^{233}U-Th fuel cycle
  - Small power version for demonstration of technologies
- Emphasis on passive systems for reactor heat removal under all scenarios and reactor conditions

Status: Initial studies being carried out for conceptual design

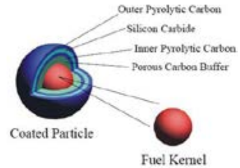
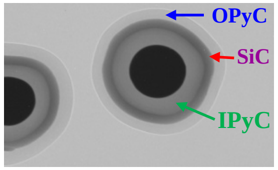
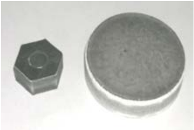
Advanced Reactors – Indian Initiatives
Compact High Temperature Reactor (CHTR)

- CHTR is a technology demonstrator with the following features:
  - Coolant exit temperature of 1000°C - Facilitate hydrogen production.
  - Compact: For use as nuclear battery in remote areas with no grid connection.
  - Fuel using ^{233}U-Th based on TRISO coated particle fuel with 15 years refuelling frequency and high burnup.
  - Ceramic core: BeO moderator, and graphite for fuel tube, downcomer tube and reflector
  - Coolant: Lead-Bismuth eutectic with 1670 °C as the boiling point.
  - Emphasis on reactor heat removal by passive systems e.g. natural circulation of coolant and high temperature heat pipes

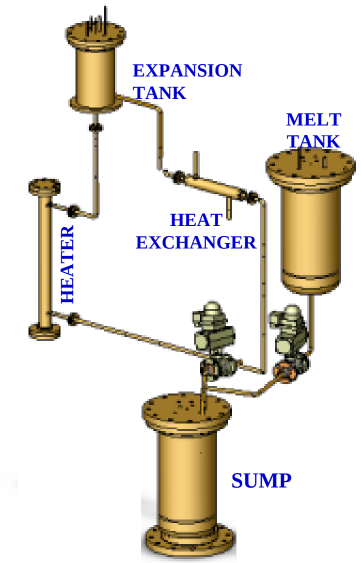




Major Research & Development Areas in Progress

Area of development	Status of development
High packing density fuel compacts based on TRISO coated particle fuel	Technology for TRISO coated particles & fuel compacts developed with surrogate material  Fuel compacts  
Materials for fuel tube, moderator and reflector ▪ Carbon based materials (graphite and carbon carbon composite) ▪ Beryllia ▪ Graphite	▪ Fuel tubes made using various techniques & materials including carbon-carbon composites ▪ Beryllia blocks manufacture technologies established  Fuel tubes  High density beryllia blocks  Graphite blocks
Metallic structural materials ▪ Nb-1%Zr-0.1%C alloy, Ta alloy	Indigenous development of alloy and manufacture of components for a lead-bismuth thermal hydraulic loop  Nb-1%Zr-0.1%C alloy
Thermal hydraulics of molten metal coolant	▪ LBE Test loops operated to validate design codes. ▪ Loop for studies at 1000 °C established.
Oxidation and corrosion resistant coatings	Technique for SiC coating on graphite and silicide based coating on Nb developed
Development of computer codes and analytical techniques for system design and analysis	Computer codes developed for reactor physics, thermal hydraulics, heat pipe design, TRISO particle fuel performance modeling.
High temperature instrumentation	Level probes, oxygen sensor, etc. developed for LBE coolant

Thermal hydraulic Studies for LBE Coolant

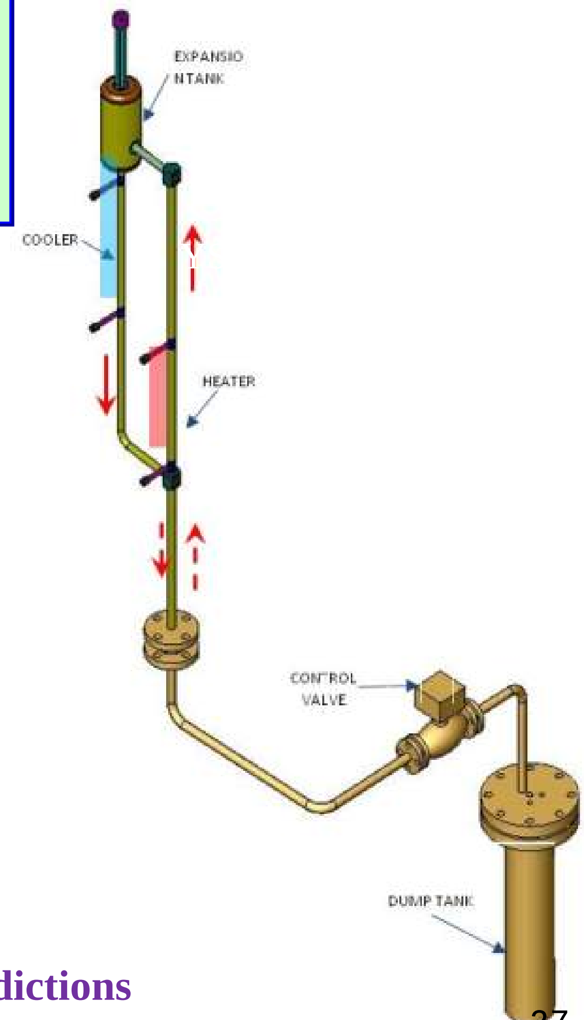


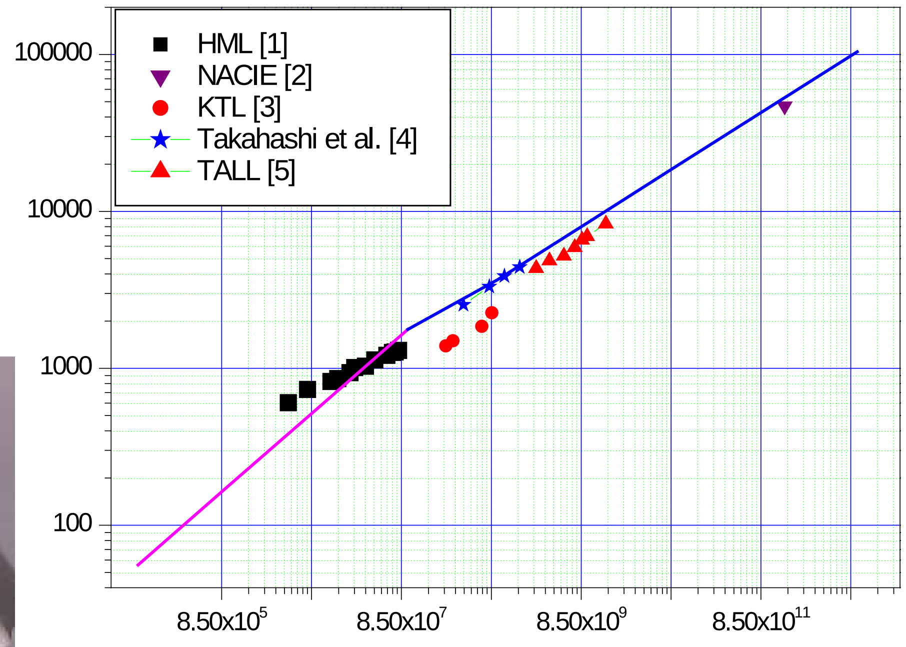
Liquid Metal Loop (Upto 550 °C)

Major areas of development

- Analytical studies and development of computer codes
- Liquid metal loop for experimental studies
  - Loop at 550 °C operating since 2009
  - KTL established and operated at 1000 ºC
- Steady state and transient tests carried out
- In-house developed code validated

Isometric of Kilo Temperature Loop (KTL)





Oxygen sensor

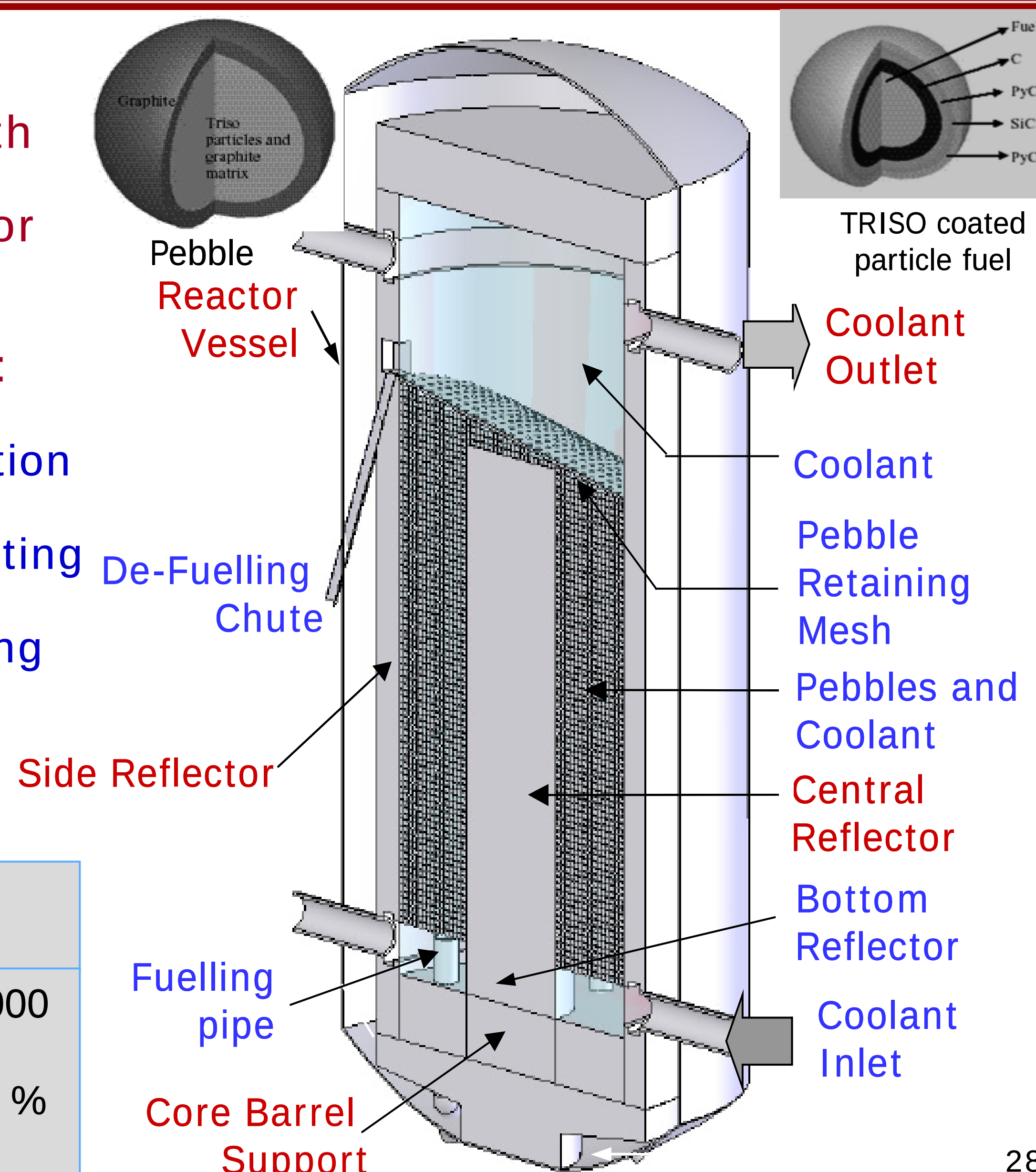
Steady state data compared with analytical predictions

Innovative High Temperature Reactor (IHTR) for commercial hydrogen production

- 600 MWth, 1000 °C, TRISO coated particle fuel
- Pebble bed reactor concept with molten salt coolant
- Natural circulation of coolant for reactor heat removal under normal operation
- Current focus on development:
  - Reactor physics and thermal hydraulic designs – Optimisation
  - Thermal and stress analysis
  - Code development for simulating pebble motion
  - Experimental set-up for tracing path of pebbles using radio-tracer technology
  - Pebble feeding and removal systems

•Hydrogen: 80,000 Nm^3 /hr
•Electricity: 18 MWe, Water: 375 m^3/hr

•No. of pebbles in the annular core ~150000
•Packing fraction of pebbles ~60%
•Packing fraction of TRISO particles ~ 8.6 %
•^{233}U Requirement 7.3 %



Molten Salt Natural Circulation Loop (MSNCL) and Molten Salt Corrosion Test Facility (MOSCOT)

Molten Salt used for experiment:

- 60:40 mixture of $NaNO_3$ and KNO_3
- Tests planned with FLiNaK

Experiments performed:

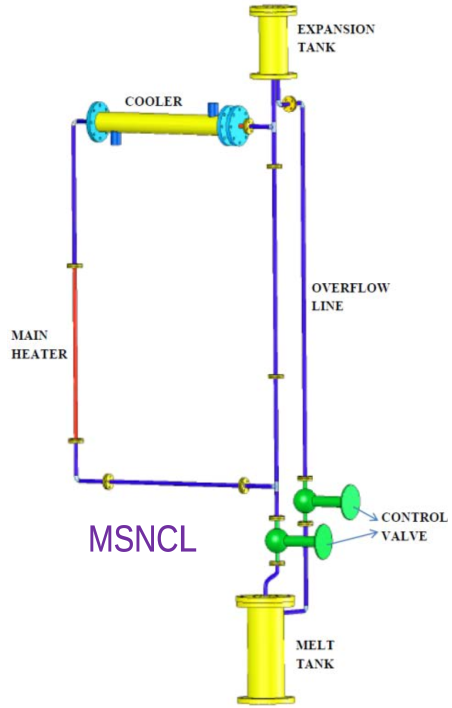
- Steady state NC performance at different powers

Operating conditions

- Cooling media: Air
- Max. power: 2kW

Loop Geometry

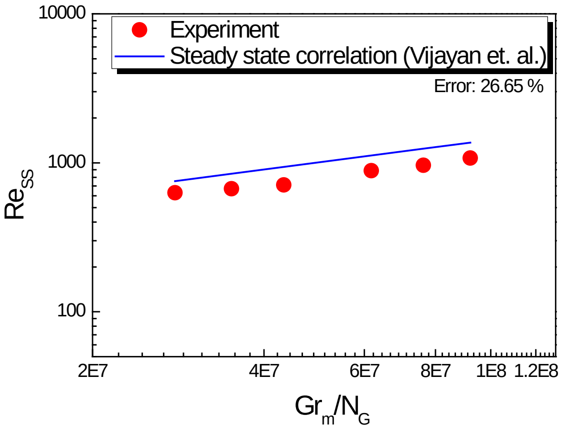
- Inside diameter: 14 mm
- Height of the loop: 2m
- Total circulation length: 6.8m



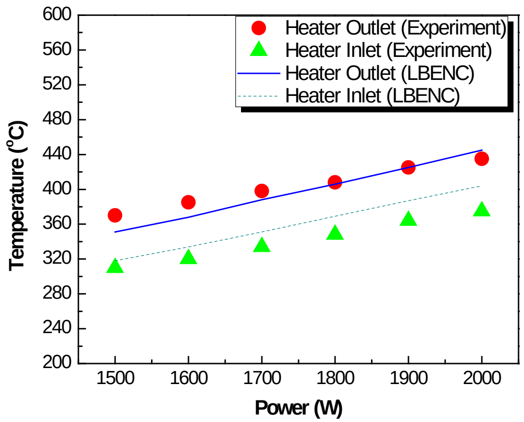
Photograph of MOSCOT Facility

Molten Salt used for corrosion study:

- Eutectic mixture of LiF-NaF-KF



Steady state performance of MS NCL



Steady state temperatures of MS NCL

Corrosion rate of different structural materials in 'mpy'

Temp °C	Inconel			Incoloy 800
	600	617	625	
550	6.2	10.0	-	17.4
600	12.2	18.2	6.1	30.1
650	25.9	22.7	5.0	31.2
700	25.4	33.9	71.3	45.4
750	15.8	97.9	127.6	33.2

Reactor for the 3^{rd} stage of INPP - Indian Molten Salt Breeder Reactor (IMSBR)

This concept is attractive to India for large scale deployment during the third stage of Indian Nuclear Power Programme since India has large thorium reserves with possibility of breeding ^{233}U in a self sustaining mode.

Major Design Guidelines for IMSBR

- Self sustaining in ^{233}U-Th cycle
- Inherent safety
  - Large scale deployment in third stage (may need to locate near population centres)
- No use of beryllium and beryllium based salts
  - Avoid chemical toxicity
  - Easier design process (no need to protect against beryllium in experimental facilities)
- Reduced fissile inventory
- Minimise waste generation
  - Avoid graphite
- Replaceability of in-core components

Evaluation of multiple conceptual design options in progress – thermal and physics design being investigated

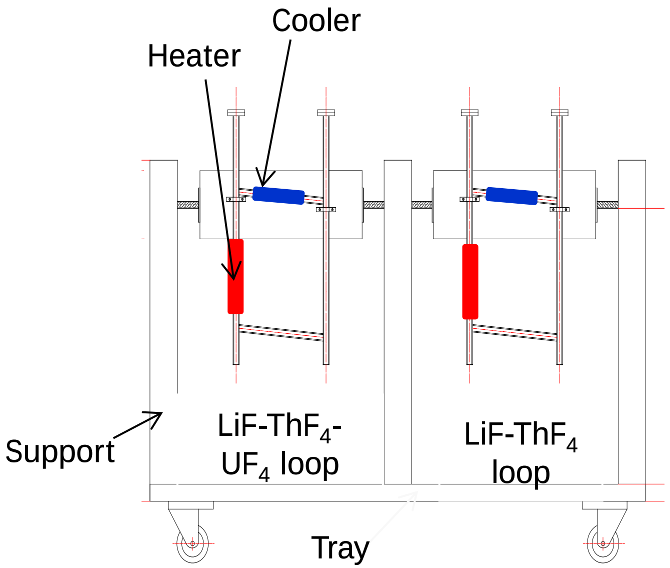
Broad areas of Research & Development Initiated for IMSBR

- Development of closely coupled neutron transport and CFD codes with capability to account for online reprocessing system
- Large scale salt preparation and purification
- Physical property characterisation for molten salts
- Metallic materials for use with molten salts up to 800 °C and qualification to meet ASME BPV Code, Section-III, Subsection – NH design rules
- Design rules for C/C composites for use in nuclear reactors
- Joining techniques for C/C composite to super alloys
- Batch mode offline reprocessing method, without requiring cooling of fuel salt
- Instrumentation for operation in high temperature, high radiation, molten salt environment
- Online chemistry control techniques
- Tritium capture
- Validation of computational chemistry methods in context of MSBRs
- Super critical CO_2 based power cycle

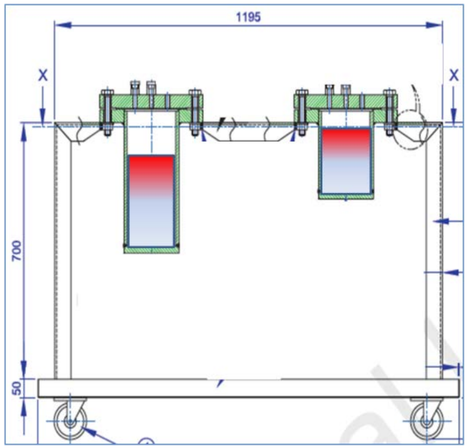
Thermal hydraulic & Corrosion studies for Fuel and Blanket Salts

Objective: To carry out natural circulation, heat transfer and static corrosion studies for structural materials

Fluid	$LiF-ThF_4-UF_4$ and $LiF-ThF_4$
Design Pressure	5 kgf/cm^2
Design Temperature	850°C
Material of construction	Nickel alloy
Salt Inventory/loop	2kg
Line Size	15NB
Main loop size	500mm x 500mm



Fuel and Blanket Salt NC Loops



Fuel and Blanket Salt corrosion facility

MSBRs have synergies with historical and ongoing R&D activities in BARC

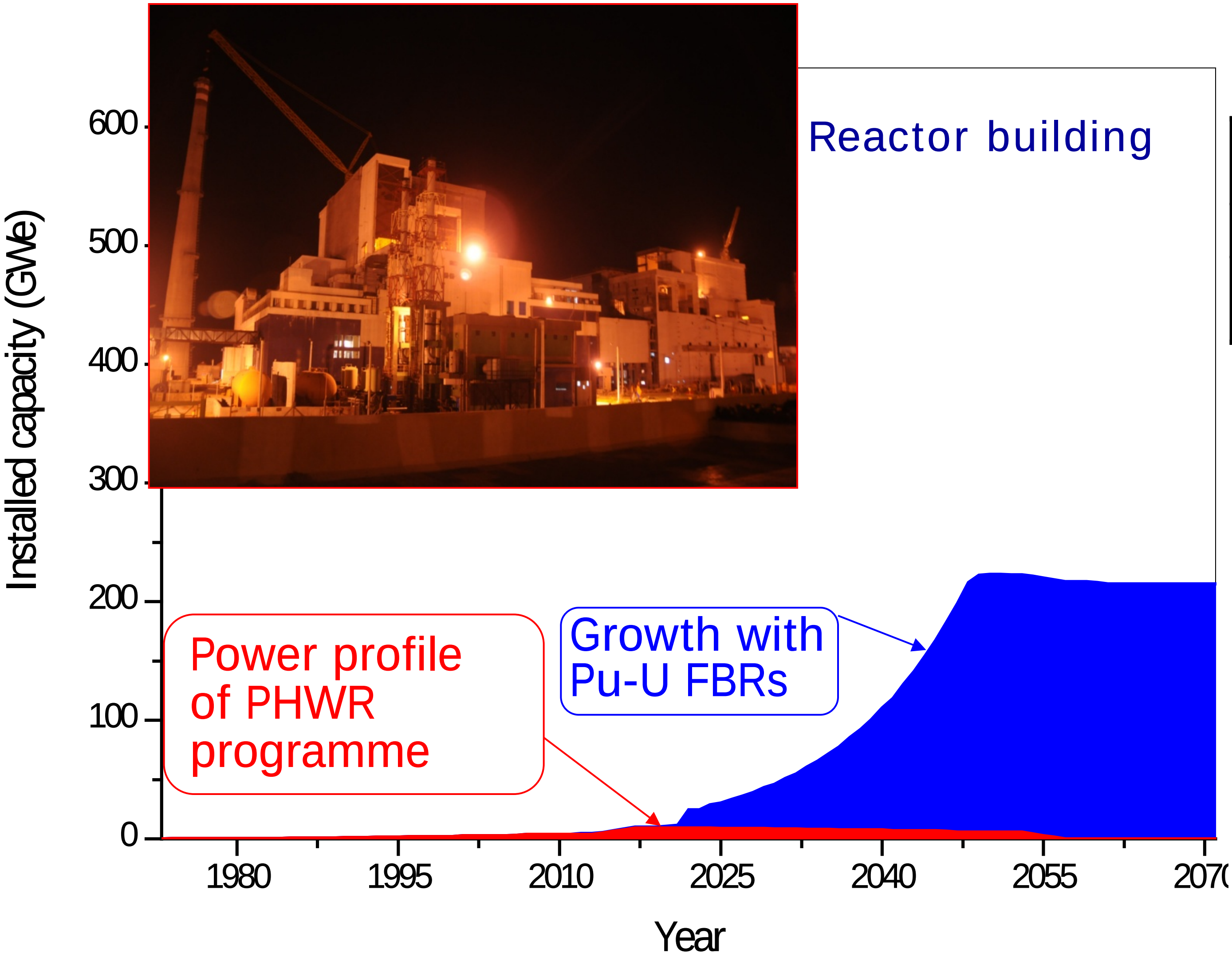
BARC-ORNL cooperation on MSBR	Preparation of pure ThF_4 and LiF_4 including development of equipment, Solubility of PuF_3 in LiF_4-BeF_2-ThF_4, Vapour pressure of materials of interest in MSBR
PURNIMA-II	^{233}U salt in fluid form
KAMINI	^{233}U fuel in plate form, beryllium oxide reflector
HFRR	High temperature coolant loop (LBE/Molten Salt) for material irradiation
AHWR	Technology demonstration for thorium fuel cycle
CHTR	TRISO coated particle fuel with beryllia and graphite moderator Development of HTR materials, coating & manufacturing technologies HTR design code establishment Handling (and purification) of molten lead and LBE at high temperatures Thermal hydraulics and instrument development for high temperature
IHTR	Molten salt cooled (salt preparation, purification, corrosion and redox control), Superalloy based high temperature materials for use with molten salts, Component development and instrumentation for high temperature
MSBR	^{233}U base fluid molten salt, super alloy compatible with molten salt, graphite (or beryllium) components, C/C composite components, SiC coating, salt preparation and purification, corrosion control and redox control, lead as coolant

Concluding Remarks

- Efficient utilisation of thorium is the cornerstone of the Indian nuclear programme
- Over a period of time, India has developed technologies for all the aspects of thorium fuel cycle
- AHWR is being developed as a technology demonstrator for industrial scale thorium fuel cycle
- India is developing the concept of thorium fuel based high temperature reactors dedicated for hydrogen production
- Large scale deployment of molten salt breeder reactors is envisaged during the third stage of Indian nuclear power programme
- Research & Development are being initiated to achieve designs with enhanced levels of safety
- In the long-term, using advanced nuclear energy systems, thorium offers sustainable large scale deployment of nuclear power

Thank you

Optimising the use of Domestic Nuclear Resources



PFBR (500 MWe)

Capital cost (Rs/kWe)	69840
UEC (Rs/kWh)	3.22
Construction period	7 years

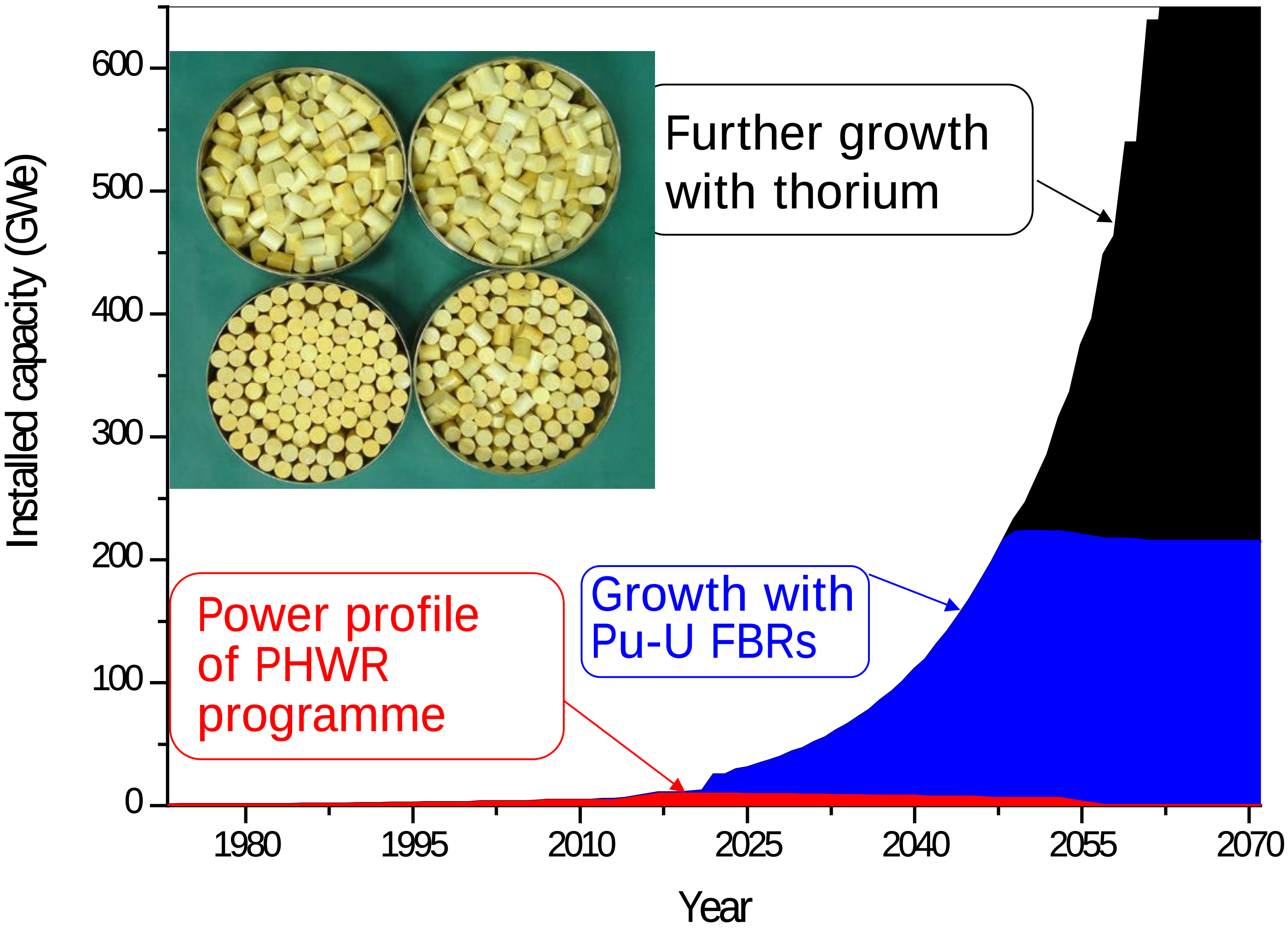
Project sanctioned in 2003

Further development being pursued to reduce doubling time and UEC

Russia is the only other country with a larger FBR under construction /operation

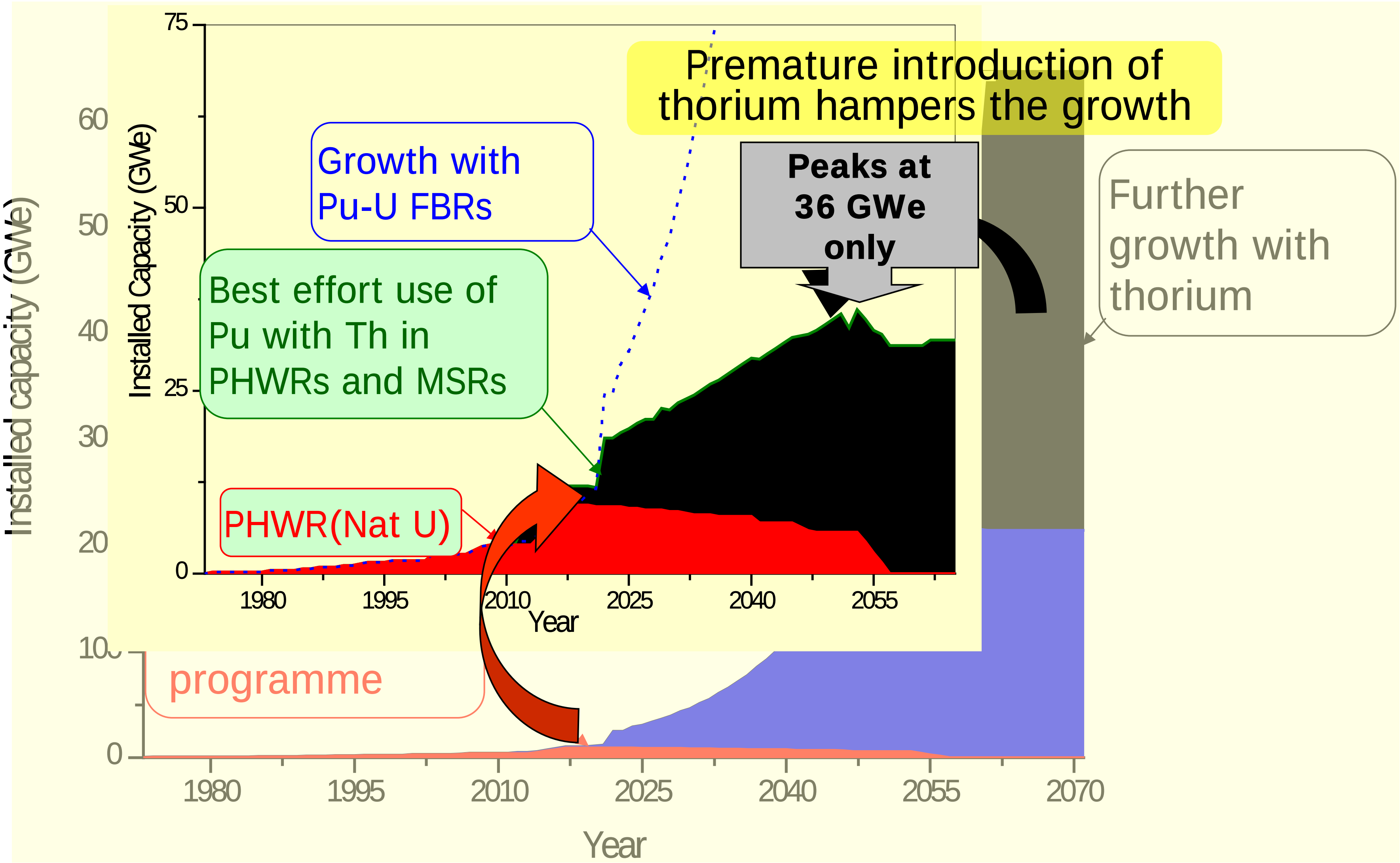
Results of a case study; assumptions 60000 te Uranium and short doubling time FBRs beyond 2021

Optimising the use of Domestic Nuclear Resources



Results of a case study; assumptions 60000 te Uranium and short doubling time FBRs beyond 2021

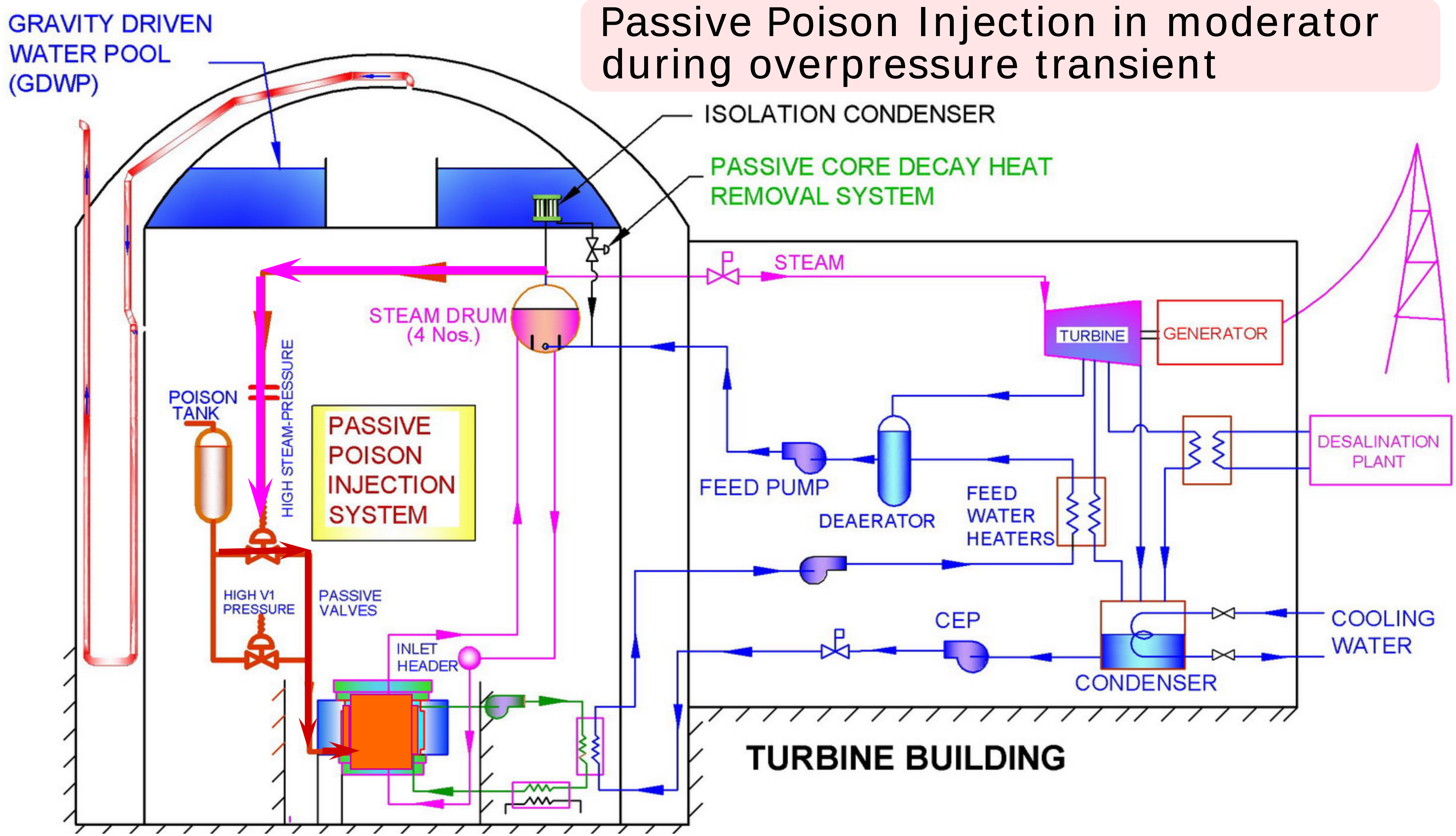
Optimising the use of Domestic Nuclear Resources



Results of a case study; assumptions 60000 te Uranium and short doubling time FBRs beyond 2021

Passive safety features of AHWR – a schematic of passive poison injection system

Passive Poison Injection in moderator during overpressure transient

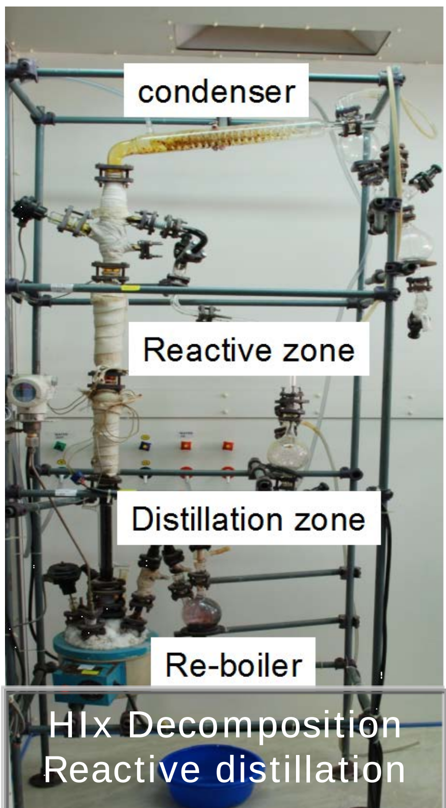


Passive Poison Injection System actuates during very low probability event of failure of wired shutdown systems (SDS#1 & SDS#2) and non-availability of Main condenser

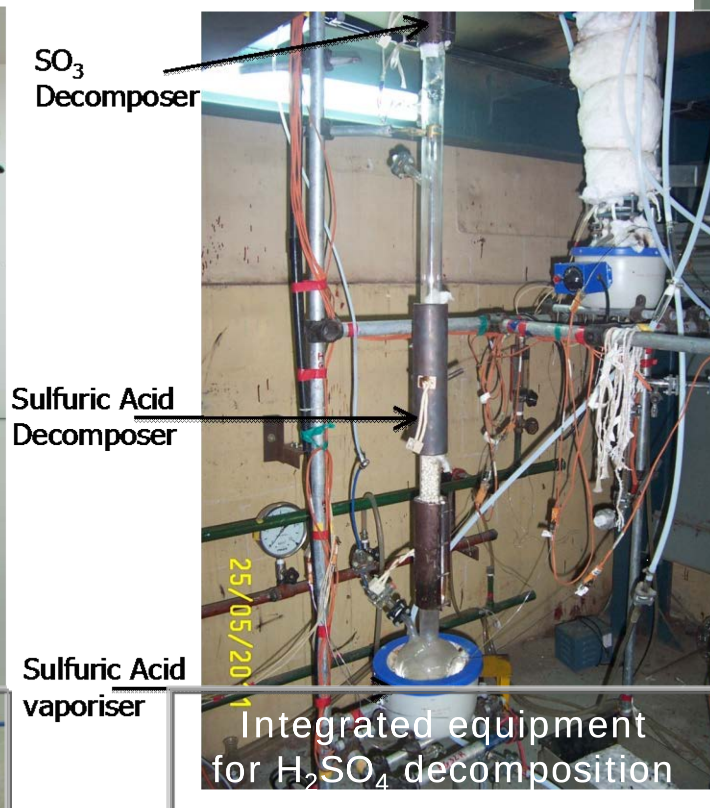
Development of hydrogen production technologies
Thermo chemical processes

- For **Bunsen section** After demonstrating feasibility in glass system, **metallic system studies** are being started
- For **HIx section**-Demonstration of reactive distillation, and development of catalysts
- For **sulfuric acid section, catalyst was developed** - Evaluation is in progress
- A **glass system** for demonstration of feasibility and stability of **closed loop operation is in operation**.

Metallic equipments of Bunsen section of S-I process



HIx Decomposition Reactive distillation



Integrated equipment for H_2SO_4 decomposition

Glass closed loop S-I process set up