Overview of the Thorium Programme in India

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  - Built research reactors based on U^{233} fuel
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  - Indian HTR Programme
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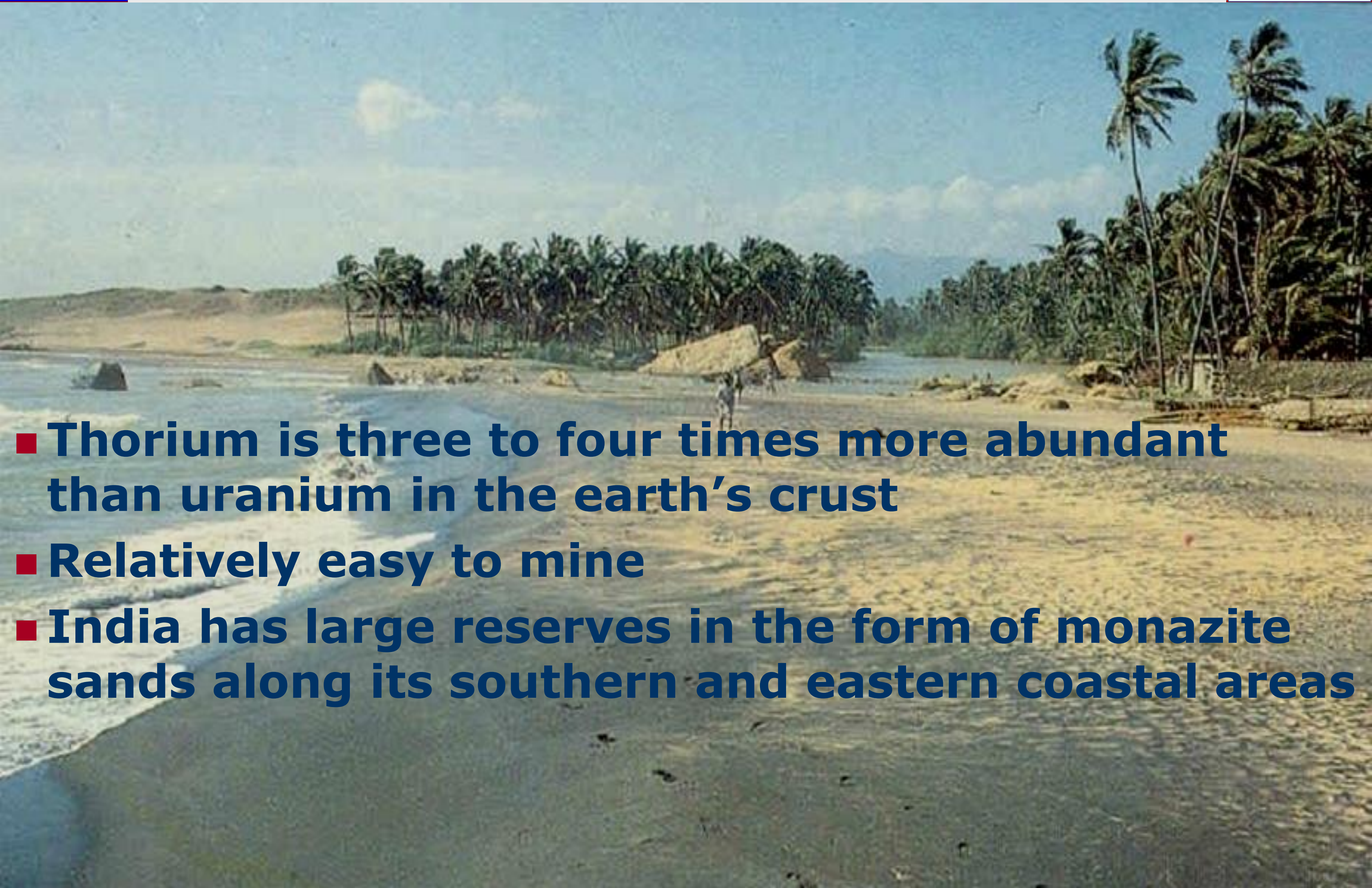
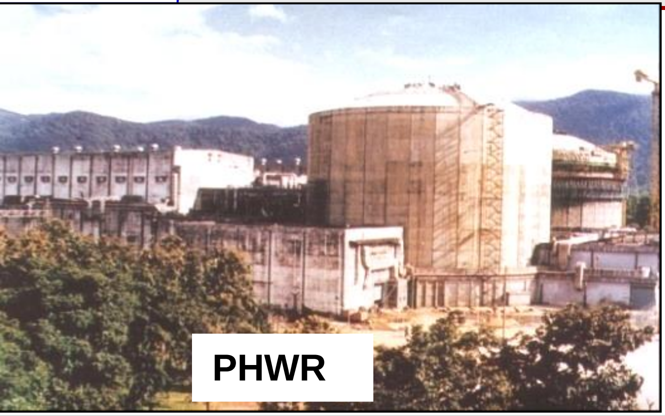
Introduction

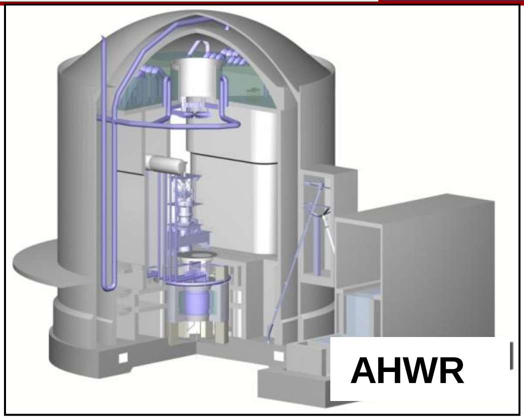


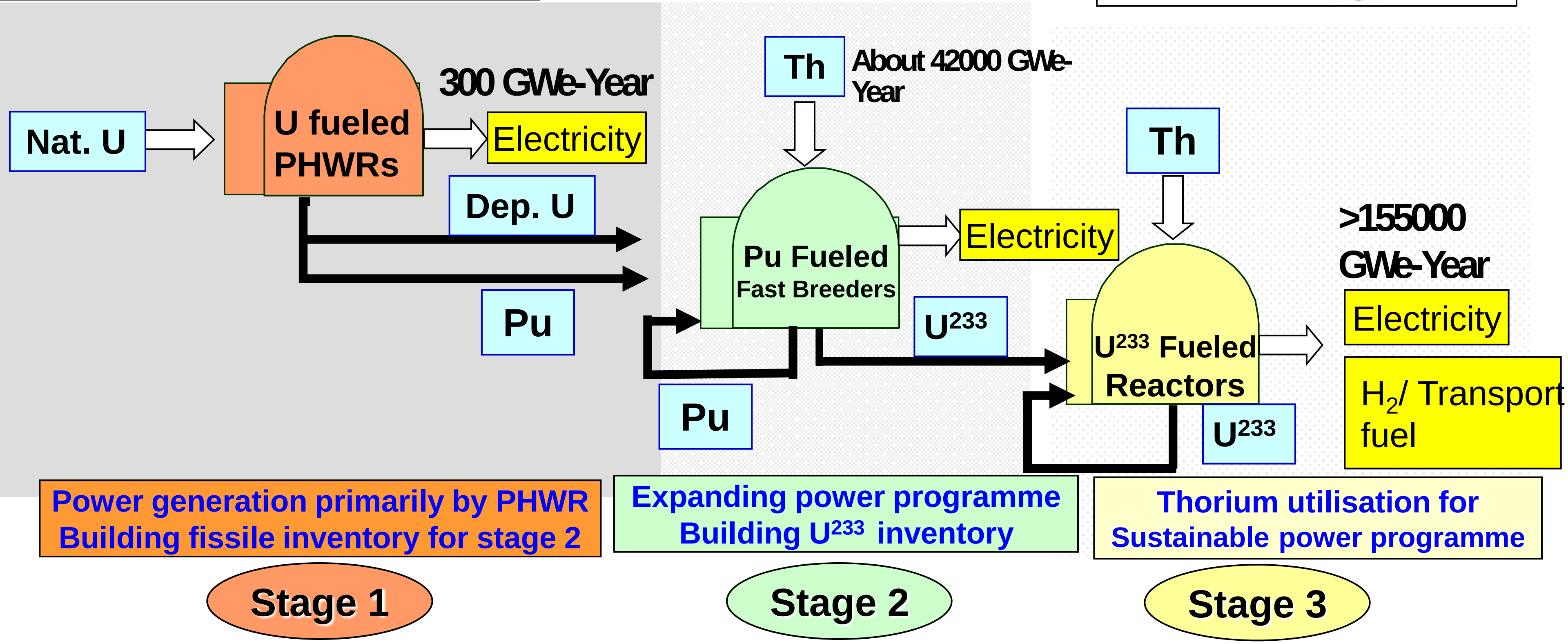
- **Thorium is three to four times more abundant than uranium in the earth's crust**
- **Relatively easy to mine**
- **India has large reserves in the form of monazite sands along its southern and eastern coastal areas**

Relevance of thorium to long term Indian nuclear power programme

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

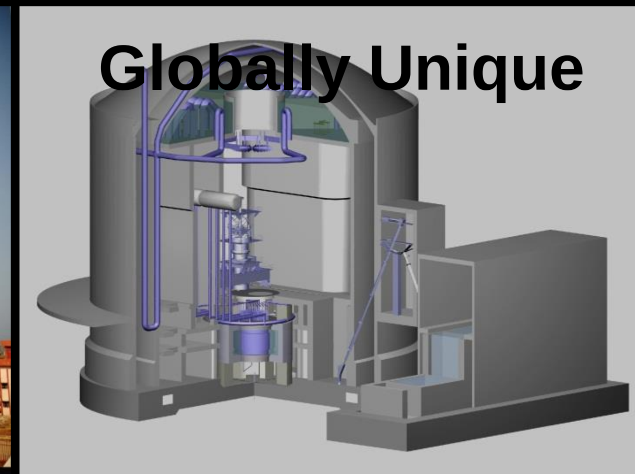






Current Status of Indian Three Stage Nuclear Power Programme

Overall view of Turbine & Nuclear Island Connected Building



Stage – I PHWRs

- 18 – Operating
- 4x700 MWe - Under construction
- Several others planned
- Gestation period has been reduced
- POWER POTENTIAL ≅ 10 GWe

LWRs

- 2 BWRs Operating
- 1 VVER– Achieved criticality (July, 2013) & commercial operation (Oct. 2013)
- 1 VVER- under construction (~96%)

Stage - II

Fast Breeder Reactors

- 40 MWth FBTR - Operating since 1985 Technology Objectives realised
- 500 MWe PFBR- Under Construction >95% complete
- TOTAL POWER POTENTIAL ≅ 42,000 GWe-year

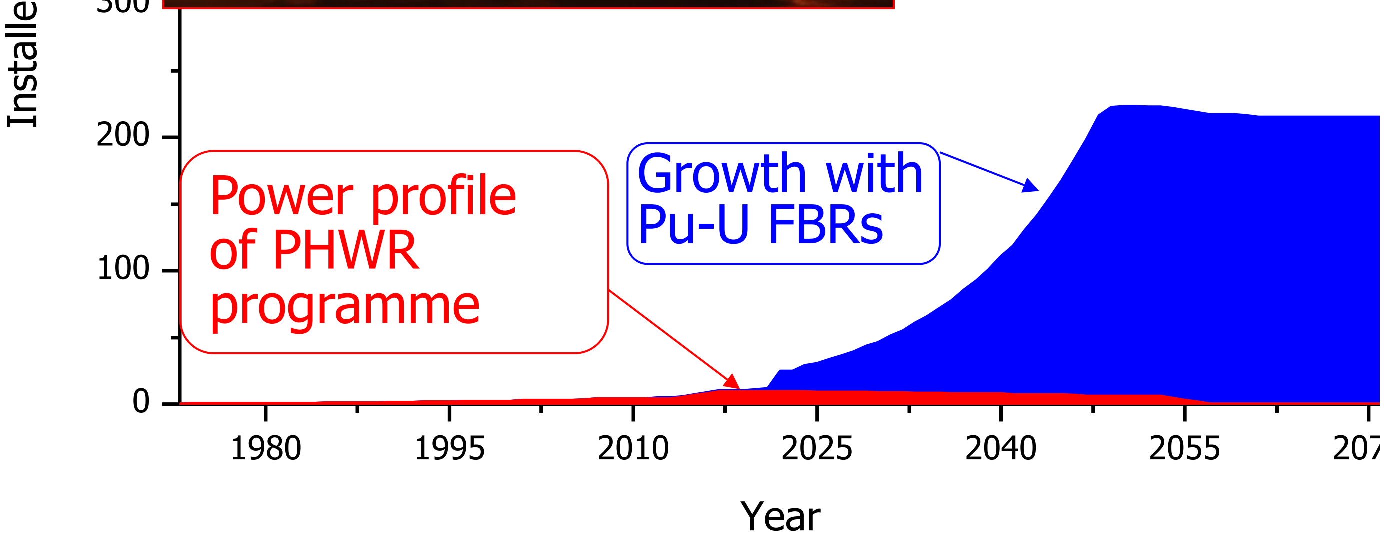
Stage - III

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

Optimising the use of Domestic Nuclear Resources

Reactor building



PFBR (500 MWe)

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

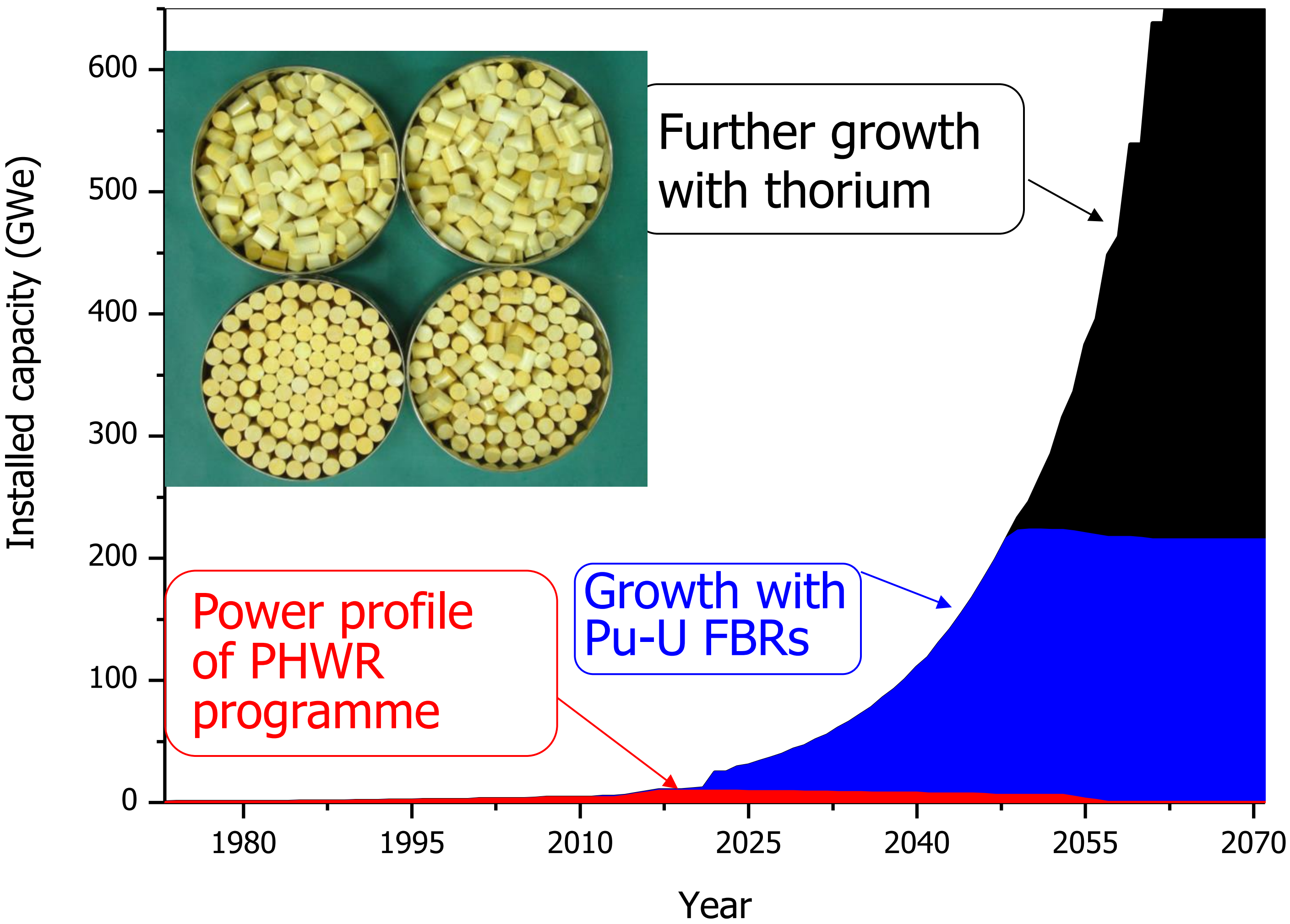
Project sanctioned in 2003

Further development being pursued to reduce doubling time and UEC

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

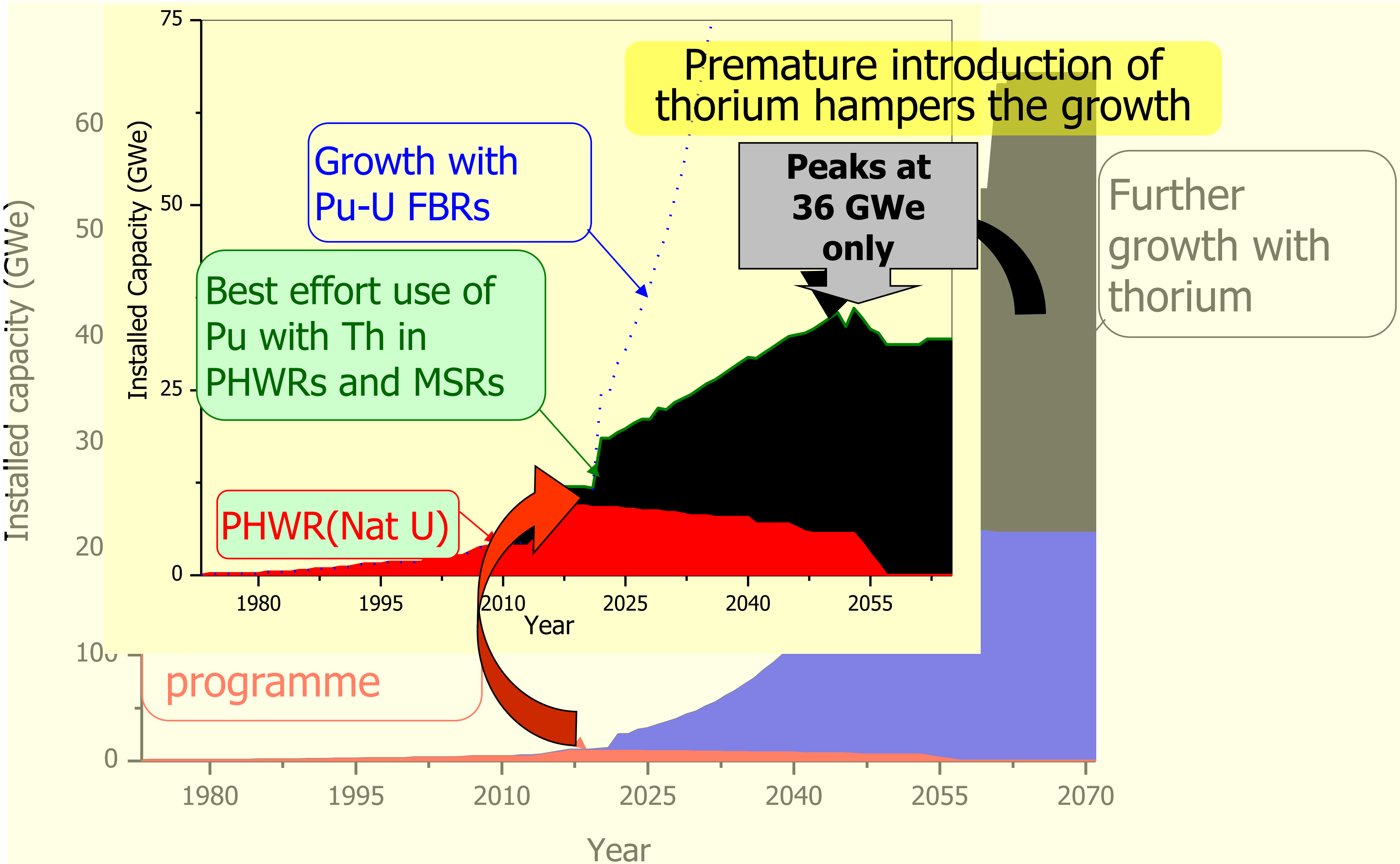
Results of a case study; assumptions 60000 te Uranium and short doubling time FBRs beyond 2021

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

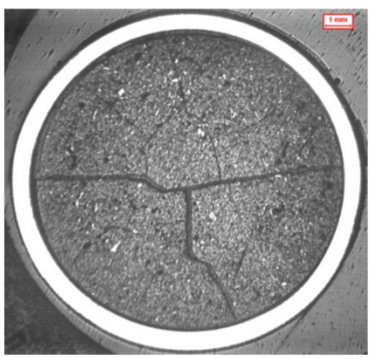
A Road Map for Deployment of Thorium Based Reactors

- Premature deployment of thorium leads to sub-optimal use of indigenous energy resource.
- Necessary to build-up a significant level of fissile material before launching thorium cycle in a big way for the third stage
- Incorporation of thorium in the blankets of metallic fuelled fast breeder reactors – after significant FBR capacity built-up
  - Full core and blanket thorium FBRs only after U-shortage felt
- Thorium based reactors expected to be deployed beyond 2070
- AHWR – Thorium fuel cycle demonstrator by 2022
- Surplus ^{233}U formed in these FBRs could drive HTRs including MSBRs

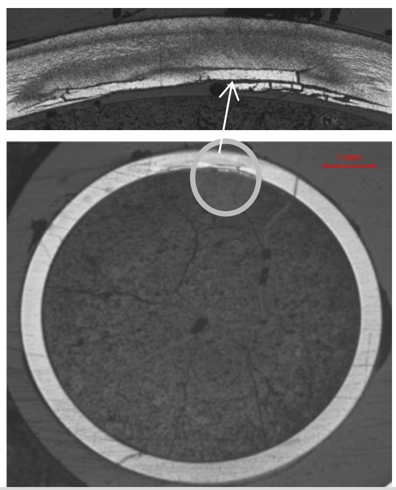
Important attributes and advantages of thorium fuel cycle

ThO_2 - Physical and Chemical properties

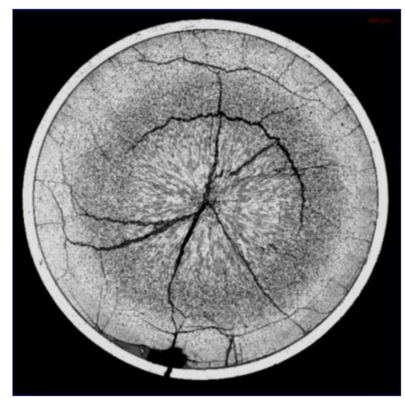
- Relatively inert. Does not oxidise unlike UO_2, which oxidizes easily to U_3O_8 and UO_3. Does not react with water.
- Higher thermal conductivity and lower co-efficient of thermal expansion compared to UO_2. Melting point 3350 °C as against 2800 °C for UO_2.
- Fission gas release rate one order of magnitude lower than that of UO_2.
- Good radiation resistance and dimensional stability



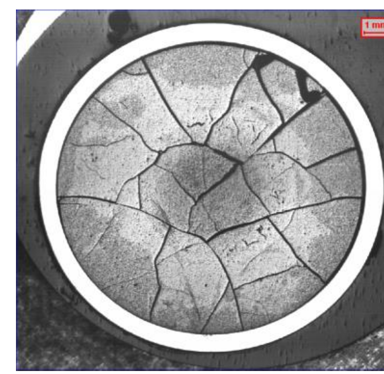
ThO_2-4%PuO_2



Failed ThO_2-4%PuO_2

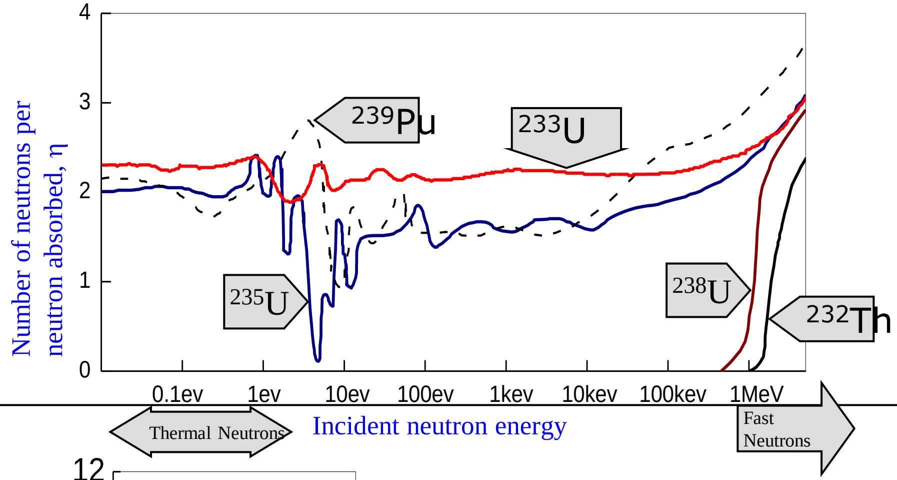


Failed UO_2



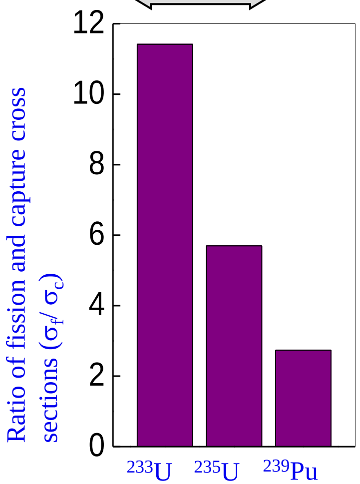
UO_2-4%PuO_2

Advantages of ^{233}U-Thorium: Important Neutronic Characteristics

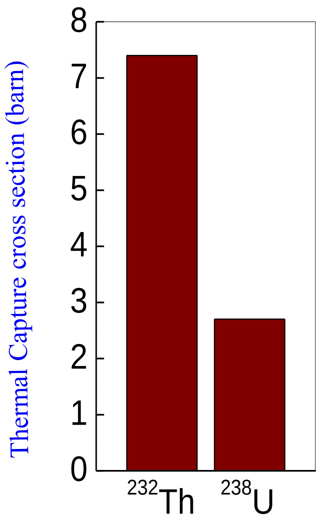


^{233}U has an η value (greater than 2.0) that remains nearly constant over a wide energy range, in thermal as well as epithermal regions, unlike ^{235}U and ^{239}Pu. This facilitates achievement of high conversion ratios with thorium utilisation in reactors operating in the thermal/epithermal spectrum.

η: Number of neutrons released per thermal neutron absorbed



Capture cross section of ^{233}U is much smaller than ^{235}U and ^{239}Pu, the fission cross section being of the same order implying lower non-fissile absorption leading to higher isotopes. This favours the feasibility of multiple recycling of ^{233}U, as compared to plutonium.

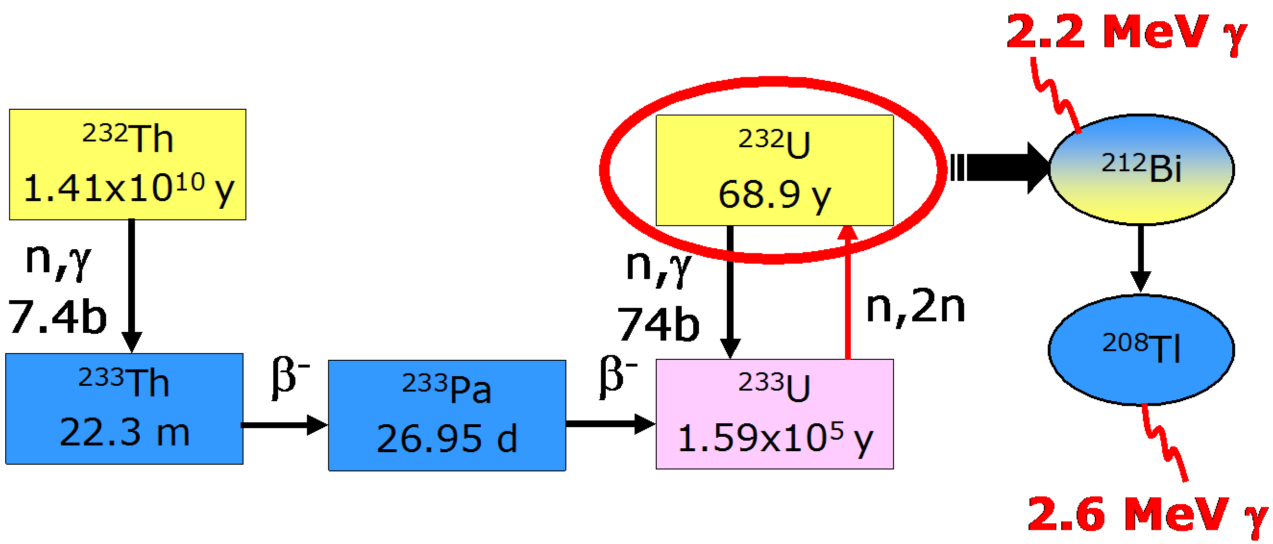


Cross section for capture of thermal neutrons in ^{232}Th is typically 2.47 times that in ^{238}U. Thus thorium offers greater competition to capture of the neutrons and lower losses to structural and other parasitic materials leading to an improvement in conversion of ^{232}Th to ^{233}U.

The neutronic characteristics of ^{233}U-Th lead to:

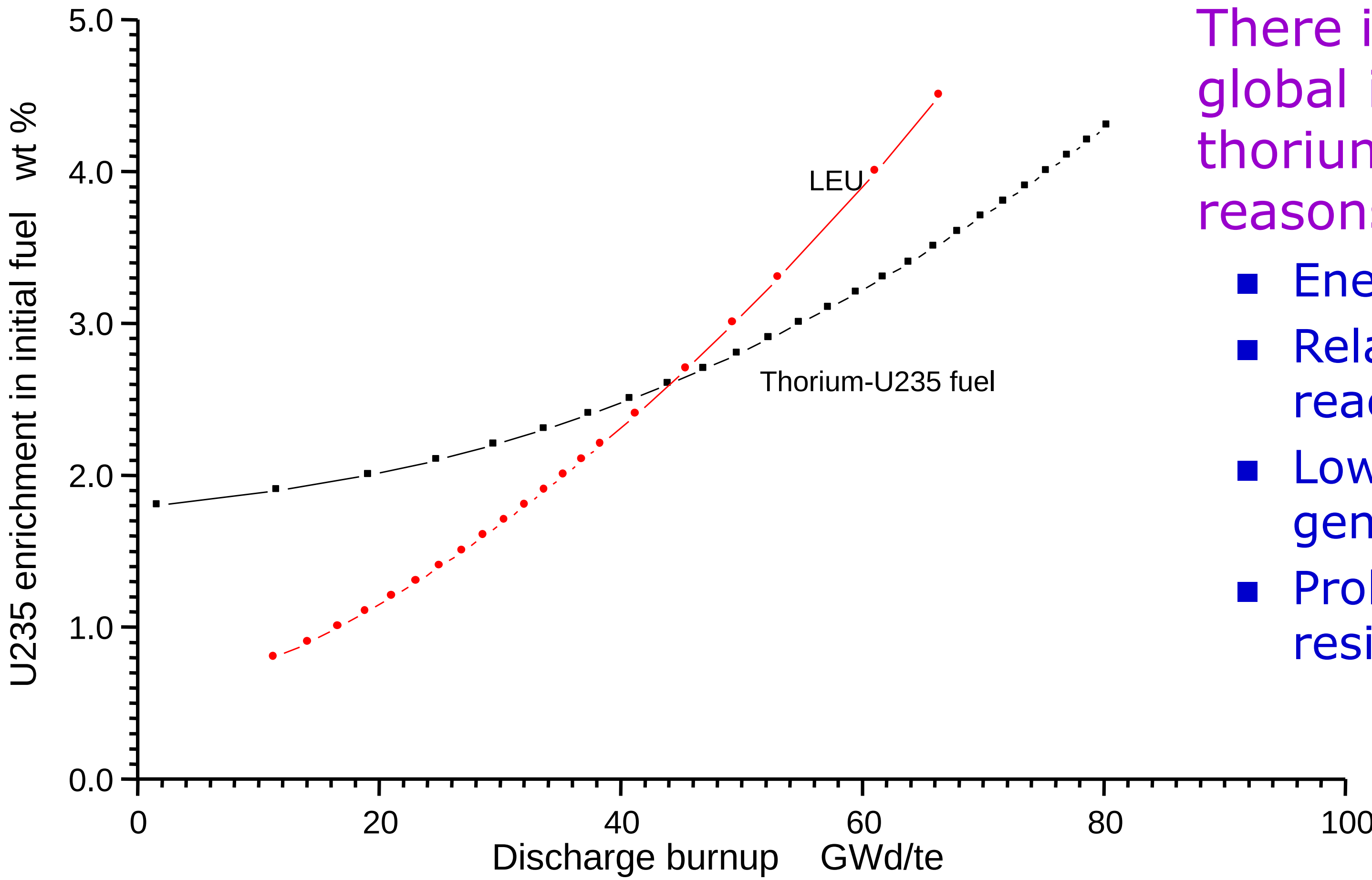
- **High potential for achieving near-breeding condition in thermal reactors**
- **High potential for tolerating higher parasitic absorption (e.g. in light water and structural materials)**

Advantages of Thorium : Presence of ^{232}U offers an Intrinsic Barrier to Proliferation



- ^{232}U is formed via (n, 2n) reactions, from ^{232}Th, ^{233}Pa and ^{233}U. The half-life of ^{232}U is about 69 years. The daughter products (^{208}Tl and ^{212}Bi) of ^{232}U are high energy gamma emitting isotopes.
- Due to presence of ^{232}U in separated ^{233}U, thorium offers good proliferation-resistant characteristics.

Advantage of thorium over uranium becomes evident if thorium based fuels are used at high burnups

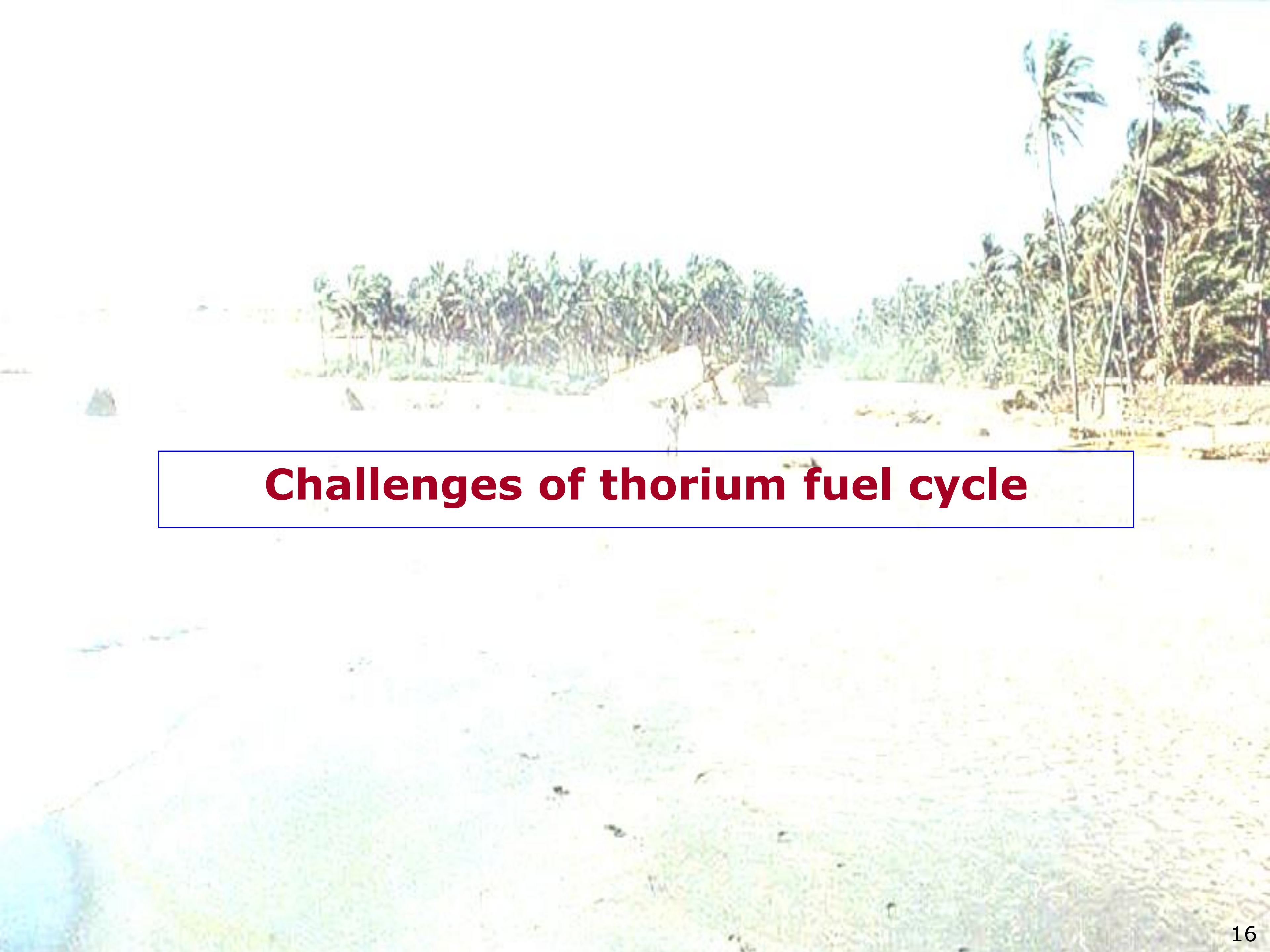


There is a growing global interest in thorium for various reasons such as

- Energy advantage
- Relatively stable core reactivity
- Low minor actinide generation
- Proliferation resistance

Performance potential of homogeneous mixture of fertile materials with ^{235}U in a PHWR

By virtue of being lower in the periodic table than uranium, the long-lived minor actinides resulting from burnup are in much lower quantity with the thorium cycle



Challenges of thorium fuel cycle

Challenges of thorium fuel cycle

- Melting point of ThO_2 (3350 °C) is much higher than that of UO_2 (2800°C).
  - Much higher sintering temperature (>1700°C) is required to produce high density ThO_2 and ThO_2–based mixed oxide fuels.
    - Admixing of 'sintering aid' (CaO, MgO, Nb_2O_5, etc) is required for achieving the desired pellet density at lower temperature.
- ThO_2 and ThO_2 based mixed oxide fuels are relatively inert and do not dissolve easily in concentrated nitric acid.
  - Addition of small quantities of HF in conc. HNO_3 is essential which cause corrosion of stainless steel equipment and piping in reprocessing plants.
  - The corrosion problem is mitigated with addition of aluminium nitrate.
  - Requires long dissolution time in boiling THOREX solution [13M HNO_3+0.05M HF+0.1M $Al(NO_3)_3$] for ThO_2 based fuel
- The irradiated Th or Th–based fuels contain significant amount of ^{232}U, having a half-life of 68.9 years and is associated with strong gamma emitting daughter products, ^{212}Bi and ^{208}Tl with very short half-life.
  - There is significant build-up of radiation dose with storage of spent Th–based fuel or separated ^{233}U,
  - This necessitates remote and automated reprocessing and re-fabrication in heavily shielded hot cells, increasing the cost of fuel cycle activities.

Challenges of thorium fuel cycle

- In the conversion chain of ^{232}Th to ^{233}U, ^{233}Pa is formed as an intermediate, which has a relatively longer half-life (~27 days) as compared to ^{239}Np (2.35 days) in the uranium fuel cycle thereby requiring longer cooling time of at least one year for completing the decay of ^{233}Pa to ^{233}U.
- Normally, Pa is passed into the fission product waste in the THOREX process, which could have long term radiological impact.
  - It is essential to separate Pa from the spent fuel solution prior to solvent extraction process for separation of ^{233}U and thorium.
- The database and experience of thorium fuels and thorium fuel cycles are very limited, as compared to UO_2 and $(U, Pu)O_2$ fuels, and need to be augmented before large investments are made for commercial utilization of thorium fuels and fuel cycles.

Indian experiences with thorium fuel cycle R&D

Evolution of thorium fuel cycle development in India

Thorium extraction

Thoria pellets/pins irradiation in research reactor

Use of thoria bundle in PHWR

Dismantling of irradiated bundle and Post-irradiation examination

Irradiated fuel reprocessing and fabrication

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

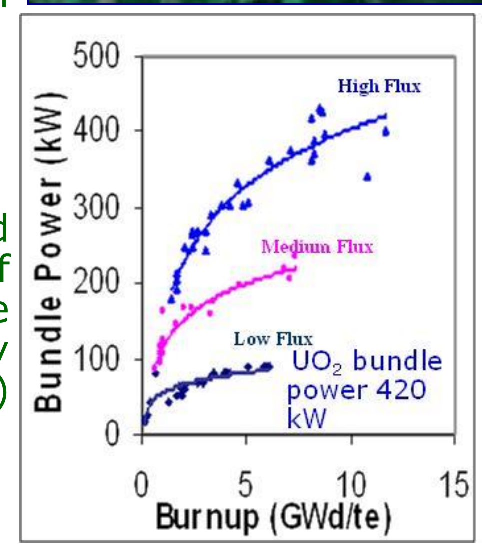
AHWR critical facility

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.
- **Dhruva**: Thoria based MOX fuel pins of AHWR are under irradiation
- **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.



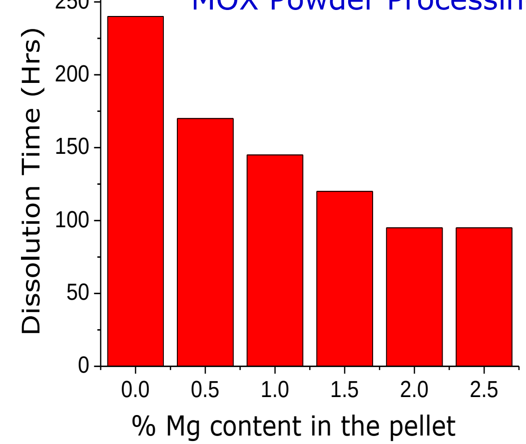
Fuel fabrication and reprocessing facility

- 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

MOX Powder Processing



Thoria dissolution studies

Bundle dismantling

Impregnation setup

Thorium fuel cycle technologies is relatively complex because of
- inert nature of thoria
- radiological aspects

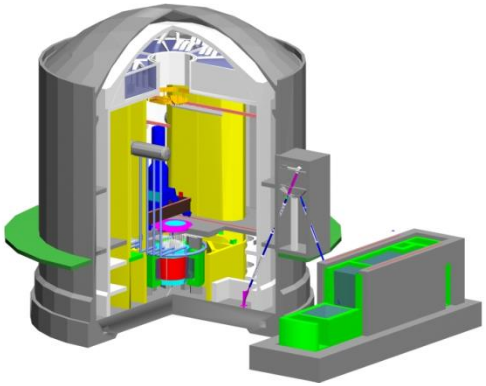
- **Indian Programme on thorium based reactors:**

  - **Advanced Heavy Water Reactor (AHWR)**

  - **Indian HTR programme**
    - **- Compact High Temperature Reactor (CHTR)**
    - **- Innovative High Temperature Reactor (IHTR)**
    - **- Molten salt Breeder reactor (MSBR)**

AHWR General Introduction

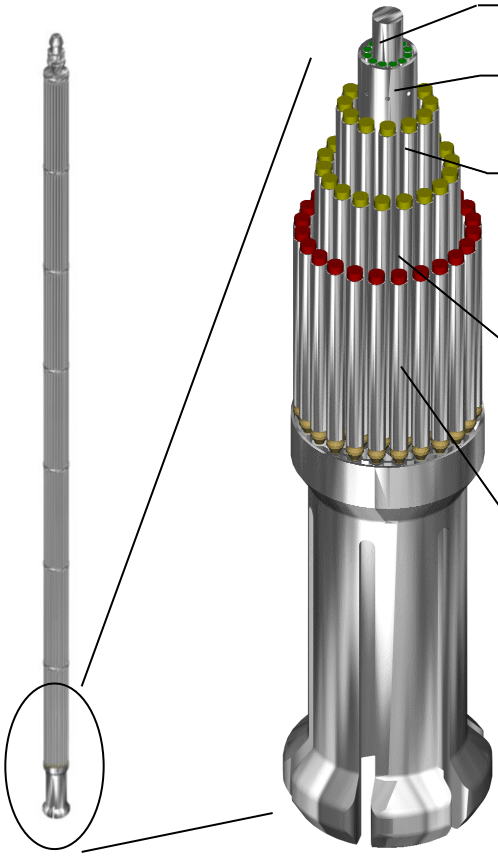
- AHWR is a 300 MWe, vertical, pressure tube type, boiling light water cooled, and heavy water moderated reactor.
- AHWR is a technology demonstration reactor designed to achieve large-scale use of thorium for power generation.
- Provides transition to 3rd stage of the Indian Nuclear Power Programme.



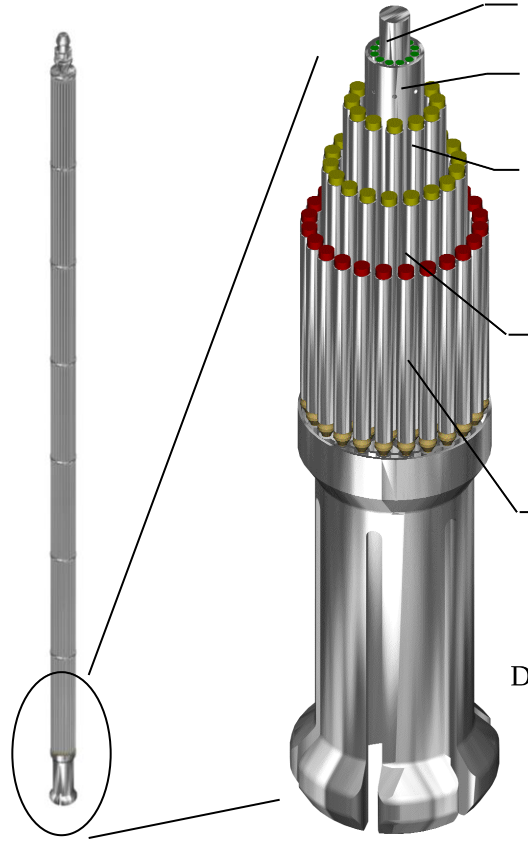
AHWR Schematic

- Addresses most issues relevant to advanced reactor designs like sustainability, enhanced safety, proliferation resistance and economic competitiveness.
- The reactor incorporates a number of passive safety features to reduce environmental impact.
- Fuel cycle flexibility
  - AHWR-LEU using (LEU-Th) MOX
  - AHWR–Pu using (Pu-Th) MOX and (Th-U^{233}) MOX

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

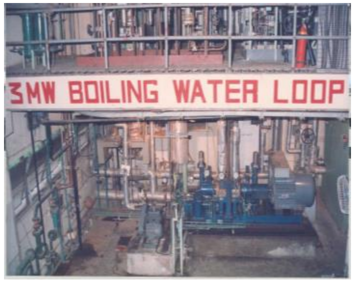


Fuel cluster of AHWR-Pu

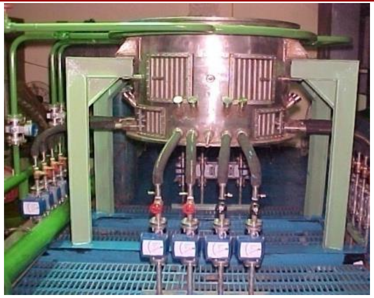


Fuel cluster of AHWR-LEU

Test Facilities for AHWR Design Validation

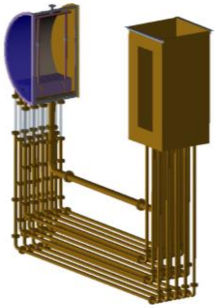



HPNCL


Moderator & liquid poison distribution


PCCTF

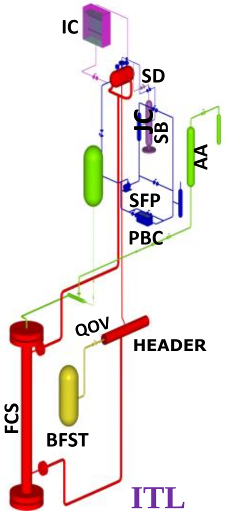

PCITF


SDTF


ATTF, Tarapur

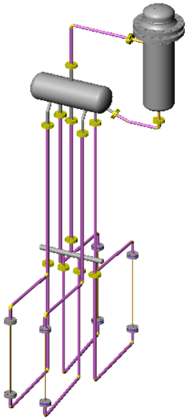

AHWR Critical Facility


Building housing ITL

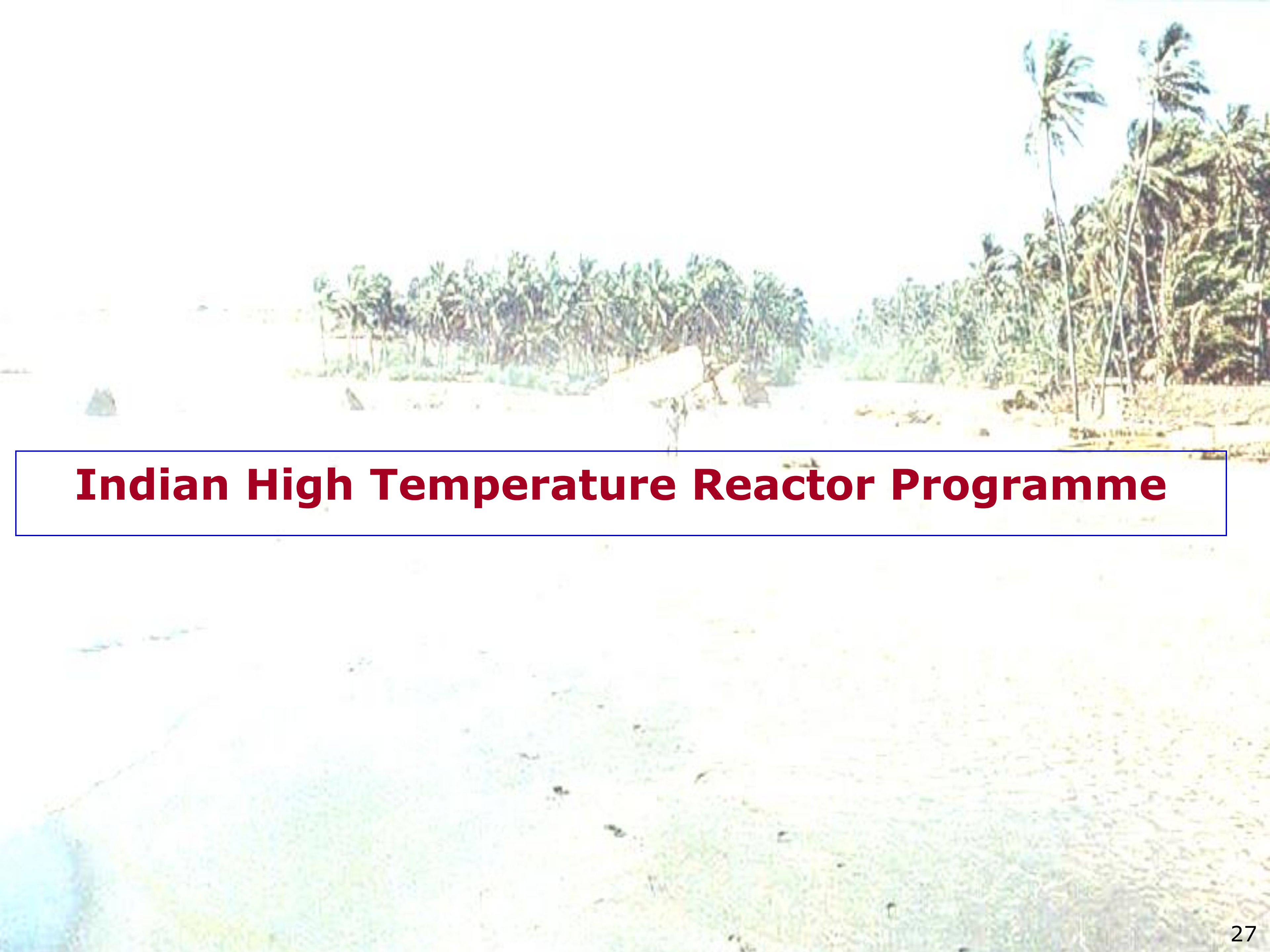


ITL


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.



Indian High Temperature Reactor Programme

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.

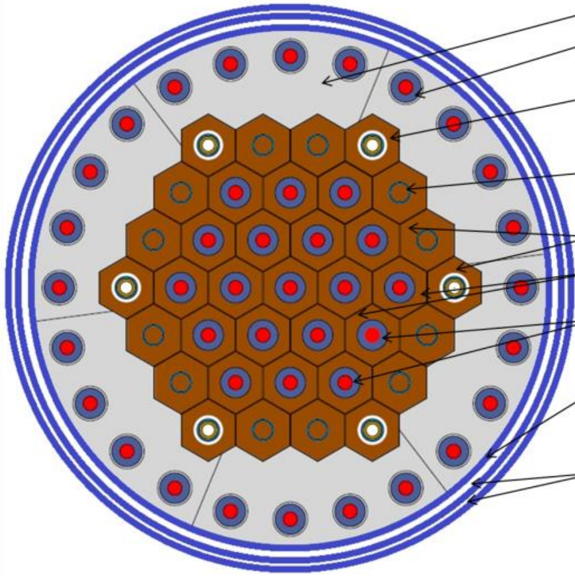
Indian Molten Salt Breeder Reactor (MSBR)

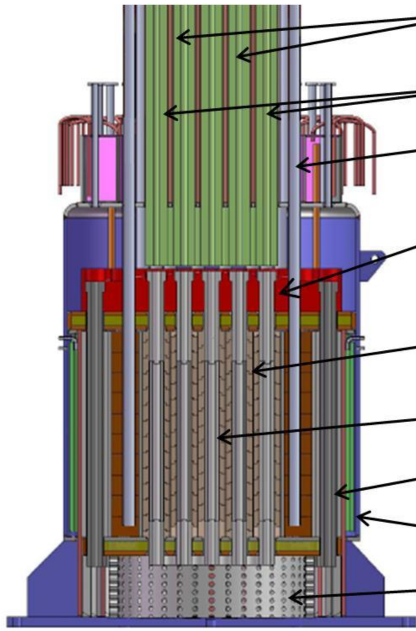
- Medium power, moderate temperature, demonstration reactor based on ^{233}U-Th fuel cycle
  - Small power version for demonstration of technologies
- Emphasis on passive systems for reactor heat removal under all scenarios and reactor conditions

Status: Initial studies being carried out for conceptual design

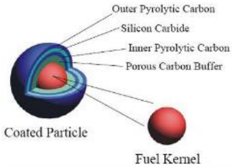
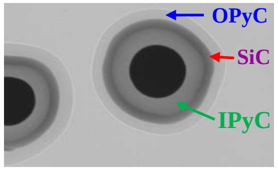
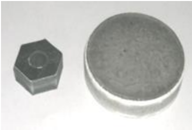
Advanced Reactors – Indian Initiatives
Compact High Temperature Reactor (CHTR)

- 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

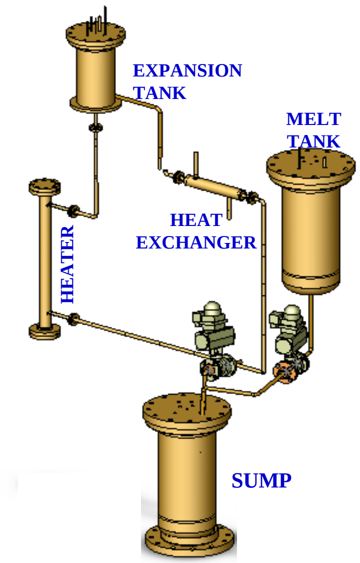




Major Research & Development Areas in Progress

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 ▪ Oxidation & corrosion resistant coating	▪ Fuel tubes made using various techniques & materials including carbon-carbon composites ▪ Beryllia blocks manufacturing technology established ▪ Techniques for SiC coating on graphite & silicide coating on Nb developed  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 LBE coolant • Code development & validation • High temperature instrumentation	▪ LBE test loop operated to validate design codes. ▪ Loop for studies at 1000 °C established. ▪ Level probes, oxygen sensor, etc. developed for LBE coolant

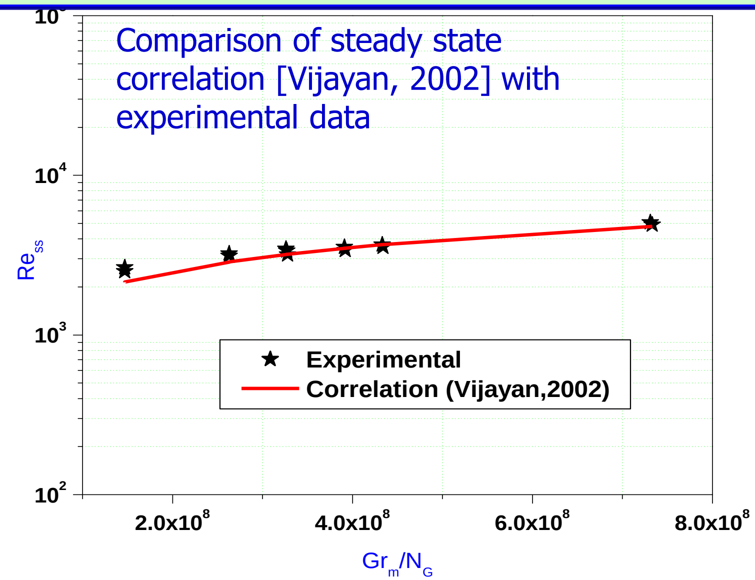
Thermalhydraulic Studies for LBE Coolant

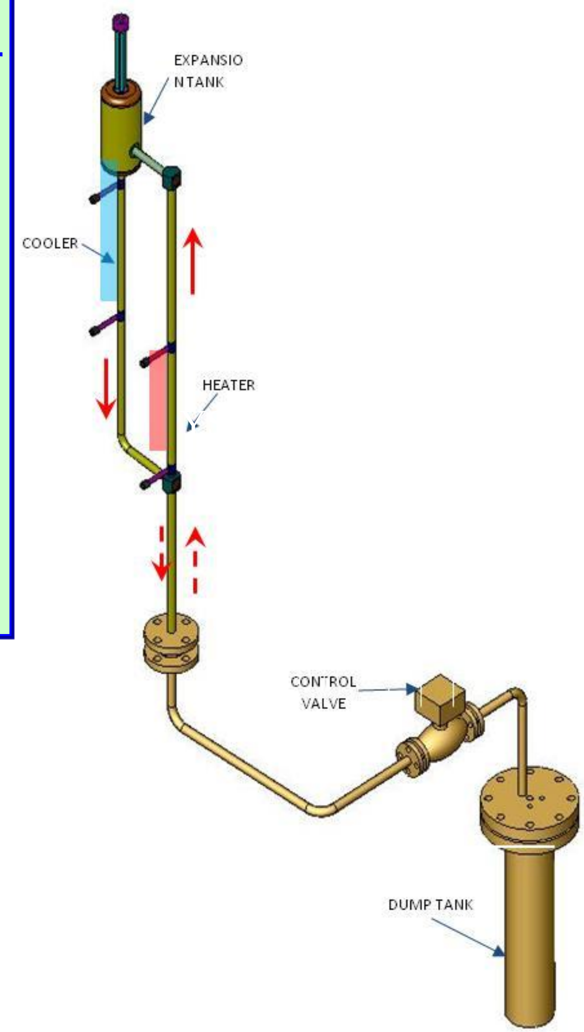


Liquid Metal Loop (2009)

Major areas of development

- Analytical studies and development of computer codes
- Liquid metal loop for experimental studies
  - Loop at 550 °C operating since 2009
  - Loop at 1000 °C established
- Steady state and transient experiments carried out
- In-house developed code validated
- Experimental and analytical studies for freezing and de-freezing of coolant
- Test bed for instrumentation development-level probes, oxygen sensor, EM pump & flowmeters





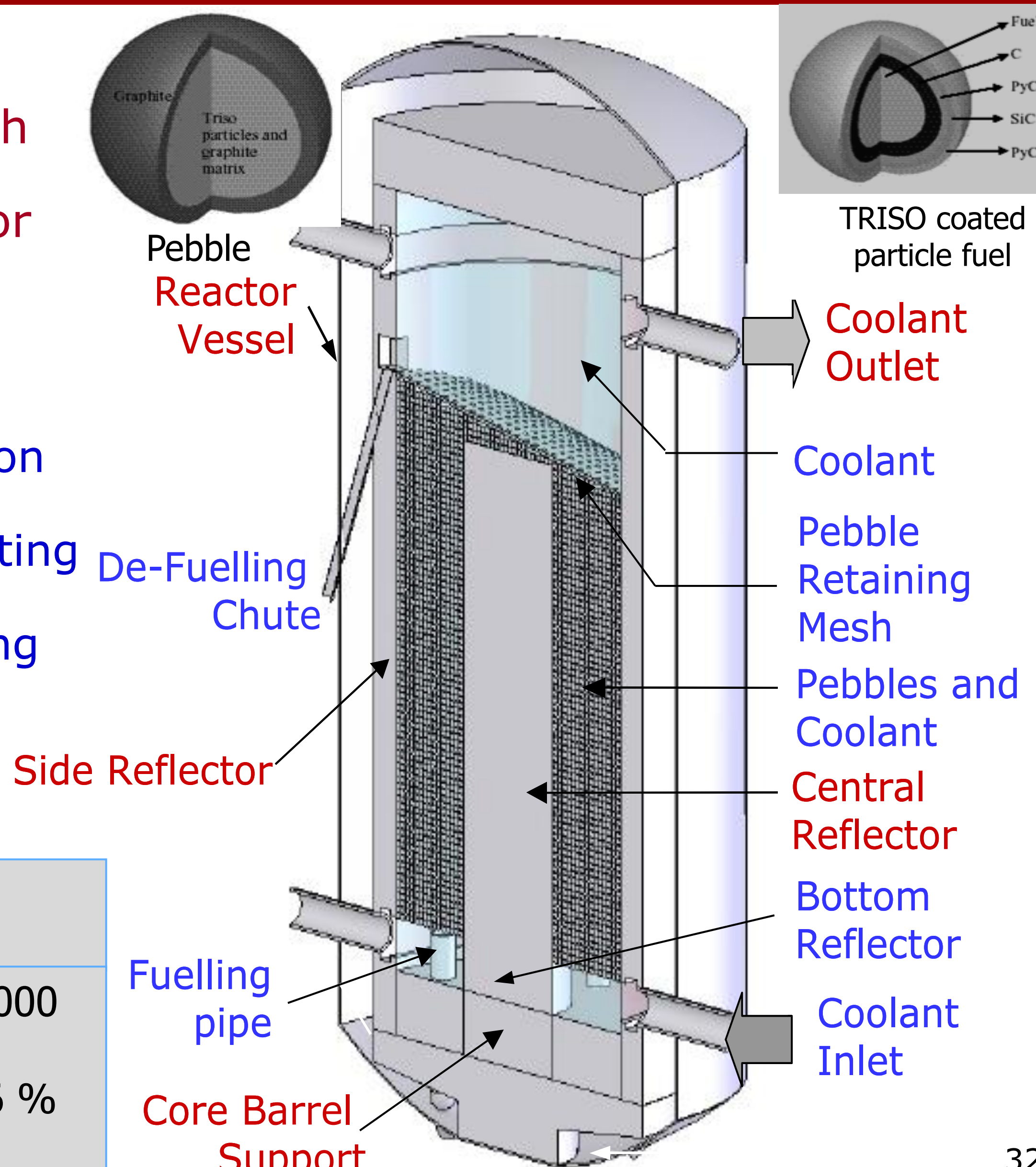
Isometric of Kilo temperature Loop

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 design – 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

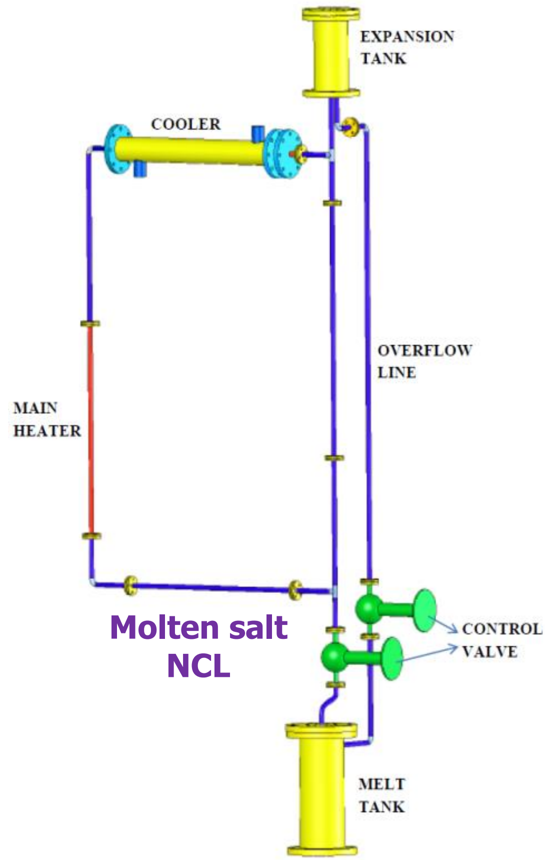
- No. of pebbles in the annular core ~150000
- Packing fraction of pebbles ~60%
- Packing fraction of TRISO particles ~ 8.6 %
- ^{233}U Requirement 7.3 %



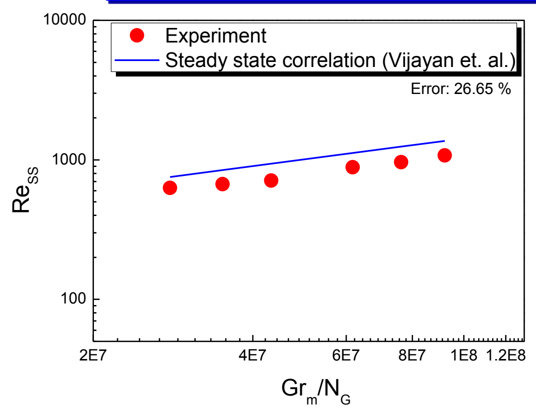
Thermal hydraulic studies and material compatibility studies for molten salt coolant

Major areas of development

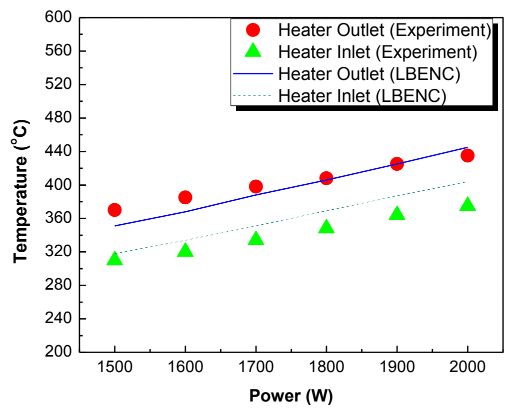
- Analytical studies and development of computer codes
- Molten salt natural circulation loop for experimental studies
- Molten fluoride salt corrosion facility using FLiNaK
    - Experiments carried out up to 750 °C mainly on Inconel materials



Molten salt NCL



Steady state NC flow



Predicted and measured temperatures

Molten salt corrosion test facility

Utilisation of ^{233}U in a Th matrix in metallic fuelled FBRs and MSBRs - a comparison

	Metallic ^{233}U/Th fuelled FBR with Th blanket [1]	Molten Salt Breeder Reactor[2]
Reactor Power	1000 MWe	1000 MWe
Cycle fissile inventory	6 tonnes*	<1 tonnes **
Breeding ratio	1.115	1.14

* Assuming out of core time (cooling + reprocessing + refabrication) to be 3 years
** Assumes online reprocessing

- Cycle fissile inventory for metallic fuelled FBRs are very high (~6 t) compared to MSBRs operating in thermal or epithermal spectrum
- Breeding ratio of 1.06 to 1.14 is possible in MSBRs operating in thermal and epithermal zone. FBRs will have BRs ~1.115
- Reprocessing in FBRs requires extensive clad removal and fuel refabrication – avoided in MSBR

MSBR provides significant advantages over metallic fuelled FBRs and hence is an attractive option for the 3rd stage and conceptual design of IMSBR in progress.

Ref.: [1] Alternate Fuel Cycles for Fast Breeder Reactors, INFCE/DEP/WG.5/78, International Nuclear Fuel Cycle Evaluation, 1978

Ref. : [2] ORNL-4528

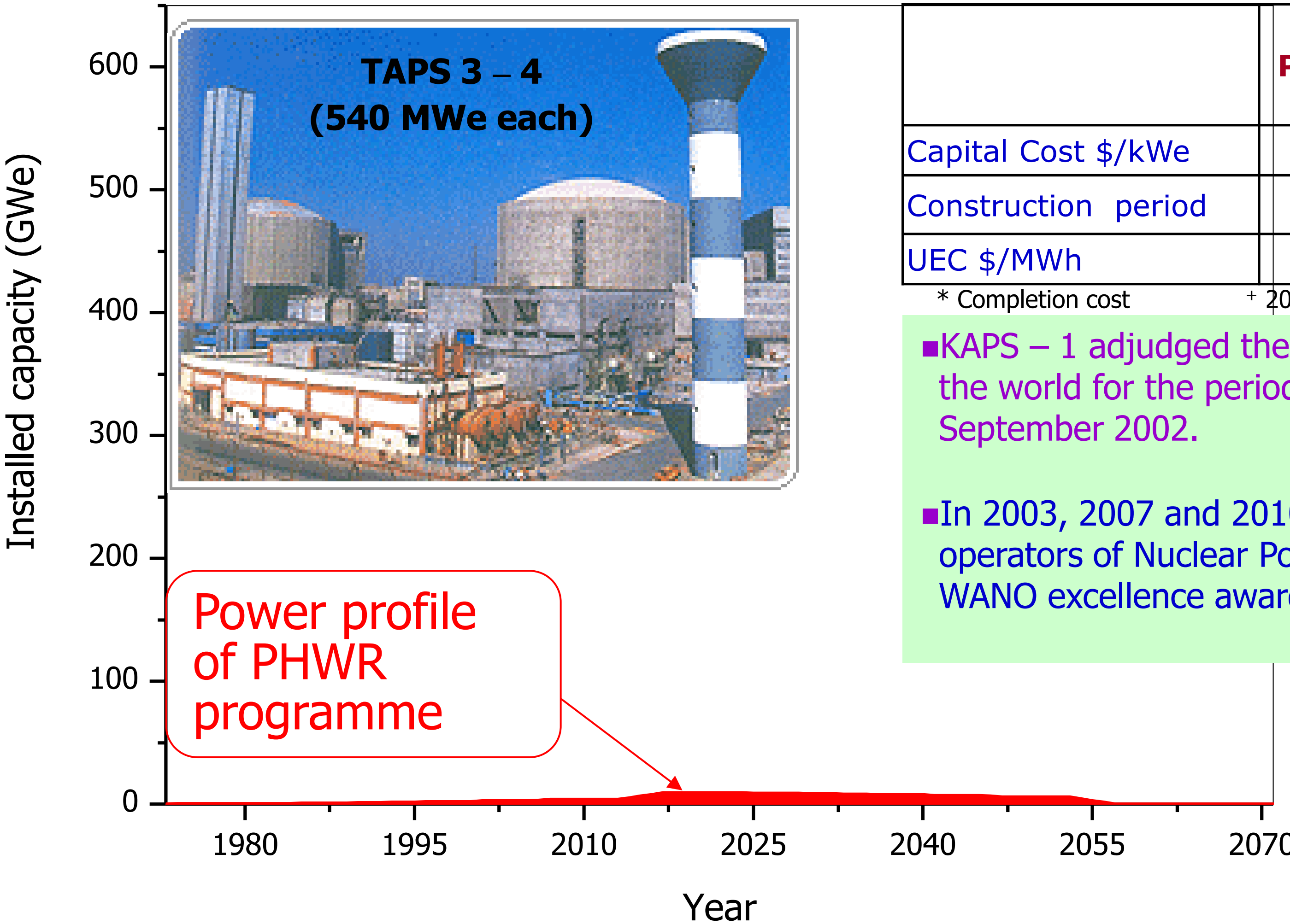
Concluding Remarks

- Thorium offers a sustainable and proliferation resistant fuel option with lower minor actinides burden
- Over a period of time, indigenous technologies developed 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 thorium based high temperature reactor concepts dedicated for hydrogen production
- Large scale deployment of molten salt breeder reactors is envisaged during the third stage of Indian nuclear power programme

Thank you

Optimising the use of Domestic Nuclear Resources

NPCIL is a AAA (CRISIL) rated company for ten years in a row.



	Indian PWHRs (700 MWe)	Global range
Capital Cost $/kWe	1700	2000-2500
Construction period	5-6 years	5-6 years
UEC $/MWh	60	60 - 70

* Completion cost + 2008 Prices

- KAPS – 1 adjudged the best performing PHWR in the world for the period October 2001 to September 2002.
- In 2003, 2007 and 2010, three senior Indian operators of Nuclear Power Stations, received the WANO excellence award.

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

Challenges of thorium fuel cycle

- Melting point of ThO_2 (3350 °C) is much higher than that of UO_2 (2800°C).
  - Much higher sintering temperature (>1700°C) is required to produce high density ThO_2 and ThO_2–based mixed oxide fuels.
    - Admixing of 'sintering aid' (CaO, MgO, Nb_2O_5, etc) is required for achieving the desired pellet density at lower temperature.
- ThO_2 and ThO_2 based mixed oxide fuels are relatively inert and do not dissolve easily in concentrated nitric acid.
  - Addition of small quantities of HF in conc. HNO_3 is essential which cause corrosion of stainless steel equipment and piping in reprocessing plants.
  - The corrosion problem is mitigated with addition of aluminium nitrate.
  - Requires long dissolution time in boiling THOREX solution [13M HNO_3+0.05M HF+0.1M $Al(NO_3)_3$] for ThO_2 based fuel
- The irradiated Th or Th–based fuels contain significant amount of ^{232}U, having a half-life of 68.9 years and is associated with strong gamma emitting daughter products, ^{212}Bi and ^{208}Tl with very short half-life.
  - There is significant build-up of radiation dose with storage of spent Th–based fuel or separated ^{233}U,
  - This necessitates remote and automated reprocessing and re-fabrication in heavily shielded hot cells, increasing the cost of fuel cycle activities.

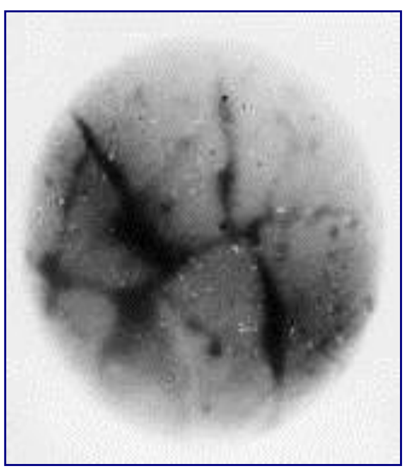
Post Irradiation Examination (PIE) of thoria fuel

- 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 (^{137}Cs) migrate to thoria pellet cracks unlike upto periphery in UO_2 fuel.

Isotopic Composition of Discharged Uranium (%)						
	^{232}U	^{233}U	^{234}U	^{235}U	^{236}U	^{238}U
Mass Spectrometric Analysis	0.0459	88.78	9.95	1.0	0.085	0.14
Theoretical Prediction *	0.0491	90.556	10.945	1.07	0.0918	-

Power peaking in the central elements
Atom % fission = 1.25%
Fission products measured were ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce-^{144}Pr, ^{154}Eu, ^{155}Eu, ^{90}Sr.
Gross activity of the bundle measured

α-Autoradiograph



β, γ - Autoradiograph

PIE hot cell facility

Fission gas analysis set up

Brief history of thorium fuel based reactors

Experience with thorium based fuels in world -Thorium fuel experience of more than three decades old exists for test reactors & power reactors of different types

Name & country	Type	Power	Fuel	Operation
Lingen, Germany	BWR	60 MWe	Test fuel $(Th+Pu)O_2$ pellets	Till 1973
MSRE, ORNL, USA	MSBR	8 MWt	^{233}U molten fluorides	1964-1969
Shippingport & Indian Point 1, USA	LWBR, PWR	60 MWe 285 MWe	$Th+^{233}U$ driver fuel, Oxide Pellets	1977-1982 1962-1980
SUSPOP/KSTR, KEMA, Netherlands	Aqueous homogeneous suspension	1 MWt	Th+HEU, Oxide pellets	1974-1977
NRU & NRX, Canada	MTR		$Th+^{235}U$, Test fuel	Irradiation of few elements

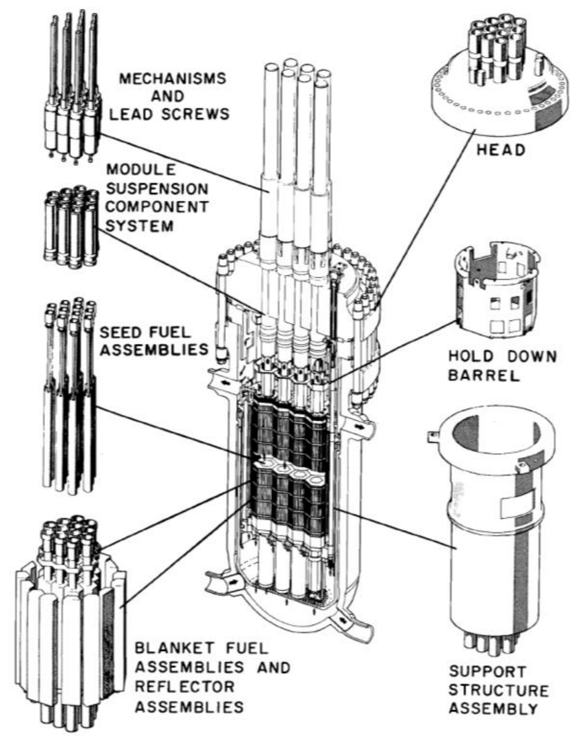
1966: MSRE
Molten-Salt Reactor Experiment (MSRE)
Operated for 17,655 h

Shipping port Reactor: A major experience in the use of thorium

First large-scale nuclear power reactor for electricity-60 MWe
- Test bench for thermal breeder using ^{233}U fuel

Operated as LWBR during 1977-1982
- 1.39% more fissile fuel at EOL

Breeding success achieved
- by high cost of sophisticated core
- by sacrificing reactor performance

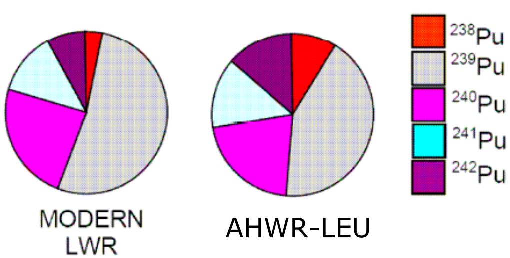


Thorium based fuels have been loaded either partially or fully in High Temperature Gas cooled Reactor (HTGR) cores.

Name & country	Type	Power	Fuel	Operation
AVR, Germany	HTGR (Pebble bed)	15 MWe	Th+^{235}U driver fuel, Coated fuel particles of oxide & dicarbides	1967-1988
THTR-300, Germany	HTGR (Pebble bed)	300 MWe	Th+^{235}U driver fuel, Coated fuel particles of oxide & dicarbides	1985-1989
Dragon, UK, OECD	HTGR (Prismatic block)	20 MWt	Th+^{235}U driver fuel, Coated fuel particles of oxide & dicarbides	1964-1976
Peach Bottom, USA	HTGR (Prismatic block)	40 MWe	Th+^{235}U driver fuel, Coated fuel particles of oxide & dicarbides	1967-1974
Fort St. Vrain, USA	HTGR (Prismatic block)	330 MWe	Th+^{235}U driver fuel, Coated fuel particles, Dicarbides	1976-1989

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

- AHWR is a technology demonstration reactor is designed to achieve large-scale use of thorium for power generation.
- Addresses most issues required in advanced reactor designs
  - Enhanced safety, Proliferation concern, Minimize waste burden
  - Maximize resource utilisation (sustainability) and
  - Economic competitiveness

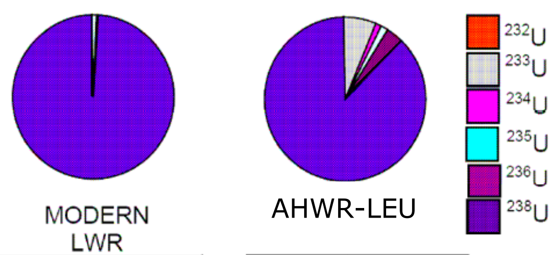


MODERN LWR

^{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	%