Thorium Utilisation in Advanced Reactor Designs

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Outline

- Thorium as nuclear fuel : advantages and challenges
- Global Experience with thorium
- Thorium : Indian perspective
- Indian Experience with thorium
- Advanced designs using thorium fuel cycle
- Advanced reactor systems in India utilising thorium
- Concluding remarks

Thorium as nuclear fuel : advantages and challenges

Thorium and Uranium : Naturally occurring materials to harness nuclear energy

- Natural Thorium
  - Abundant
  - Does not have fissile component
- Natural Uranium
  - With 0.7% content of fissile U-235
  - Well developed fuel cycle
- Presently, global nuclear energy programme is predominantly based on fissile U-235

- Sustainability of nuclear energy requires
  - Adoption of closed fuel cycle with breeding
  - Use of the fertile thorium resource
  - Use of thorium has to begin well before using up uranium resource

Attractive features of thorium

- Abundance of thorium
  - Uniformly distributed in earth crust
  - 3 to 4 times abundant than uranium
- Better Performance Characteristics
  - Higher melting point
  - Better thermal conductivity
  - Lower fission gas release
  - Good radiation resistance and dimensional stability
  - Reduced fuel deterioration in the event of failure
- Waste Management
  - No oxidation during permanent disposal in repository
  - Generates less plutonium and higher actinides

Challenges in the use of thorium

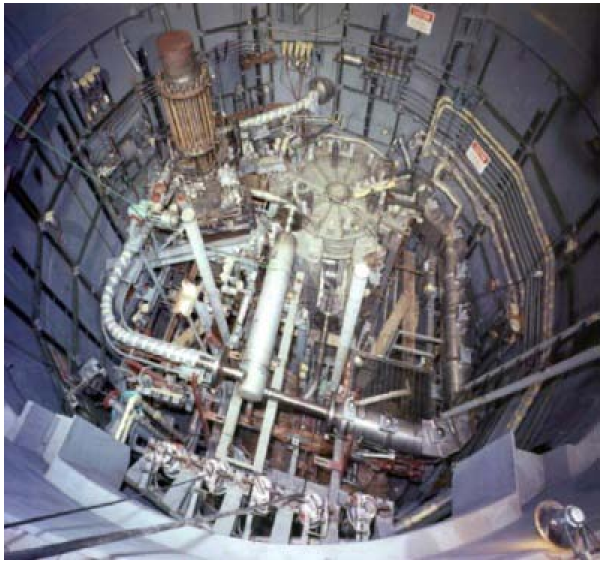
- Higher sintering temperature required for fabrication
- Presence of ^{232}U in ^{233}U
  - Recovered ^{233}U will always be contaminated with ^{232}U
  - The daughter products of ^{232}U, ^{212}Bi and ^{208}Tl are emitters of hard gamma rays.
  - Recycling of uranium requires fuel fabrication to be carried out in shielded hot-cells remotely and with considerable automation.
- Problems in reprocessing
  - Stable nature poses a major challenge in dissolution
  - Single Oxidation State of Th. Difference in the selectivity has to be optimized to meet the desired Partitioning.
  - Third phase formation
- Waste management aspects
  - HLLW from thoria based spent fuel reprocessing will contain Th, Al and corrosive F.

Global Experience with Thorium

BARC

Early years in the use of thorium

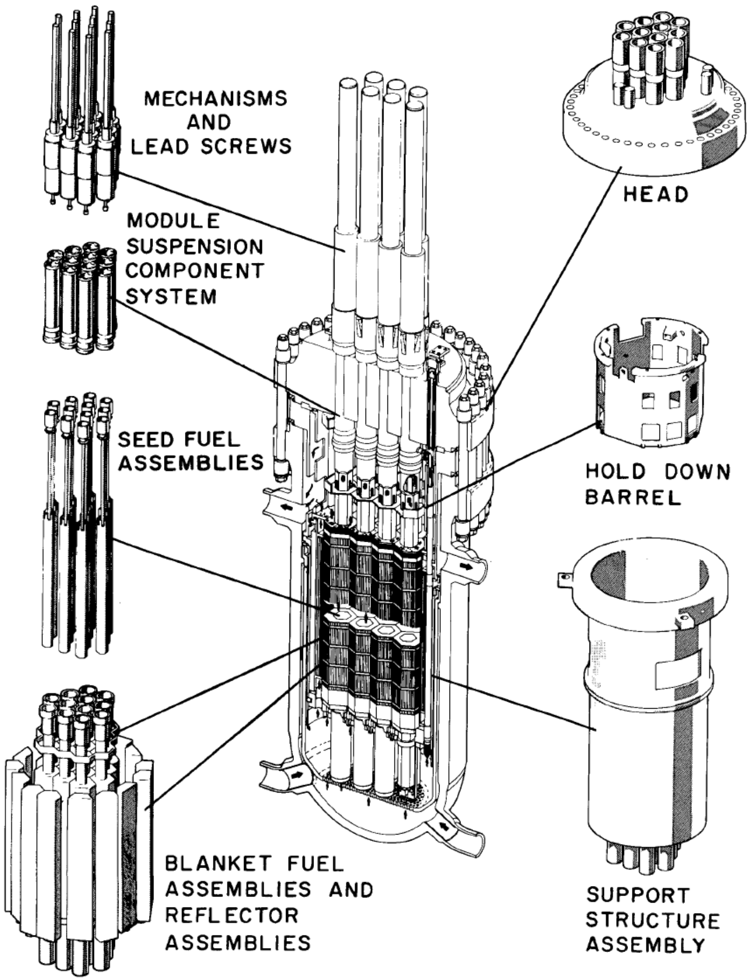
- 1956 : BORAX-IV
  - Boiling Water Reactor
  - Explored the thorium fuel cycle and uranium-233 fuel with a power of 20 MW thermal
- 1962 : Indian Point-1
  - PWR designed to produce 275 MWe
  - First core used thorium-based fuel
- 1966 : MSRE
  - Molten-Salt Reactor Experiment (MSRE)
  - Operated for 17,655 h (~ 2 years)
- 1967 : AVR, Germany
  - AVR (a pebble-bed high temperature research reactor)
  - Operated for more than 2 decades using HEU-Th-fuel
  - fuel burnups of more than 140 000 MW·d/tHM achieved



MSRE

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

- First large-scale nuclear power reactor for electricity
  - Net station output - 60 MWe
  - Test bench for thermal breeder using ^{233}U fuel
- Operated as LWBR
  - 1977-1982
  - 1.39% more fissile fuel at EOL
- Breeding success achieved
  - by high cost of sophisticated core
  - by sacrificing reactor performance



Experience with Thorium based fuels in world

- Thorium fuel experience exists for test reactors and power reactors of different types. All these experiences are more than three decades old.

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 USA	LWBR	60 MWe	$Th+^{233}U$ driver fuel,	1962-1980
Indian Point 1, USA	PWR	285 MWe	$Th+^{233}U$ driver fuel, Oxide pellets	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

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

Thorium Utilisation : Indian Perspective

Indian Nuclear Fuel Resources

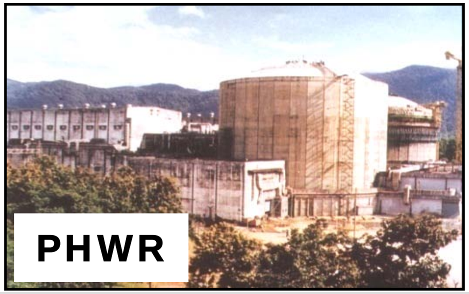
- ^{232}Th found abundantly in beach sands of Kerala and Orissa
- > 225000 tonnes of thorium

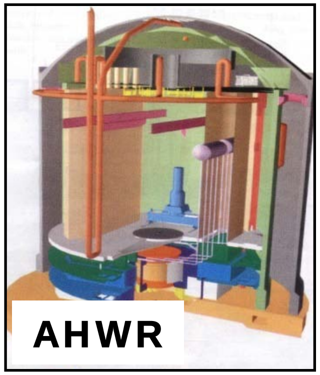
- Uranium to be mined (~ 61000 t of uranium)
- Prominent mines are located at Jaduguda, Bhatin, Narwapahar, Turamdih and Bagjata

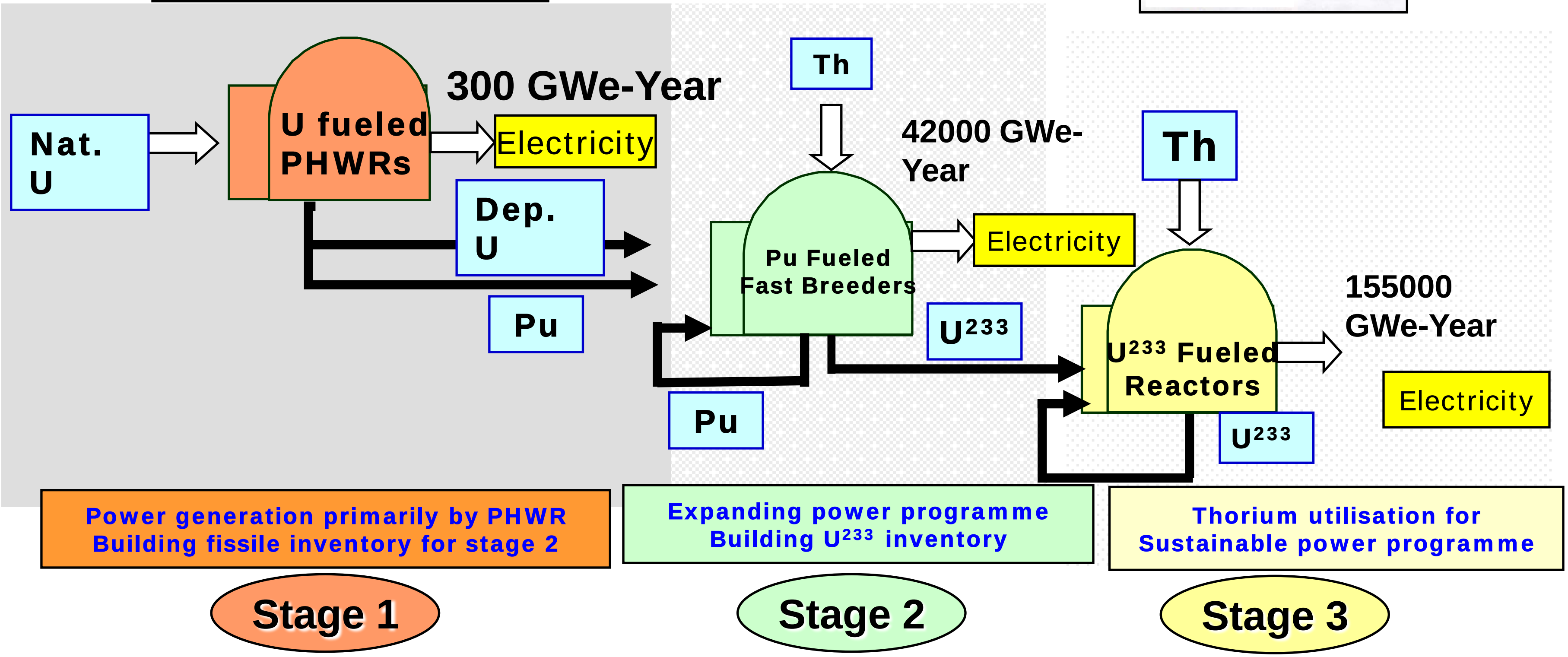
Resource	Quantity (tonne)	Energy Potential (GWe-yr)
Uranium	61,000	328 in PHWR 42,230 in Fast Breeders
Thorium	> 225,000	>155,500 in Breeders

Three Stage Indian Nuclear Programme

An important role for Thorium





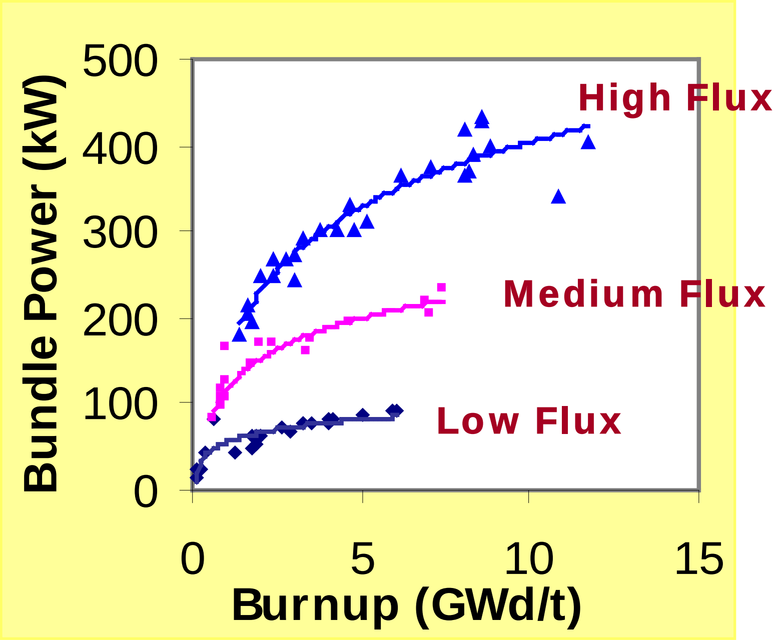


Experience with Thorium in India

Summary of thorium fuel cycle experience in India

Fabrication	J-rods of CIRUS ThO_2 fuel for Dhruva Thoria fuel bundles for PHWR Thoria fuel assemblies for FBTR blanket
Irradiation	CIRUS J-rod position Dhruva regular fuel location PHWR initial flux flattening FBTR blanket Experimental thoria based MOX fuel pins of BWR & PHWR type
Reprocessing	J-rods of CIRUS at BARC & IGCAR New facility PRTRF for PHWR Thoria bundles is being constructed at BARC
Utilisation of U-233	PURNIMA-II liquid fuel (Uranyl nitrate solution) KAMINI plate type fuel

Use of Thoria in PHWRs



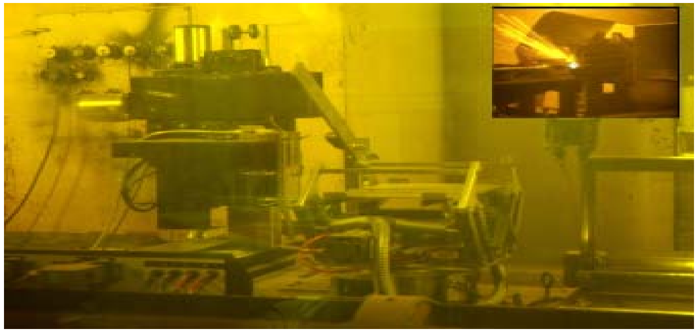
Reactor	No. of bundles
Madras- I	4
Kakrapar-I	35
Kakrapar-II	35
Rajasthan- II	18
Rajasthan -III	35
Kaiga-II	35
Rajasthan-IV	35
Kaiga-I	35

Test Irradiations with (Th-Pu) MOX in PWL, CIRUS

- (Th-4%Pu) MOX fuel pins of TAPS-BWR design
- (Th-6.75%Pu) MOX fuel pins of PHWR design
- (Th-8%Pu)MOX fuel pins of AHWR design


PIE of Thoria Assemblies

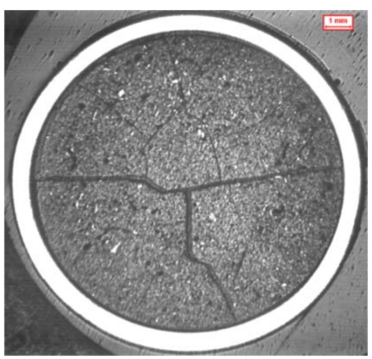
- Power Peaking in the central elements
- Atom % fission = 1.25%
- Fission products measured were
- ^{125}Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce-^{144}Pr, ^{154}Eu, ^{155}Eu, ^{90}Sr.
- Gross activity of the bundle measured



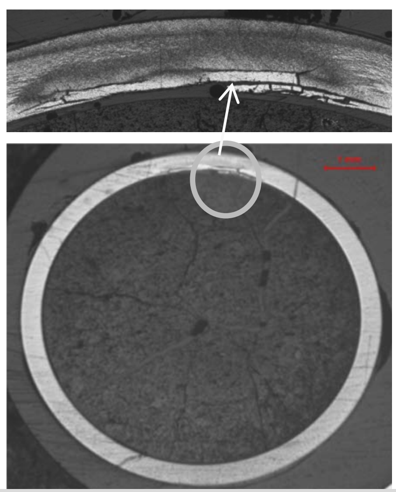
Isotopic Composition of Discharged Uranium (%)						
	^{232}U	^{233}U	^{234}U	^{235}U	^{236}U	^{238}U
Mass Spectrometric Analysis	0.0459	88.78	9.95	1.0	0.085	0.14
Theoretical Prediction *	0.0491	90.556	10.945	1.07	0.0918	-

ThO_2 is more tolerant to clad failure!

- Comparison of Thoria & Urania based fuel behaviour
  - Inherently stable
  - Single valency
  - Lower diffusivity
  - Better thermal conductivity
  - Higher FG retention



ThO_2-4%PuO_2



Failed ThO_2-4%PuO_2

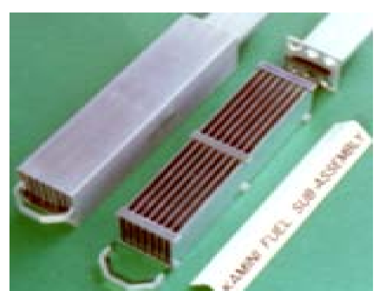
Failed UO_2

UO_2-4%PuO_2

Experience with ^{233}U in India

- PURNIMA II (1984-86)
  - Experiments with uranyl nitrate solution containing ^{233}U reflected by BeO blocks.
- PURNIMA III (1990-93)
  - Experiments were performed with ^{233}U-Al Dispersion Fuel in the form of plates
  - These measurements helped in finalising the core of KAMINI reactor.

- KAMINI (1996)
  - A 30 KW reactor based on ^{233}U fuel in the form of U-Al alloy
  - It is the only operating reactor in the world with ^{233}U as fuel.



Advanced Reactor Designs using Thorium Fuel Cycle

Thorium fuel cycles becoming attractive worldwide

- The Generation IV International Forum (GIF)
  - Two out of six nuclear energy systems selected can potentially use thorium fuel cycle as a breeder; a burner of actinides from spent fuel using thorium matrices
    - Lead-cooled fast reactor (LFR)
    - Molten Salt Reactor (MSR)

- International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)
  - Identified a number of thorium-based fuel cycle options
  - Coordinated Research Project (CRP) on Assessment of Thorium Fuel Cycle for Thermal and Fast Reactors

Studies to Utilise Thorium

- Russia
  - Kurchatov Institute (VVERT reactor, MSR, HTGR), IPPE (WWER type reactors, FR, MSR), VINIEF, ITEF (HWR, ADS).
- USA
  - PWR - Heterogeneous, seed/blanket (SBU), fuel assembly being designated as Radkowsky Thorium Fuel (RTF) concept
  - LWR - develop a ThO_2-UO_2 fuel compatible with existing LWRs
  - High Conversion, Boiling Water Reactor (HCBWR) concept
  - Gas Turbine – Modular Helium Reactor (GT-MHR), based on thorium
  - Liquid-Fluoride Thorium Reactor (LFTR) concept uses uranium and thorium dissolved in fluoride salts of lithium and beryllium
- Other
  - CANDU/PHWR reactors have been studied to adopt thorium cycle and can act as Pu incinerator
  - China - Thorium molten-salt reactor technology

Thorium in ADS

- ADS based systems
  - Rubbia's concept (1994) of running an accelerator-driven thorium-based reactor (for generating less toxic waste in the future, compared to uranium) immersed in a liquid lead bath (for passive safety) may provide an elegant method of long-lived waste transmutation

- Thorium Molten-Salt Nuclear Energy Synergetic System [THORIMS-NES] based on the thorium–uranium-233 cycle.
  - The energy is produced in molten-salt reactors (FUJI) and fissile ^{233}U is produced by spallation in Accelerator Molten-Salt Breeders (AMSB), which would breed U-233 at a high rate in accelerator-enhanced molten-salt breeders.

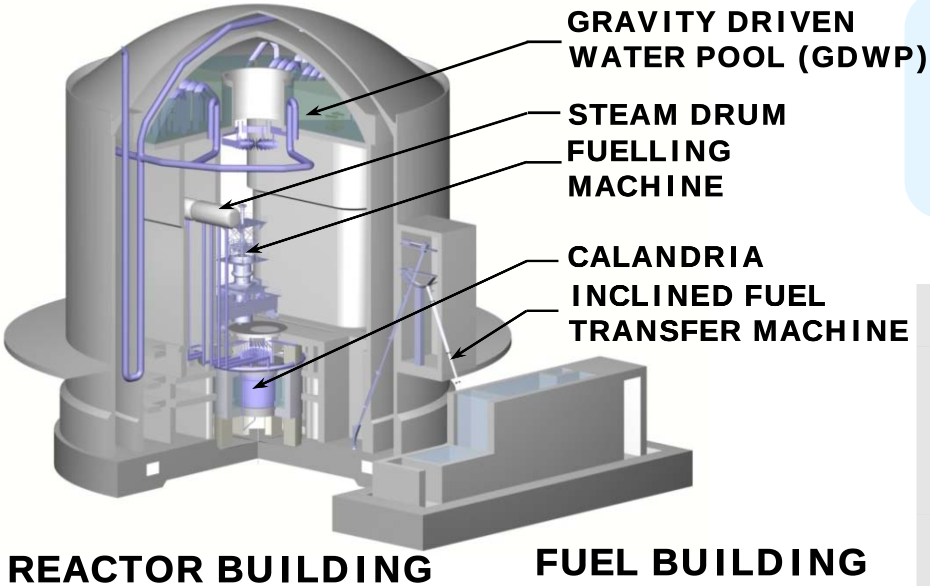
Advanced Reactor Systems in India to Utilise Thorium

AHWR

- AHWR is being set up as a technology demonstration reactor keeping in mind the long term deployment of Thorium based reactors in the third phase.
  - Provides transition to Phase III of Indian Nuclear Power Programme.

- For sustainable development of nuclear energy a number of issues are to be addressed in the reactor design.
  - Enhanced safety
  - Proliferation concern
  - Minimise waste burden
  - Maximise resource utilisation

Advanced Heavy Water Reactor (AHWR)



AHWR is a vertical pressure tube type, boiling light water cooled and heavy water moderated reactor using ^{233}U-Th MOX and Pu-Th MOX fuel.

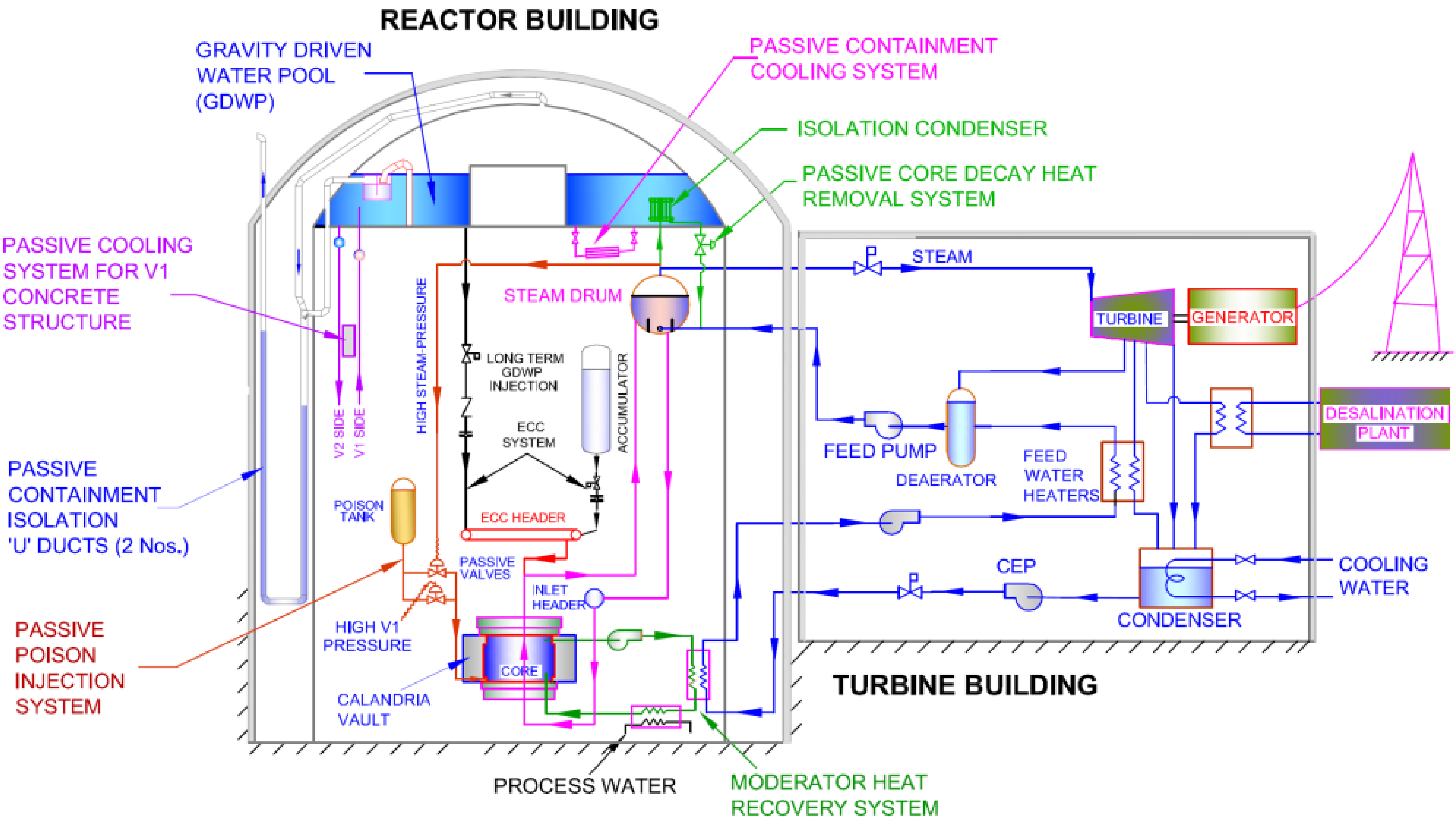
Design Objectives

1. Thorium utilisation & Energy Security
2. Incorporation of Passive Safety Systems
3. Plant location in a populated area
4. Electric Power output – 300 MWe
5. Design life of 100 years

Major Design Parameters

Reactor power	:	300 MWe with 500 m^3/day desalinated water
Moderator	:	Heavy water
Coolant	:	Boiling light water under natural circulation
Coolant Channels	:	452 No.
Lattice pitch	:	225 mm square pitch
Fuel cluster- 54 pins	:	$(Th\text{-}Pu)O_2$: 24 pins $(Th\text{-}^{233}U)O_2$: 30 pins
Fuel burn up	:	38,000 MWd/Te (Avg)
Primary Shut Down System	:	37 Shut off rods
Secondary Shut Down System	:	Liquid poison injection in moderator

AHWR - Schematic



AHWR – Passive Safety Features

- Core heat removal by natural circulation during normal operation and shut down conditions.
- Slightly negative void coefficient of reactivity .
- Emergency Core Cooling during accidental condition (LOCA) by direct injection of coolant in fuel from accumulators and Gravity Driven Water Pool (GDWP). Core submergence following LOCA.
- Grace period of 10 days.
- Double containment.
- Containment heat removal during LOCA by vapour suppression in GDWP and Passive Containment Coolers suspended below GDWP.
- Containment isolation during LOCA by formation of water seal in ventilation ducts.
- Passive Poison Injection in moderator during overpressure transient.

Selection of Fuel Material for AHWR

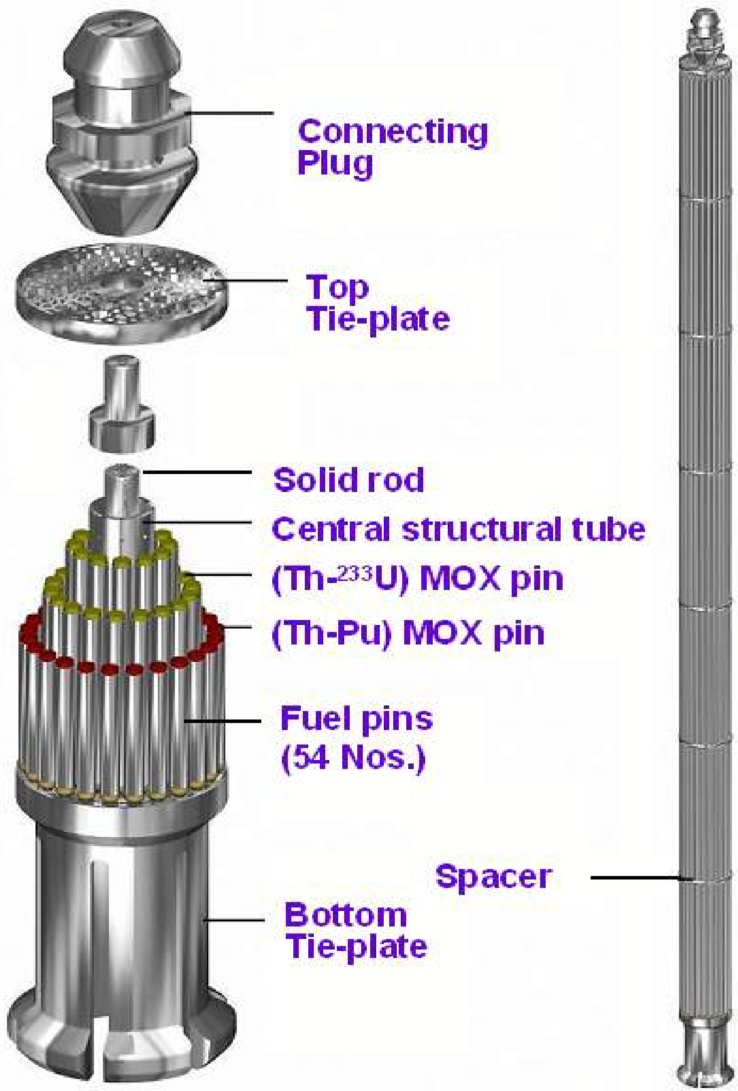
(Th-^{233}U) MOX and (Th-Pu) MOX

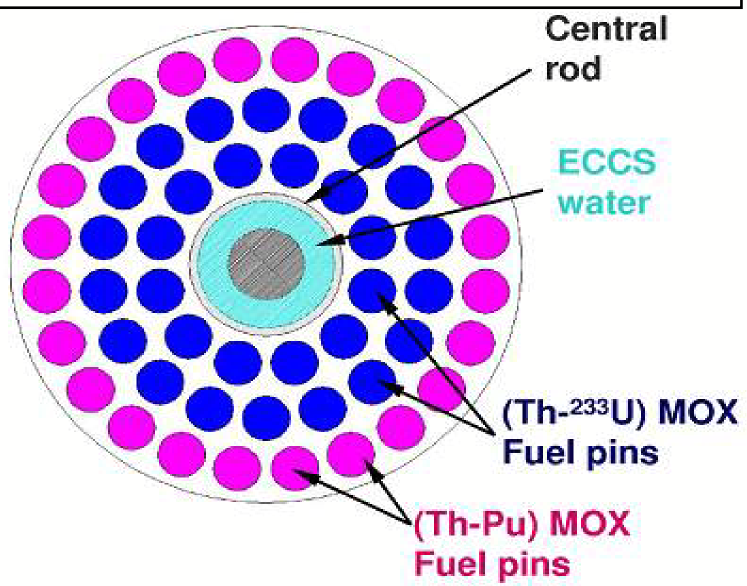
- Closed fuel cycle to maximise energy generation from thoria
- Recycling of self-generated ^{233}U and thoria
- External fissile feed of plutonium

- 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

Fuel Cluster for AHWR

- 54 pins, 3.5m. fuel stack length
- Ø11.2mm, free standing clad





	PCD (mm)	No of pins	Composition
Inner ring	51.4	12	$(Th\text{-}^{233}U)\,O_2$
Middle ring	77.4	18	$(Th\text{-}^{233}U)\,O_2$
Outer ring	103.7	24	$(Th\text{-}Pu)O_2$

AHWR300-LEU with (LEU-Th) MOX Fuel

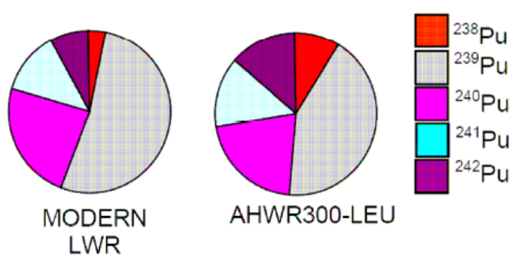
• Proliferation resistance

- 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

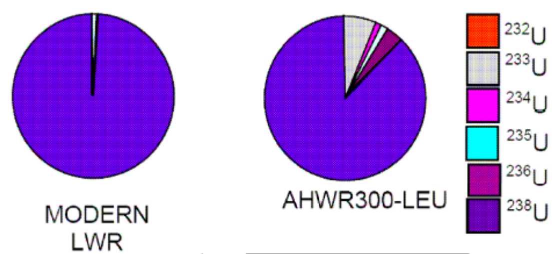
• Waste management

- The AHWR300-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.



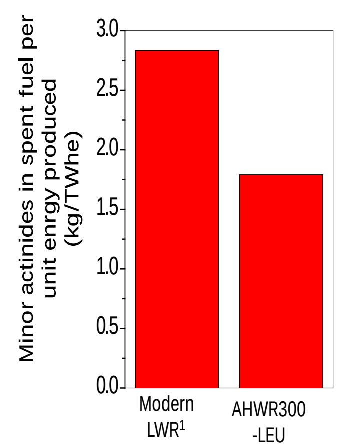
^{238}Pu	3.50	%
^{239}Pu	51.87	%
^{240}Pu	23.81	%
^{241}Pu	12.91	%
^{242}Pu	7.91	%

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



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

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



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Neutronics: effects of thorium in water reactors-1

Parameter	Th-U233	Effect
Moderator temperature Coeff.	Progressively negative with burnup	• Less effect on lattice design changes
Doppler coefficient	More negative, less so with burnup	• Improved transient response to rapid severe reactivity, (hence power) increases
Xenon worth	Slightly less	• Reduces reactor control needed Higher stability against Xenon oscillations
Fission product poisoning	Slightly different	• Only slightly disadvantageous
Delayed neutron fraction, β	Decrease with burnup is slightly more than all-U core β(U233) < β (U235) β(U233) ~ β(Pu239)	• Similar detrimental effect on large reactivity insertion accidents • More rapid power decrease during scram

Neutronics: effects of thorium in water reactors-2

Parameter	Th-U233	Effect
Reactivity loss due to burnup	Appreciably less	• Less poison reactivity requirement at BOC • Burnup prediction more sensitive to errors
Hot to cold reactivity difference	Smaller	• No control modification needed to accommodate use of Th
Control requirements	Reduced overall	• Can reduce burnable poison concentration • Easier to design long cycles/high burnup cores
Local power peaking	Somewhat less	• More thermal hydraulic margins, easier to meet design constraints
Fertile capture product	Pa-233 more important absorber than Np-239	• Delays U-233 production • Both neutrons and U-233 are lost by capture

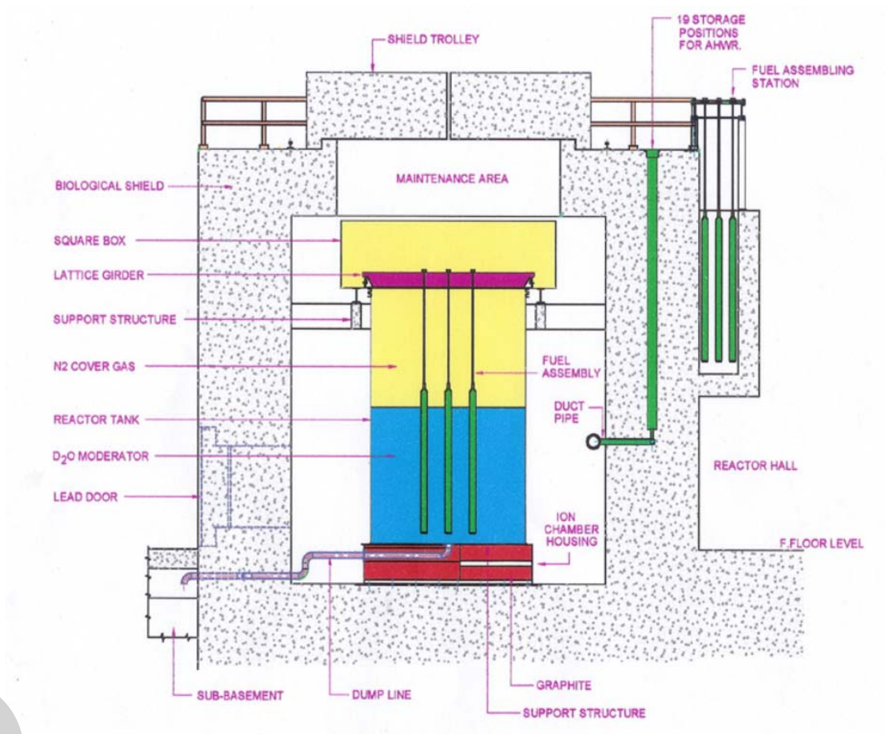
Critical Facility for AHWR

Objective

Validation of Physics Simulation models and nuclear data

Features

- Thermal Neutron Flux (Ave) : 10^8 n/cm^2/s
- Nominal Fission power : 100 W
- Core : 330 cm ID X 500 cm Ht.
- Variable lattice pitch : 20 cm to 30 cm
- Types of cores : Reference core, AHWR Core and PHWR Core

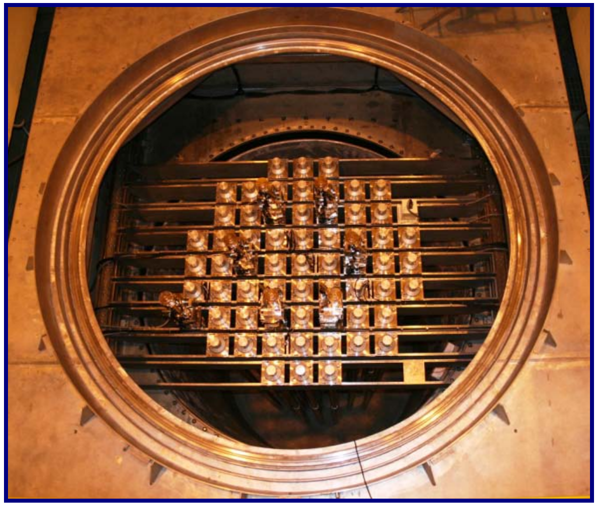


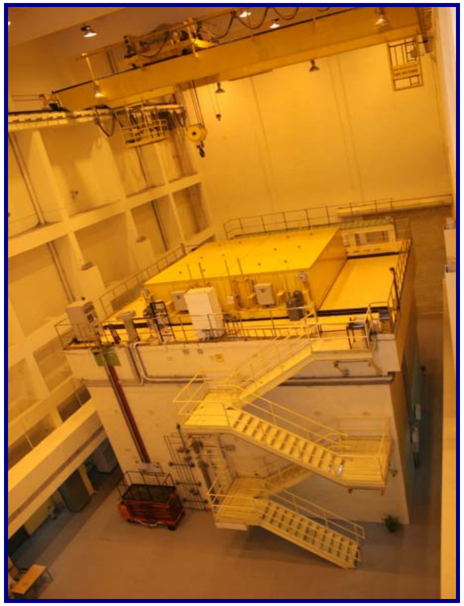
Experiments to be performed

- First approach to criticality in all the types of cores
- Dynamic tests for the shut down device
- Measurement of critical height, level coefficient of reactivity etc.
- Assessing coolant voiding reactivity effects
- Measurement of reaction rates and neutron spectrum
- Neutron flux profile measurement by introducing SPNDs

Critical Facility for AHWR

- AHWR Critical Facility has been designed 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.





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High Temperature Reactors

Compact High Temperature Reactor (CHTR)- Technology Demonstrator

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

Larger power version can also be used as a small nuclear power pack to supply electricity in remote areas

Status: Feasibility studies carried out. Materials, fuel and experimental setups under development

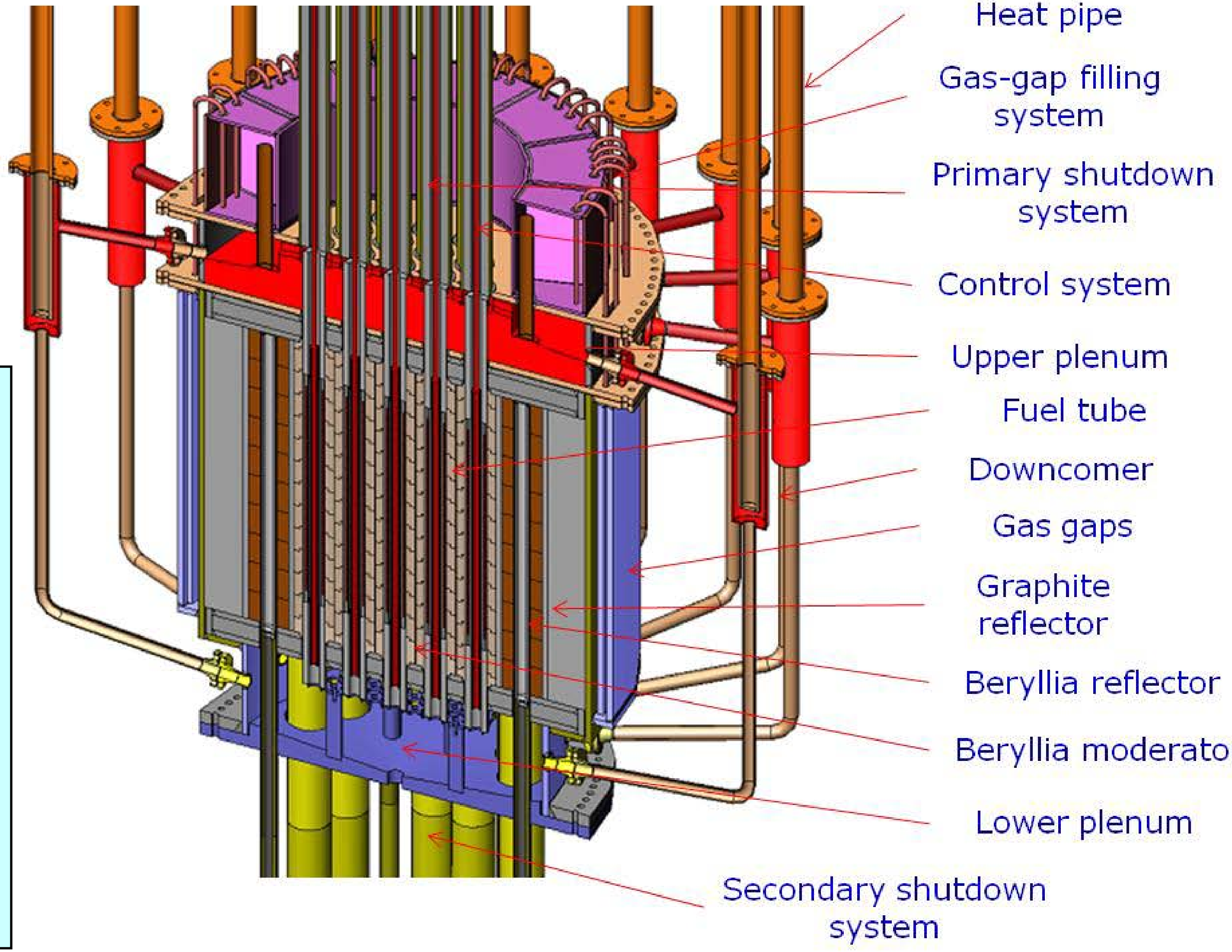
Innovative High Temperature Reactor for Hydrogen Production (IHTR-H)

- 600 MWth , 1000 °C, TRISO coated particle fuel
- Combination of active and passive systems for control and cooling
- Medium life core

Status: Conceptual design carried out
R & D programme for detailed design initiated

In addition a 5 MWth Compact Nuclear Power Pack (550 °C) is also being designed

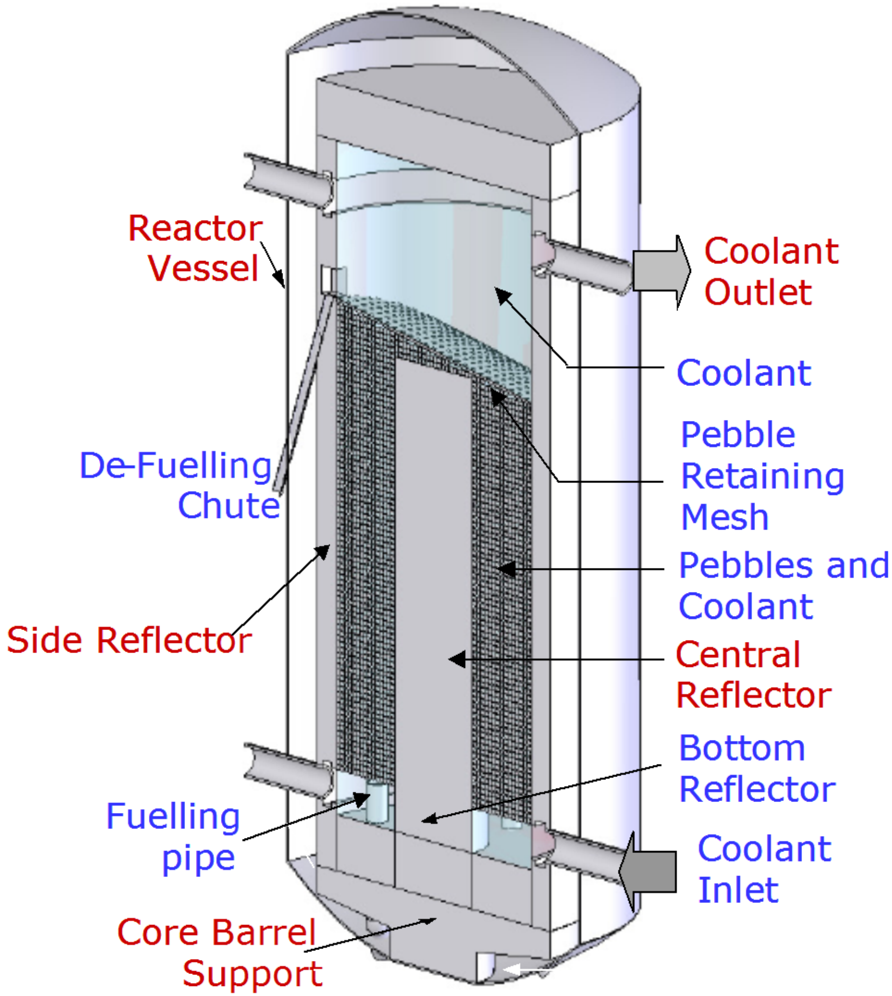
Compact High Temperature Reactor (CHTR)



- Very high temperature (1000 °C)
- Compact
- Lead alloy cooled
- Long life core (15 years)
- Burnup 68 GWd/te
- Passive heat removal
- Ceramic core components

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
- Status:
  - Reactor physics and thermal hydraulic designs being optimised
  - Preliminary thermal and stress analysis carried out
  - Code and experimental set-up under development for simulating pebble motion, pebble feeding and removal



Concluding Remarks

CONCLUDING REMARKS

- Renewed interest in several developed countries on thorium fuels and fuel cycles and their utilization in LWR, PHWR, ACR, HTR, Fast Reactors, MSBR and ADS
- Thorium can play an important role in the near future for market conditions which can arise due to increase in uranium prices.
- Provides intrinsic proliferation resistance in an open fuel cycle.
- Has a very important role to play in India's long-term sustainability of resources.
- India has demonstrated experience in all aspects of thorium fuel cycles
- AHWR will demonstrate large-scale utilisation of thorium using existing technologies
- Commercial demonstration of technologies required for the 3rd stage of Indian nuclear power programme

Thank You