Physics Considerations for Utilisation of Thorium in Power Reactors

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Thorium Energy Conference 2015 (ThEC15), Mumbai, India, October 12–15, 2015
Outline of Talk

- Advantages of Th Fuel Cycle
- Experience with Th fuel cycle in India
  - Experimental Reactors
  - Power Reactors
- Studies for Large Scale Utilization of Th Fuel Cycle in AHWR
- Studies for Large Scale Utilization of Th Fuel Cycle in IMSBR
Advantages of $^{233}\text{U}$-Thorium Cycle

High $\sigma_f / \sigma_c$ favors the feasibility of multiple recycling of $^{233}\text{U}$ compared to Pu.

Capture cross-section of $^{232}\text{Th}$ for thermal neutrons is typically 2.47 times that of $^{238}\text{U}$.

Therefore Thorium offers greater competition to parasitic capture of thermal neutrons and improves the conversion of $^{232}\text{Th}$ to $^{233}\text{U}$. 
Advantages of $^{233}$U-Thorium Cycle

Disposition of Pu:: Thorium vs Inert Matrix

Thorium matrix is as good as Inert Matrix

IM fuel makes the fuel temperature reactivity coefficient so small that it raises serious safety concerns.

However, with thorium matrix negative fuel temperature coefficient is assured

Lower reactivity swing with burn-up
Advantages of $^{233}\text{U}$-Thorium Cycle

Characteristics of Thorium are favorable for high burn-up

- Better thermo-mechanical properties and slower fuel deterioration
- $\text{ThO}_2$ has a very high melting point of 3300°C
- Lower fuel temperatures due to better thermal conductivity of $\text{ThO}_2$
- Less fission gas release
- Better dimensional stability

Fertile $^{238}\text{U}$ converts into $^{239}\text{Pu}$ and fertile $^{232}\text{Th}$ converts into $^{233}\text{U}$

During conversion, at higher burn-up the fissile plutonium saturates at about 0.6% and $^{233}\text{U}$ saturates nearly 1.4%.

Therefore at higher burn-up the thorium as fertile host overtakes $^{238}\text{U}$ as host.
Advantages of $^{233}$U-Thorium Cycle

Due to presence of $^{232}$U in separated $^{233}$U, thorium offers good proliferation-resistant characteristics.
Thorium utilisation in India - Research reactors

- **PURNIMA-II (1984-1986):** Uranyl nitrate solution fuelled experimental reactor

- **PURNIMA-III (1990-1993):** U-Al dispersion fuel

- **KAMINI (1996-till date):** A $^{233}$U fuelled test reactor KAMINI commissioned in 1996

- **CIRUS:** (1) Irradiation of Thoria rods in CIRUS (J rods locations) (2) (Th,Pu) MOX fuel was irradiated and tested for high burn-up in dedicated engineering loops of CIRUS

- **Dhruva:** Irradiation of a few Thoria bundles in the Dhruva core

- **FBTR:** Thoria bundles irradiated in the FBTR blanket
Thorium utilisation in the India - Power reactors

- 35 Thoria bundles loading in the initial core of Indian PHWRs
  - Same level of flattening was achieved by using 35 Thoria bundles
  - Six PHWRs adopted the same loading pattern to irradiate total 210 Thoria bundles
  - Irradiation was varied from 200 to 600 FPDs
  - No fuel failures
  - Maximum burn-up ~ 14,000 MWd/Te
  - PIE / chemical/ spectroscopic analysis of samples from these bundles
  - Measured $^{232}$U was ~ 500 ppm

- RAPS-2 startup after EMCCCR used 18 ThO$_2$ for flux flattening of Initial Core

ThO$_2$ bundles loading in Initial core of PHWR
Exploring options for using thorium in current generation reactor configurations

Objectives:

- Proliferation resistance
- Disposition of weapons grade fissile materials
- All reactivity coefficients are within required limits
- Maximise fissile consumption, minimise waste generation and plutonium in the spent fuel
- Lower fuel cycle cost per unit energy produced

A once-through fuel cycle has been considered for near-term deployment. For long term sustainability, a closed fuel cycle is required.
Thorium cycles in Heavy Water Reactors

- Self-sustaining equilibrium thorium cycle in PHWRs
  - Fissile feed - $^{233}\text{U}$
    - $^{233}\text{U}$ content in thermal reactors
      - ~ 1.5 % H.M. - Self-sustaining characteristic
    - Low burn-up ~ 11,000 MWD/T
    - Cycle can be initiated by $^{235}\text{U}$ or Pu
- Once through Thorium (OTT)
  - Use of pure thorium with SEU in segregated channels
- (Th,Pu) MOX cycles in PHWRs
  - MOX-TH24 (2.4% Pu) - 15,000 MWD/T
  - MOX-TH20 (2.0% Pu) - 8,000 MWD/T
  - Presence of Pu lowers $\beta_{\text{eff}}$ initially
  - Worth of shut-down systems lower than $\text{UO}_2$
Advanced Heavy Water Reactor: Features

AHWR:

- Power: 300 MWe
- Vertical
- Pressure tube type
- Coolant: Boiling light water
- Moderator: Heavy water

Design Features of AHWR:

- Natural Circulation
- Negative Void Coefficient
- Online Refueling
- 100 year design life
Advanced Heavy Water Reactor: Fuel

- **AHWR-Ref Core: (Th$^{233}$U-Pu)MOX fuel in closed cycle mode**

- **AHWR-LEU Core: Th-LEU (19.75 wt % U-235) MOX fuel in once through mode**
Advanced Heavy Water Reactor: Passive Features

- Core cooling by natural circulation
- Negative void coefficient of reactivity
- Large heat sink as Gravity Driven Water Pool
- Passive Core decay heat removal
- Passive containment cooling
- Emergency core cooling injection in Fuel
- Passive poison injection in moderator in the event of non-availability of both the primary as well as secondary shut down system due to failure or malevolent insider
- Core submergence
- Double Containment
Fuel-cycle in AHWR-Ref Core

**Initial core**
(DU, Pu)MOX fuel

Initial Core refueled with Transition Clusters (Th, Pu) MOX fuel

**Transition core**
(Th, Pu)MOX fuel

Transition Core refueled with Equilibrium Clusters (Th, Pu-U) MOX fuel

**Equilibrium core**
(Th, Pu-U)MOX
Fuel-cycle in AHWR-LEU

Initial core
(Th, 13\% LEU)MOX

Initial Core refueled with Equilibrium Clusters (Th, LEU) MOX fuel

Equilibrium core
(Th, 21.8\% LEU)MOX
Equilibrium Core

AHWR Reference Core

<table>
<thead>
<tr>
<th>Axial Gradation</th>
<th>PU / $^{233}$U content (%) in cluster rings</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner</td>
<td></td>
</tr>
<tr>
<td>Middle</td>
<td></td>
</tr>
<tr>
<td>Outer</td>
<td></td>
</tr>
<tr>
<td>Upper</td>
<td>6.0% Pu</td>
</tr>
<tr>
<td></td>
<td>3.9% Pu</td>
</tr>
<tr>
<td></td>
<td>3.3% U</td>
</tr>
<tr>
<td>Lower</td>
<td>6.0% Pu</td>
</tr>
<tr>
<td></td>
<td>3.9% Pu</td>
</tr>
<tr>
<td></td>
<td>3.8% U</td>
</tr>
</tbody>
</table>

AHWR-LEU Core

<table>
<thead>
<tr>
<th>Axial Gradation</th>
<th>LEU content (%) in cluster rings</th>
<th>U-235 Content</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner</td>
<td>Middle</td>
<td>Outer</td>
</tr>
<tr>
<td>Upper</td>
<td>30</td>
<td>24</td>
</tr>
<tr>
<td>Lower</td>
<td>30</td>
<td>24</td>
</tr>
</tbody>
</table>
## Important Design Parameters of AHWR

<table>
<thead>
<tr>
<th>Parameters</th>
<th>AHWR-Reference</th>
<th>AHWR-LEU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Power</td>
<td>300 MWe (920MWth)</td>
<td>300 MWe (920MWth)</td>
</tr>
<tr>
<td>Fuel</td>
<td>(Th, U) MOX + (Th, Pu) MOX</td>
<td>(Th, LEU) MOX</td>
</tr>
<tr>
<td>Coolant</td>
<td>Boiling Light Water</td>
<td>Boiling Light Water</td>
</tr>
<tr>
<td>Moderator</td>
<td>Heavy Water</td>
<td>Heavy Water</td>
</tr>
<tr>
<td>Fuelling</td>
<td>On Power (mini batch)</td>
<td>On-power (Mini Batch)</td>
</tr>
<tr>
<td>Discharge Burn-up (Avg.)</td>
<td>40 GWd/te</td>
<td>60 GWd/te</td>
</tr>
<tr>
<td>Total No. of Channels</td>
<td>513</td>
<td>513</td>
</tr>
<tr>
<td>No. of Fuel Channels</td>
<td>452</td>
<td>444</td>
</tr>
<tr>
<td>Lattice Pitch</td>
<td>225 mm</td>
<td>225 mm</td>
</tr>
<tr>
<td>Primary Shut down System</td>
<td>37 shut off rods</td>
<td>45 shut off rods</td>
</tr>
<tr>
<td>Secondary Shut down System</td>
<td>Liquid poison injection in moderator</td>
<td>Liquid poison injection in moderator</td>
</tr>
<tr>
<td>No. of Control Rods (AR, RR &amp; SRs)</td>
<td>24 (8+8+8)</td>
<td>24 (8+8+8)</td>
</tr>
</tbody>
</table>
# Important Design Parameters of AHWR

<table>
<thead>
<tr>
<th>Parameters</th>
<th>AHWR-Reference</th>
<th>AHWR-LEU</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel type / No. of pins in</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ring 1 (10 Fuel Pins)</td>
<td>(Th, Pu) MOX / 6 % Pu</td>
<td>(Th, LEU) MOX / 30 % LEU</td>
</tr>
<tr>
<td>Ring 1 (2 Fuel Pins)</td>
<td>(Th, Pu) MOX / 6% Pu &amp; 2% Gd</td>
<td>(Th, LEU) MOX / 30 % LEU &amp; 5% Gd</td>
</tr>
<tr>
<td>Ring 2 (18 Fuel Pins)</td>
<td>(Th, Pu) MOX / 3.9 % Pu</td>
<td>(Th, LEU) MOX / 24 % LEU</td>
</tr>
<tr>
<td>Ring 3 (24 Fuel Pins, Axial Gradation)</td>
<td>(Th, U) MOX</td>
<td>(Th, LEU) MOX</td>
</tr>
<tr>
<td></td>
<td>Bot: 3.8 % U (80% fissile)</td>
<td>Bottom: 18 % LEU</td>
</tr>
<tr>
<td></td>
<td>Top: 3.3 % U (80% fissile)</td>
<td>Top: 14 % LEU</td>
</tr>
<tr>
<td>No. of Control rods (AR/RR/SR), worth ,mk</td>
<td>24 (8each) 10.8/11.1/10.5</td>
<td>24 (8each) 12.2/12.8/9.4</td>
</tr>
<tr>
<td>RR worth in nominal configuration, mk</td>
<td>6.4</td>
<td>6.8</td>
</tr>
<tr>
<td>Shut Down system-1 ,Worth(mk)/ No. of SOR</td>
<td>-69.4 (37 SOR)</td>
<td>-78.8 (45 SOR)</td>
</tr>
<tr>
<td>Worth when 2 max. worth rods not available, mk</td>
<td>-49.2 (35 SOR)</td>
<td>-56.0 (43 SOR)</td>
</tr>
<tr>
<td>Power from thorium</td>
<td>61%</td>
<td>38%</td>
</tr>
</tbody>
</table>

**Reactivity coefficients:**

- **Fuel temperature**, $\Delta k / k / ^\circ K$:
  - AHWR-Reference: $-21.2 \times 10^{-6}$
  - AHWR-LEU: $-25.5 \times 10^{-6}$

- **Channel temperature**, $\Delta k / k / ^\circ K$:
  - AHWR-Reference: $+16.1 \times 10^{-6}$
  - AHWR-LEU: $-29.9 \times 10^{-6}$

- **Void coefficient (0.45 to 0.0 g/cc)**, $\Delta k / k / $ % void:
  - AHWR-Reference: $-120.0 \times 10^{-6}$
  - AHWR-LEU: $-25.0 \times 10^{-6}$

- **Moderator temperature**, $\Delta k / k / ^\circ K$:
  - AHWR-Reference: $+29.6 \times 10^{-6}$
  - AHWR-LEU: $-17.4 \times 10^{-6}$
Power fraction from thorium in AHWR

- Power from $^{233}\text{U}/\text{Th}$ is high (~60%) at core average burn-up of 20 GWd/Te in AHWR-Ref core.

- Power from $^{233}\text{U}/\text{Th}$ is relatively low (~38%) at core average burn-up of 30 GWd/Te in AHWR-LEU core.
<table>
<thead>
<tr>
<th>Burnup</th>
<th>(^{232}\text{U})</th>
<th>(^{233}\text{U})</th>
<th>(^{234}\text{U})</th>
<th>(^{235}\text{U})</th>
<th>(^{236}\text{U})</th>
<th>(^{237}\text{U})</th>
<th>(^{238}\text{U})</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0 GWd/Te</td>
<td>0.25</td>
<td>77.6</td>
<td>17.6</td>
<td>3.5</td>
<td>1.0</td>
<td>0.0</td>
<td>0.05</td>
</tr>
<tr>
<td>(40 GWd/Te)</td>
<td>0.25</td>
<td>72.9</td>
<td>20.5</td>
<td>4.4</td>
<td>1.9</td>
<td>0.0</td>
<td>0.05</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Burnup</th>
<th>(^{238}\text{Pu})</th>
<th>(^{239}\text{Pu})</th>
<th>(^{240}\text{Pu})</th>
<th>(^{241}\text{Pu})</th>
<th>(^{242}\text{Pu})</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0 GWd/te</td>
<td>0.0</td>
<td>68.8</td>
<td>24.6</td>
<td>5.3</td>
<td>1.3</td>
</tr>
<tr>
<td>(40 GWd/te)</td>
<td>1.8</td>
<td>13.6</td>
<td>48.7</td>
<td>21.5</td>
<td>14.4</td>
</tr>
</tbody>
</table>

- Presence of significant amount of \(^{232}\text{U}\) in discharged uranium makes it proliferation resistant
- Uranium in freshly loaded fuel at 0 GWd/Te has ~80% fissile content
- Uranium in discharged fuel at 40 GWd/Te has ~75% fissile content
- The plutonium discharged has 35.1% fissile Pu.
- There is efficient burning of plutonium in AHWR-reference, the fissile content reduces from 75% to 35%
### Isotopic vector of U/Pu in Equilibrium Fuel Cluster of AHWR-LEU Core

<table>
<thead>
<tr>
<th>Burnup</th>
<th>$^{232}\text{U}$</th>
<th>$^{233}\text{U}$</th>
<th>$^{234}\text{U}$</th>
<th>$^{235}\text{U}$</th>
<th>$^{236}\text{U}$</th>
<th>$^{237}\text{U}$</th>
<th>$^{238}\text{U}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0 GWd/Te</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>19.75</td>
<td>0.0</td>
<td>0.0</td>
<td>80.25</td>
</tr>
<tr>
<td>(60 GWd/Te)</td>
<td>0.0</td>
<td>5.83</td>
<td>1.14</td>
<td>2.38</td>
<td>3.24</td>
<td>0.0</td>
<td>87.39</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Burnup</th>
<th>$^{238}\text{Pu}$</th>
<th>$^{239}\text{Pu}$</th>
<th>$^{240}\text{Pu}$</th>
<th>$^{241}\text{Pu}$</th>
<th>$^{242}\text{Pu}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0 GWd/Te</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>(60 GWd/T)</td>
<td>8.6</td>
<td>42.8</td>
<td>21.1</td>
<td>14.7</td>
<td>12.8</td>
</tr>
</tbody>
</table>

- The uranium in (Th, LEU) MOX fuel **discharged** at 60 GWd/Te has 8.2% fissile content.
- **The discharged $^{233}\text{U}$ is denatured by the presence of $^{238}\text{U}$**
- $^{238}\text{Pu}$ content in discharged plutonium of AHWR-LEU is about 8.6% Which makes it proliferation resistant.
- The plutonium discharged in AHWR-LEU core has 57.5% fissile plutonium but its amount is very small.
Fuel economics

AHWR-Reference Core (Equilibrium Core):

• Requirements:
  – Annual requirement of U (~80% fissile): ~ 90 Kg
  – Annual requirement of Pu (~75% fissile): ~ 150 Kg

• Discharged (40 GWd/Te):
  – Uranium (~75% fissile) discharged annually: ~ 110 Kg
  – Plutonium (~35% fissile) discharged annually: ~ 54 Kg
  – M. A. (excluding Pu) discharged annually: ~ 5.4 Kg

AHWR-LEU (Equilibrium Core):

– Annual LEU (19.75% fissile U) Requirement: 760 Kg
– M. A. (excluding Pu) discharged annually: ~ 2.4 Kg
## Natural Uranium Resources Utilisation

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Reactor systems for Comparison</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>KK Project (1000 MWe)</td>
</tr>
<tr>
<td>Burnup</td>
<td>43 GWd/Te</td>
</tr>
<tr>
<td>Fission Power</td>
<td>3000 MW</td>
</tr>
<tr>
<td>U-235 in Tails</td>
<td>0.2%</td>
</tr>
<tr>
<td>Capacity Factor</td>
<td>100%</td>
</tr>
<tr>
<td>Annual Uranium Required</td>
<td>25.5 Te (\textsuperscript{235}U~3.92%)</td>
</tr>
<tr>
<td>Mined Uranium (Annual)</td>
<td>190 Te</td>
</tr>
<tr>
<td>TWHrs (annual)</td>
<td>8.76</td>
</tr>
<tr>
<td>Nat. U (Te) per TWhr</td>
<td>21.7</td>
</tr>
</tbody>
</table>
• The AHWR-Ref design is good: The fuel requirement is less and power from thorium (app. 60 %) is large

AHWR-LEU Design

• LEU (19.75% $^{235}\text{U}$): A good external feed in thorium oxide fuel in AHWR having all reactivity coefficients negative

• LEU content of 21.7% mixed with ThO$_2$ gives high discharge burnup of about 60 GWd/Te

• Fissile uranium requirement per unit energy is significantly lower in case of AHWR-LEU

• About 38% of power comes from $^{233}\text{U}/\text{Th}$

• The uranium in the discharged fuel has 8.4% fissile content which can be used in other reactors
SUMMARY

- The discharged uranium is denatured and contains significant amount of $^{232}\text{U}$
- The discharged Pu has 57% fissile and ~8.5% $^{238}\text{Pu}$
- Plutonium produced per unit energy is much less in case of AHWR-LEU
- The presence of $^{232}\text{U}$ in reprocessed uranium and $^{238}\text{Pu}$ in reprocessed plutonium enhances its proliferation resistance characteristics
- Minor actinides produced per unit energy in this reactor is also less
Molten Salt Breeder Reactor
Molten Salt Reactor: Features

- Online refueling & reprocessing
- Continuous removal of gaseous fission products
- Better neutron economy & high breeding
- No radiation damage limit to burnup
- Better fuel utilization even in once-through cycle
- Low level of long lived actinides wastes
- Large –ve feedback & void coefficient of reactivity
- Fuel drained to critically safe storage tank during accident

Theoretically, possible to have breeder with Th-233U fuel cycle in thermal, epithermal and fast spectrum.
## Thermal, Epithermal or Fast MSRs

<table>
<thead>
<tr>
<th></th>
<th>Thermal</th>
<th>Epithermal</th>
<th>Fast</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fissile Inventory/ GWe</td>
<td>Least (~0.9 T of (U^{233}))</td>
<td>~ 2 T of (U^{233})</td>
<td>~ 5 T of (U^{233}) #</td>
</tr>
<tr>
<td>Breeding Ratio</td>
<td>Slightly &gt; 1 (~ 1.01)</td>
<td>~1.004 ; Practically difficult to achieve</td>
<td>~ 1.12</td>
</tr>
<tr>
<td>Moderator Life time</td>
<td>Graphite (~ 6 years)</td>
<td>Graphite (~ 1.5 years)</td>
<td>-</td>
</tr>
<tr>
<td>Feedback coefficient (pcm/K)</td>
<td>Tolerable feedback coefficient with good BR is difficult. (~ -1.0)</td>
<td>Tolerable feedback coefficient with good BR is difficult. ( ~ -2.4)</td>
<td>Large –ve feedback coefficient ~-5.3</td>
</tr>
<tr>
<td>Design Flexibility Margin</td>
<td>Least flexible</td>
<td>Least flexible</td>
<td>Most flexible</td>
</tr>
<tr>
<td>Core design complexity</td>
<td>Complex</td>
<td>Complex</td>
<td>Simple</td>
</tr>
<tr>
<td>Poisoning (FP)</td>
<td>Most (faster reprocessing required to make breeder)</td>
<td>Most (faster reprocessing required to make breeder)</td>
<td>Least</td>
</tr>
<tr>
<td>Production of TRUs</td>
<td>&gt; fast reactor</td>
<td>&gt; fast reactor</td>
<td>Least</td>
</tr>
</tbody>
</table>

* The numbers shown are for some optimized cases of TMSR (French) in thermal, epithermal and fast spectrum region.

# MSFR ~ 3.26 T of \(U^{233}\)

❑ Advantage
❑ Disadvantage
❑ Theoretically acceptable
# MSR Concepts Worldwide

<table>
<thead>
<tr>
<th>Family</th>
<th>Concept (Country)</th>
<th>Spectrum</th>
<th>Fuel Cycle</th>
<th>Power ($MW_{th}$)</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>MSR- (Breeder/Near-breeder)</td>
<td>MSBR (US)</td>
<td>T</td>
<td>233U-Th</td>
<td>2250</td>
<td>BR $\sim$ 1.05; FRC &gt; 0 (Slightly +ve)</td>
</tr>
<tr>
<td></td>
<td>AMSTER-B (F)</td>
<td>T</td>
<td>233U-Th</td>
<td>2250</td>
<td>BR &gt; 0.95</td>
</tr>
<tr>
<td></td>
<td>REBUS* (F)</td>
<td>F</td>
<td>U-Pu</td>
<td>3700</td>
<td>BR $\sim$ 1.03; FRC &lt; 0</td>
</tr>
<tr>
<td></td>
<td>FUJI (J)</td>
<td>T</td>
<td>233U-Th</td>
<td>450</td>
<td>BR $\sim$ 0.97; FRC &lt; 0</td>
</tr>
<tr>
<td></td>
<td>TMSR (F)</td>
<td>T, E, F</td>
<td>233U-Th</td>
<td>2500</td>
<td>BR &gt; 1 &amp; FRC &lt; 0 in both T, F</td>
</tr>
<tr>
<td></td>
<td>MSFR# (F)</td>
<td>F</td>
<td>233U-Th</td>
<td>3000</td>
<td>BR $\sim$ 1.12 &amp; FRC &lt; 0</td>
</tr>
<tr>
<td>MSR-Burner</td>
<td>AMSTER-I (F)</td>
<td>T</td>
<td>U-Pu-MA</td>
<td>2250</td>
<td></td>
</tr>
<tr>
<td></td>
<td>SPHINX (Cz)</td>
<td>F</td>
<td>Pu-MA</td>
<td>1208</td>
<td></td>
</tr>
<tr>
<td></td>
<td>MOSART (R)</td>
<td>F</td>
<td>Pu-MA</td>
<td>2400</td>
<td>FRC $\sim$ -3.9 pcm/k</td>
</tr>
</tbody>
</table>

Note: China launched 2 $MW_{th}$ research Thorium Molten-Salt fuelled and cooled Reactor (TMSR) in 2011 scheduled for completion in 2020.

(F) – French, (J) – Japan, (Cz) – Czech, (R) - Russia

* Chloride based fuel salts
# MSFR derived from non-moderated TMSR
Two design concepts were analysed for possible core configuration of IMSBR:

- A Loop type: A smaller core with fuel and coolant salts as one salt and flowing out of the core in external circuits and coming back.
- A Pool type: The core is submerged in a pool of molten fuel/coolant salt.
Molten Salt Fast Reactor (MSFR)

The simulation methodology for IMSBR have been first examined by analysing and producing some of key results of French MSFR concept.

- Core Dimension (m): 2.2 X 2.2
- Power : 3000 MW\(_{th}\)/ 1300 MW\(_e\)
- Power density : 330 W\(_{th}\)/ cm\(^3\)
- Breeding Ratio : 1.12
- Initial fissile inventory : 3.26 T/Gw\(_e\)
- Mean Fuel salt temperature : 750 °C

- Fuel Salt : LiF (77.5%)-ThF\(_4\) (20%)-\(^{233}\)UF\(_4\) (2.5%)
- Blanket Salt : LiF (77.5%)-ThF\(_4\) (22.5%)
MSFR core simulation: Results

- MSFR found to be CRITICAL for given composition
- Breeding Ratio: 1.17
- Initial fissile inventory: 3.23 T/GW$_e$

Comparison of reactor spectrum with published results (left)
Indian MSR Concept: Loop type design

- Fuel Salt: LiF (77.6%) - ThF₄ (19.7%) - ^{233}\text{UF}_4 (2.7%)
- Blanket Salt: LiF (77.6%) - ThF₄ (22.4%)
- Core Dimension (m): 2 x 2
- Power: 1900 MWₜₜ/ 850 MWₑ
- Power density: 300 Wₜₜ/ cm³
- Breeding Ratio: 1.09
- Initial fissile inventory: 3.8 T (4.27 T/GWₑ)
- Forced circulation

<table>
<thead>
<tr>
<th>Composition: LiF-ThF₄-UF₄</th>
<th>He (% mol)</th>
<th>$K_{eff}$</th>
<th>ICR_fuel</th>
<th>ICR_blanket</th>
<th>ICR_total</th>
</tr>
</thead>
<tbody>
<tr>
<td>77.6% - 19.9% - 2.5%</td>
<td>0%</td>
<td>0.9975</td>
<td>0.988</td>
<td>0.092</td>
<td>1.08</td>
</tr>
<tr>
<td>77.6% - 19.9% - 2.5%</td>
<td>0.5%</td>
<td>1.0006</td>
<td>0.979</td>
<td>0.086</td>
<td>1.065</td>
</tr>
<tr>
<td>77.6% - 19.7% - 2.7%</td>
<td>0%</td>
<td>1.0428</td>
<td>0.998</td>
<td>0.093</td>
<td>1.091</td>
</tr>
<tr>
<td>77.6% - 19.7% - 2.7%</td>
<td>0.5%</td>
<td>1.0447</td>
<td>0.989</td>
<td>0.089</td>
<td>1.078</td>
</tr>
<tr>
<td>77.6% - 19.4% - 3.0%</td>
<td>0%</td>
<td>1.1024</td>
<td>1.011</td>
<td>0.097</td>
<td>1.108</td>
</tr>
</tbody>
</table>
Indian MSR Concept: Pool type design

- Fuel Salt: LiF (77.6%)-ThF₄ (19.7%)-²³³UF₄ (2.7%)
- Blanket Salt: LiF (77.6%)-ThF₄ (22.4%)
- Core Dimension (m): 2m X 2 m
- Power: 1900 MWₜₙ/ 850 MWₑ
- Power density: 300 Wₜₙ/ cm³
- Breeding Ratio: 1.1
- Initial fissile inventory: 5.4 T (6.1 T/GWₑ)
- Natural circulation

<table>
<thead>
<tr>
<th>Composition: LiF-ThF₄-UF₄</th>
<th>He (% mol)</th>
<th>Keff</th>
<th>ICR (fuel)</th>
<th>ICR(blanket)</th>
<th>ICR (total)</th>
</tr>
</thead>
<tbody>
<tr>
<td>77.6% -19.9% -2.5%</td>
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<td>0.9852</td>
<td>0.962</td>
<td>0.150</td>
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<tr>
<td>77.6% -19.7% -2.7%</td>
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<td>0.972</td>
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<td>1.125</td>
</tr>
<tr>
<td>77.6% -19.4% -3.0%</td>
<td>0</td>
<td>1.0872</td>
<td>0.988</td>
<td>0.155</td>
<td>1.143</td>
</tr>
<tr>
<td>77.6% -19.9% -2.5%</td>
<td>0.5</td>
<td>0.9872</td>
<td>0.953</td>
<td>0.142</td>
<td>1.095</td>
</tr>
</tbody>
</table>

Motor of primary circulation pump

Secondary coolant salt inlet/ outlet

Blanket salt Inlet/outlet

Primary HX

Reflectors

Core

Sacrificial salt layer

Schematic of pool type MSR based on natural circulation
Indian MSR Concept: Refueling studies

- Analysis carried by lattice code ITRAN
- K-eff evaluated by supplying geometry buckling in ITRAN input
- Refueling and removal of molten salt fuel possible at any desired burnup
- Possible to analyze the effect of addition and removal of any of 96 nuclides at any desired burnup

Figure 2: Two step refueling & effect of removal of different amount of fuel
Safety Studies for Circulating Fuel Reactors

Loss of Reactivity due to Fuel Flow

- The effective delayed neutron fraction is computed as function of the flow rate of $^{235}\text{U}$ based fuel salt.
- The reactivity loss due to fuel circulation is close to saturation around nominal flow rates. It means that small variations in flow rate can insert or remove a very little amount of reactivity.
- At low flow rate, little modifications in the operating conditions cause high reactivity insertion/removal.

![Fig.3: Effective delayed neutron fraction vs fraction of nominal flow rate.](image)

Table 2 Beta eff for static and circulating $^{235}\text{U}$ and $^{233}\text{U}$ based fuel salt

<table>
<thead>
<tr>
<th>Library</th>
<th>Beta eff Static (pcm)</th>
<th>Beta eff Circulating (pcm)</th>
<th>Reference (pcm)</th>
<th>Beta eff Static (pcm)</th>
<th>Beta eff Circulating (pcm)</th>
<th>Reference (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Jeff 3.1</td>
<td>701</td>
<td>274.81</td>
<td>275</td>
<td>305</td>
<td>118</td>
<td>124</td>
</tr>
<tr>
<td>ORNL</td>
<td>666</td>
<td>261.09</td>
<td>------</td>
<td>264</td>
<td>102.6</td>
<td>------</td>
</tr>
</tbody>
</table>
Point-Kinetics Model of CFR with Thermal feedback

- A program has been developed to solve first order coupled set of equations using delay differential equation solver.

- The loss of reactivity at reference velocity of fuel salt for $^{233}\text{U}$ and $^{235}\text{U}$ has been computed.

- The reactor operation with $^{233}\text{U}$ shows 41% loss of reactivity as compared to 37% loss for $^{235}\text{U}$ based fuel.

Response of MSRE with U-233 and U-235 loading to the reactivity insertion of 50pcm calculated by the 0D-0D code.
A 5 MWth Experimental MSR with Low Inventory

- Core simulated from Monte Carlo code OpenMC
- Fuel Composition: LiF₄ (77.6%) – ²³²ThF₄ (13.4%) – ²³³UF₄ (9%)
- Blanket Salt: LiF₄ (77.6%) – ²³²ThF₄ (22.4%)
- Fissile inventory (U-233): 30.2 kg
- U-233 Content: 40.1% in HM
- K-eff: 1.034 ± 0.0005
- Fuel salt volume: 30000 cm³
- Radius: 22 cm; Height: 20 cm
- Fast Spectrum

- Note: Analysis carried in static, hot operating condition and detailed analysis under progress
Simulation methodology was examined by simulating MSFR and reproducing key results.

Two types of core configurations were analyzed viz. Loop type and Pool type IMSRs.

Fast spectrum offers greater design flexibility and higher BR.

MSR – Long term alternative to solid fuel reactors with several advantages and unique potential e.g. inherently safe, breeder, high burnup and better fuel utilization etc.