RECYCLE FUEL FABRICATION IN INDIA

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IAEA TM on “Country Nuclear Fuel Cycle Profiles” at Fukui, Japan on 1-2 Dec, 2008
Outline

1. Introduction.
2. MOX fuel fabrication for BWR & PHWR.
3. Mixed Carbide fuel fabrication for FBTR.
4. PFBR MOX fuel development & fabrication.
5. AHWR MOX fuel development
“There is no power as costly as No-Power” – Homi Bhabha

Nuclear Energy would provide Prosperity to developing Nations--
FIRST UN Conf PUAE, Geneva, 1955

Source of the Data: World Bank, 1999
Three Stage Indian Nuclear Power Programme

1. To **maximize** the energy potential of **Uranium** and empower **Thorium**.

2. **Self-Reliance** for Energy security and **Energy Independence**.

3. **Minimize** emissions to the environment while meeting the huge demand for electricity.

Closed fuel cycle is the integral part of this strategy.
India is one of the fastest growing economy.
- Recorded 9% GDP growth IN 2007-2008.

Population is over 1 billion but India’s power consumption is about 600 kWh well below the global average of 2500 kWh.

Demand for energy continues to rise because of the growth in economy & population.
India’s GDP to Grow Even During Current Global Financial Crisis

- 2/3 of India’s one billion people live in energy deficiency.
- Out of 600,000 villages, about 25% have no grid electricity.
- Domestic savings rate is 35% of GDP.
- Worlds Largest young demographic profile.

Projected GDP growth rates for India, US & Japan

IMF Projections (Nov. 2008)
**Indian Installed Electric Capacity**

<table>
<thead>
<tr>
<th>Source</th>
<th>%</th>
<th>Capacity (MW)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coal</td>
<td>55%</td>
<td>68,300</td>
</tr>
<tr>
<td>Gas</td>
<td>10%</td>
<td>12,300</td>
</tr>
<tr>
<td>Oil</td>
<td>1%</td>
<td>1,200</td>
</tr>
<tr>
<td>Hydro</td>
<td>26%</td>
<td>32,135</td>
</tr>
<tr>
<td>Renewable</td>
<td>4.5%</td>
<td>6,150</td>
</tr>
<tr>
<td>Nuclear</td>
<td>3.5%</td>
<td>4,120</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td>~130 GW</td>
</tr>
</tbody>
</table>

**Projected Installed Power Capacity**

Projected per capita (kWh)

Projected Installed Power Capacity
India’s green house gas emission (GHG) is the lowest in per-capita terms.

4% of the World GHG emission in spite of having 17% of the World population.

Rapid economic growth is leading to increased demand for electricity. Nuclear energy has to play an important role.

Dec.1-2, 2008
Indian Nuclear Energy Resource Position

<table>
<thead>
<tr>
<th>Resource</th>
<th>Quantity (Tonnes)</th>
<th>Energy Potential (GWe-Yr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium</td>
<td>61,000</td>
<td>328 in PHWR</td>
</tr>
<tr>
<td></td>
<td></td>
<td>42,330 in FBR</td>
</tr>
<tr>
<td>Thorium</td>
<td>300,000</td>
<td>155,500 in Breeders</td>
</tr>
</tbody>
</table>

Uranium is 1% of the World resources. Thorium resource is one of the largest in the World.
Closed Fuel Cycle

1. Resource extension & Sustainability.
2. Waste Classification & Isolation.
3. Reduction in Repository space.
PRESENT STAGE OF NUCLEAR POWER PROGRAM

**Stage - I**

**PHWRs**
- 15 Operating
- 3- Under construction
- Several others planned
- POWER POTENTIAL ≅ 10,000 MWe

**LWRs**
- 2 BWRs Operating
- 2 VVERs under construction

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**Stage - II**

**Fast Breeder Reactors**
- 40 MWth FBTR - Operating Technology Objectives realised
- 500 MWe PFBR - Under Construction
- POWER POTENTIAL ≅ 530,000 MWe

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**Stage - III**

**Thorium Based Reactors**
- 30 kWth KAMINI - Operating
- 300 MWe AHWR - Under development
- POWER POTENTIAL ≅ Very Large. Availability of ADS can enable early introduction of Thorium on a large scale
Development for MOX fuel for TAPS:

Phase I: [Early 80’s]
Fabrication of MOX fuel experimental clusters for irradiation in Cirus

Phase II: [Mid 90’s]
Industrial scale MOX fabrication plant set-up at Tarapur.
Pu bearing Fuels for Thermal Reactors

MOX Fuel for BWRs

- Several lead MOX Fuel Assemblies irradiated in TAPS-1 and TAPS-2 (BWRs) successfully to the design burn-up.
- MOX fabrication plant at Tarapur uses ATTRITOR technology for highly homogeneous fuel.
MOX Fuels for PHWRs

Development of high burn-up fuels for use in PHWRs

- 50 MOX fuel bundles fabricated at AFFF, BARC, Tarapur.
- All the MOX bundles are irradiated in KAPS-1 and peak burn up achieved is 20,000Mwd/t.
- Demonstration of capability to manufacture & utilize MOX fuels for PHWRS
Fast Breeder Test Reactor at Kalpakkam ($40\text{MW}_{\text{th}}$) is the test bed for development of fuel, blanket and structural materials for FBR programme.

FBTR achieved criticality on October 18, 1985 with unique Pu rich mixed carbide fuel developed by BARC.

The peak burn-up achieved on MK I is 155 GWD/T.
FBTR at Kalpakkam is the cradle for development of LMFR in India.

An indigenous Pu rich (U-Pu)MOX fuel was first considered and flow-sheet developed.

Difficulties with respect to

(a) Poor thermal conductivity
(b) Higher oxygen potential

raising concern on fuel-coolant and fuel clad compatibility issues.

Pu rich carbide appeared attractive option.
Fabrication of Mixed Carbide Fuel for FBTR at Kalpakkam

a) \((U_{0.3}Pu_{0.7})\) Mixed Carbide (MK I) and \((U_{0.45}Pu_{0.55})\) Mixed Carbide fuel (MK II) was chosen as fuel for FBTR at Kalpakkam.

b) High thermal conductivity and metal density in MC lead to compact core & high breeding ratio.

c) Performance of this fuel has been proved to be excellent and is at present beyond 150,000 MWD/T.

d) Technology is highly complex but can be adopted for small cores.
Issues Carbide fuel

High ‘Pu’ carbide

- Lower solidus temperature,
- Lower thermal conductivity,
- ‘O’ and ‘N’ pick up from cover gas
- ‘Pu’ loss during fabrication.

Carbide fuels in general

- Clad Carburization
- Swelling
- Susceptible to Oxidation and Hydrolysis by Moisture and Oxygen
- Pyrophorocitv
Challenges?

Unique composition; highly enriched ‘Pu’ carbide

Lack of Out-of-pile and In-pile Data in literature
Evaluation of Fuel properties

Thermal Expansion
Thermal Conductivity
Hot Hardness
Solidus Temperature
Out-of-pile fuel-clad-coolant compatibility
Post Irradiation Examination (PIE)
Thermal expansion as a function of temperature for sintered hyperstoichiometric $(Pu_{0.55}U_{0.45})C$
Thermophysical Properties

Thermal Conductivity

Temperature, K

Log Hardness (MPa)

Temperature (K)

Thermal Conductivity, W m$^{-1}$ K$^{-1}$

Hot hardness

(\(U_{0.3}Pu_{0.7})C - MK I\)

(\(U_{0.31}Pu_{0.69}C_{0.93}\) Ref (9))

(\(U_{0.79}Pu_{0.21}\)C)

(\(U_{0.45}Pu_{0.55}\)C - MK II)

\(\text{BARC}\)
Effect of ‘Pu’ content on ‘C’ potential, ‘CO’ pressure, ‘Pu’ pressure and ‘N’ pressure
The average hardness of reference samples: 179 VHN (at 973K for 1000hrs) and 152 VHN (1123K for 1000hrs). The hardness data of as received material was 230VHN.
Process Flow Sheet for Fabrication of (U,Pu)C Pellets

1. Carbothermic synthesis
   - \( \text{M} = (\text{PuO}_{2.93}\text{UO}_{2.07}) \)
   - MK II

2. Milling & Grinding
3. Tableting

4. Carbothermic Reduction
   - At 1673K in vacuum

5. Crushing

6. Compaction

7. Milling

8. Sintering
   - At 1923K in Ar-8%H₂

9. Pellet Inspection

Milling & Grinding, Tableting, Carbothermic Reduction, Crushing, Compaction, Milling, Sintering, Pellet Inspection
Reject Recycling

Dry Method

a) Mechanical Pulverization and re-consolidation

b) Thermal Pulverization to oxide phase and re-carbothermic reduction
Mixed Carbide Fabrication line

Homi Bhabha Centenary Year

Dec.1-2, 2008

IAEA TM, Fukui
<table>
<thead>
<tr>
<th>Burn-Up (GWd/t)</th>
<th>Fuel Swelling</th>
<th>Fission Gas Release</th>
<th>Clad Diameter Increase</th>
<th>Diametral Strain</th>
</tr>
</thead>
<tbody>
<tr>
<td>25</td>
<td>1.2% per at%</td>
<td>Negligible</td>
<td>No increase</td>
<td>5%</td>
</tr>
<tr>
<td>50</td>
<td>1.5%</td>
<td>5 to 6%</td>
<td>by 1.2 – 1.6%</td>
<td></td>
</tr>
<tr>
<td>110</td>
<td>&lt;1% per atom</td>
<td>14%</td>
<td>No fuel-clad gap, No</td>
<td></td>
</tr>
<tr>
<td>155</td>
<td>&lt;1% per atom</td>
<td></td>
<td>visible clad carburization</td>
<td></td>
</tr>
</tbody>
</table>

LHR raised to 400 W/cm after 35 GWd/t
(U-44\% Pu) MOX fuel
For Hybrid core of FBTR
Issues of (U- 44% PU)MOX fuel

Phase Stability

Thermo-physical Properties

Thermal conductivity / diffusivity, thermal expansion, high temperature hardness

Fuel- Coolant Chemical compatibility
Thermal conductivity of \((U_{0.55}Pu_{0.45})O_2\) pellet

XRD Pattern of \((UO_2-45\% PuO_2)\) fuel

X-ray radiograph of MOX \((UO_2-45\% PuO_2)\) fuel – Sodium compatibility capsules

Photomicrograph of MOX pellets after compatibility tests at 700°C for 400 h
X-ray radiograph of fuel-coolant compatibility capsules loaded with MOX (PFBR type) pellets (700°C after 400 hrs.)
MICROSTRUCTURES

800°C, 260hrs. x200

700°C, 350hrs.

breach

700°C, 350hrs

D9 clad

MOX

700°C, 450hrs

Na

Clad breach

x200

MOX
1. FBTR now operates with expanded hybrid core of mixed carbide and high Pu MOX.

2. 20% of the core is High Pu (45%Pu) MOX.

3. Experimental PFBR MOX fuel assembly at the centre of FBTR.
37 pins of UO$_2$-29%PuO$_2$ fabricated at A3F(T), BARC are under irradiation in FBTR.

U$^{233}$ is added to simulate high linear rating of PFBR with same (U-Pu) chemical composition.

FA loaded in the centre of FBTR core.

Burn-up reached - 80 GWd/t at 450 W/cm & continuing.
MOX Fuel for PFBR: Why MOX?

1) Industrial scale operational experience in fuel cycle facilities
2) Safe fuel cycle operations and proven safety response of the MOX fuel in the reactors
3) Economic competitiveness
4) High burn-up potential
5) Closed fuel cycle with minimum in and out of core inventory of Pu
### PFBR FUEL SUBASSEMBLY

#### Salient Details

<table>
<thead>
<tr>
<th>Property</th>
<th>Value</th>
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</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>(Pu-U)O₂</td>
</tr>
<tr>
<td>Pellet OD/ID</td>
<td>5.55/1.8 mm</td>
</tr>
<tr>
<td>Pin OD/ID</td>
<td>6.6/5.7 mm</td>
</tr>
<tr>
<td>Peak Linear Power</td>
<td>450 W/cm</td>
</tr>
<tr>
<td>Active core height</td>
<td>1000 mm</td>
</tr>
<tr>
<td>Breeding Ratio</td>
<td>1.05</td>
</tr>
<tr>
<td>Clad &amp; Wrapper</td>
<td>20% CW D9</td>
</tr>
<tr>
<td>No. of Pins</td>
<td>217</td>
</tr>
<tr>
<td>Width Across Flats</td>
<td>131.3 mm</td>
</tr>
<tr>
<td>Peak target Burnup</td>
<td>100 GWd/t</td>
</tr>
<tr>
<td>Peak neutron dose</td>
<td>85 dpa</td>
</tr>
<tr>
<td>No of Fuel SA</td>
<td>181</td>
</tr>
<tr>
<td>Total SA</td>
<td>1758</td>
</tr>
</tbody>
</table>

**Diagram:**
- **SECTION-MM**
- **SECTION-EE**
- **SECTION-BB**
- **SECTION-HH**
Safety vessel successfully erected on June 24th, 2008 – A major milestone
CRO

Dep. UO₂ Powder

PuO₂ Powder

Weighing

Mixing & Milling

Pre-compaction & Granulation

Final Compaction

Sintering

MOX Pellets Degassing

Stack Making and Loading

Top End Plug Welding (He)

Decontamination

Wire Wrapping & Spot Welding

Fuel Pin Inspection & Storage

SS Tubes & Hardwares

Bottom End Plug Welding (Ar)

Loading in Tubes

Oversize pellets

C. Grinding

Degassed depleted UO₂ pellets

PuO₂ Enrich.

Visual
He Leak
Metrology
X-Radiography

U, Pu - Analysis, Dissol Test, Visual, Dimensional, Linear Mass, Metallography, α - Autorad.

O/M, Total Gas, Metallic & Non-metallic Impurities

Stack Length Check

Decontamination check

Visual, He Leak, XGAR, Metrology, X-Radiography, γ - Scan, Cover Gas in Pins

PFBR Fuel Pin Fabrication Flowsheet with QC Steps

To CSP
Conveying of powder containers at AFFF
COMPACTION PRESS AND COMPACTED PELLETS
Inspection of End-plug Welding of D-9 Fuel Clad Tubes

Solidification cracking

TIG welding Machine inside a glove-box

Defect free weld zone

Metallographic Evaluation

REAL TIME MOTION RADIOGRAPHY

ULTRASONIC END CAP WELD INSPECTION
PFBR FUEL PINS
Metallic fuels for future FBRs

- Faster Growth Rate (BR 1.3 - 1.4)
- Enhanced Reactor Safety
- Integrated Fuel Cycle
1) Thermophysical, chemical, compatibility and thermodynamics studies of metallic fuels.

2) Development of technology for fabrication of fuel/blanket columns for sodium bonded and mechanical bonded fuel and demonstration with U/U-Zr alloy.
Metallic Fuels for FBRS

Ref metallic fuel for FBR - U-15Pu-10Zr (Sodium Bonded)

• zirconium increases the liquidus/solidus temperature of (U,Pu) alloy

• retards interdiffusion of fuel and clad constituents

Design for high burn-up metallic fuels

• reduction in smear density to 75% to accommodate swelling and to reduce FCMI

• provision of adequate gas plenum for fission gas release

• measures to reduce FCCI
1. Barrier clad to minimise fuel clad chemical interactions.
3. New U-Pu alloys ( Ternary or Binary ).
Maximum LHR for Power to Melt and Eutectic formation with T91 Cladding

- $T_{\text{melt}} = 1243 \, \text{K}$
- Clad limit = 923 K
- Centreline
- Clad, T91

**Linear Heat Rate (W/cm)**

**Temperature (K)**

- Maximum LHR for Power to Melt
- Formation with T91 Cladding

- Homi Bhabha Centenary Year
AS-PILGERED CLAD TUBE WITH LINER

OD = 6.645 – 6.650 mm

ID = 5.39 – 5.44 mm

LINER THICKNESS = 137 – 149 μ

NO GAP WAS FOUND BETWEEN SS TUBE & ZIRCALOY LINER
Why Mechanical Bond?

1. In case of metallic fuels better heat transfer is ensured by better contact of fuel meat with clad by Swaging operation.

2. Chemical reactivity of sodium during fabrication needs stringent safety measures.

3. Additional head end problems during reprocessing.


5. F.G. column only at the top and also F.G. mobility in sodium.
R&D on Fuel Fabrication for FBRs

Metallic Fuel for Fast Reactors

Preparation of U-Zr alloys by Arc Melting

Induction melting with addition of Pu & Injection Casting

Demoulding

Length cutting

ECT for defect detection

Length, diameter & weight measurement

Sodium/Mechanical bonding with clad

Swaging

ECT for Bond Inspection

Drawing

8mm diameter (Zr clad)
6 mm diameter

Injection Casting of Uranium
**INJECTION CASTING SET-UP**

- Mould Pallet up-down mechanism
- Top Clamp
- Top Chamber
- View Port
- Bottom Clamp
- Bottom Chamber

**Control Panel**
INJECTION-CAST & SWAGED URANIUM RODS

LENGTH = 160 mm, DIAMETER = 4.67±0.04
CO-SWAGED SS-ROD WITH CLAD/LINER
Thermophysical & compatibility studies of Metallic Fuels

Thermo physical Property Evaluation Laboratory has the following facilities:
1. Thermal conductivity
2. Specific heat
3. Coefficient of linear expansion
4. Chemical compatibility
Hardness profile along mechanically bonded U/Zr 4/D-9 composite
SEM of U-Zr-D9 Metallic fuel pin after out-of-pile chemical compatibility test showing no reaction layer at the interfaces.

- Thicknesses of ‘Zr’ barrier layer: 150μm and D9 cladding material: 450μm.

(a) As received  (b) 650°C, 500 hrs;  (c) 650°C, 1500hrs
Chemical Compatibility Studies

880°C, 15hrs.

onset of reaction
Out-of-pile Chemical compatibility studies [U/Zr/T91]

- **700°C**
  - 500 hrs:
    - Zr
    - U
  - 1000 hrs:
    - Zr
    - T91
  - 1500 hrs:
    - Zr
    - T91

- **650°C**
  - T91
  - Zr
  - U

Homi Bhabha Centenary Year
Thermal expansion values of U, Zr, D-9 and T-91
Thorium based MOX Fuels in Nuclear Power

a) Prominent Role in Future Nuclear Power Programme.
b) Non-Proliferation Aspects.
c) Spent Fuel – Almost a Stable Waste Form.
d) Lower Minor Actinide Generation.
e) Lower Reactivity Change with Burn-up.
f) Abundance of Resource.
Attributes of Thorium

- Three times more abundant than uranium

Better Performance Characteristics
- Higher melting point
- Better thermal conductivity
- Lower fission gas release
- Good radiation resistance and dimensional stability

Better Chemical Stability
- Reduced fuel deterioration in the event of failure
- No oxidation during permanent disposal in repository
- Poses problem in dissolution during reprocessing
PHWR Thoria Bundles

<table>
<thead>
<tr>
<th>Reactor</th>
<th>No. of bundles</th>
</tr>
</thead>
<tbody>
<tr>
<td>Madras- I</td>
<td>4</td>
</tr>
<tr>
<td>Kakrapar-I</td>
<td>35</td>
</tr>
<tr>
<td>Kakrapar-II</td>
<td>35</td>
</tr>
<tr>
<td>Rajasthan- II</td>
<td>18</td>
</tr>
<tr>
<td>Rajasthan -III</td>
<td>35</td>
</tr>
<tr>
<td>Kaiga-II</td>
<td>35</td>
</tr>
<tr>
<td>Rajasthan-IV</td>
<td>35</td>
</tr>
<tr>
<td>Kaiga-I</td>
<td>35</td>
</tr>
</tbody>
</table>
### Irradiation Experience Of (Th-Pu) MOX in PWL, CIRUS

<table>
<thead>
<tr>
<th>Cluster</th>
<th>Fuel</th>
<th>Clad type</th>
<th>GWd/t</th>
<th>kW/m</th>
</tr>
</thead>
<tbody>
<tr>
<td>AC-6</td>
<td>(Th-4%Pu)O₂</td>
<td>Free standing (TAPS-BWR)</td>
<td>18.5</td>
<td>40</td>
</tr>
<tr>
<td>BC-8</td>
<td>(Th-6.75%Pu)O₂ ThO₂</td>
<td>Collapsible (PHWR)</td>
<td>10.2</td>
<td>42</td>
</tr>
</tbody>
</table>
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.

**Major Design Objectives**

- Power output – 300 MWe with 500 m$^3$/d of desalinated water.
- A large fraction (65%) of power from thorium.
- Extensive deployment of passive safety features – 3 days grace period, and no need for planning off-site emergency measures.
- Design life of 100 years.
- Easily replaceable coolant channels.

**Salient Features of Pressure tube**

- 120mm ID x 6300 mm length
- Replaceable through top end-fitting
- Unique shape by Pilgering route.
- Thicker at one end, tapering at the other
- Controlled cold work to achieve required tensile properties.
Thorium Utilization—Initiated by Pu

Advanced Heavy Water Reactor: 300 MWe

Equilibrium Core:
Fuel cluster containing 54 pins arranged in 3 rings of 12, 18 and 24 pins
- $\text{ThO}_2$-3.25%$\text{PuO}_2$ (outer 24 pins)
- $\text{ThO}_2$-3.75%$\text{U}^{233}\text{O}_2$ (18 pins)
- $\text{ThO}_2$-3.00%$\text{U}^{233}\text{O}_2$ (12 pins)

Initial core will have fuel assemblies containing all $(\text{Th-Pu})\text{MOX}$ pins.
Proliferation Resistance Characteristics of AHWR

1. High energy gamma from U$^{233}$ daughters makes U$^{232}$ handling difficult.
2. Reprocessing & re-fabrication needs high technology.
3. Denaturing of U$^{233}$ is possible with U$^{238}$.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Feed isotopic %</th>
<th>Discharge Isotopic %</th>
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</thead>
<tbody>
<tr>
<td>Pu239</td>
<td>68.7</td>
<td>6.5</td>
</tr>
<tr>
<td>Pu240</td>
<td>24.6</td>
<td>50.29</td>
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<tr>
<td>Pu241</td>
<td>5.26</td>
<td>21.50</td>
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<tr>
<td>Pu242</td>
<td>1.30</td>
<td>21.70</td>
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<tr>
<td>U232</td>
<td>-</td>
<td>&gt;0.1</td>
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<tr>
<td>U233</td>
<td>-</td>
<td>92.46</td>
</tr>
<tr>
<td>U234</td>
<td>-</td>
<td>6.93</td>
</tr>
<tr>
<td>U235</td>
<td>-</td>
<td>0.606</td>
</tr>
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</table>

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Feed isotopic %</th>
<th>Discharge Isotopic %</th>
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</thead>
<tbody>
<tr>
<td>U232</td>
<td>-</td>
<td>&gt;1000 ppm</td>
</tr>
<tr>
<td>U233</td>
<td>92.46</td>
<td>85</td>
</tr>
<tr>
<td>U234</td>
<td>6.93</td>
<td>12.87</td>
</tr>
<tr>
<td>U235</td>
<td>0.60</td>
<td>1.97</td>
</tr>
<tr>
<td>U236</td>
<td>0.04</td>
<td>0.14</td>
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</table>
AHWR Critical Facility

Fuel for AHWR Critical Facility
Built at Trombay to conduct lattice physics experiments for validation of AHWR

- Validate reactor physics design on various Fuel types.
- Reference core consists of 19 element U metallic fuel clusters - 65 Nos.
- Thoria, (Th-Pu) & (Th-U233)MOX fuel clusters to be introduced in phases for Physics study.
- Criticality achieved in April 2008.
Difficulties in making (Th-U)O₂ Fuel

Fabrication of (Th,²³³U)O₂ fuel is difficult because it usually contains daughters of ²³²U (half-life 73.6 years) namely, ²¹²Bi and ²⁰⁸Tl, which emit strong gamma radiations.

Therefore the fabrication of the thorium fuel requires operations in the shielded glove-boxes to protect the operators from radiation.

\[ \begin{align*}
\text{²³²U (68 years)} & \rightarrow \text{²²⁸Th (1.9116 years)} \\
\text{α - decay} & \\
\text{²²⁴Ra (3.66 days)} & \rightarrow \text{²²⁰Rn (55.6 seconds)} \\
\text{α - decay} & \\
\text{²¹⁶Po (0.145 second)} & \rightarrow \text{²¹²Pb (10.64 hours)} \\
\text{α - decay} & \\
\text{²¹²Bi (60.55 minutes)} & \rightarrow \text{²⁰⁸Tl (3.053 minutes)} \\
\text{α - decay} & \\
\text{²⁰⁸Pb (stable)} & \rightarrow \text{²¹²Pb (0.299 μ seconds)} \\
\text{β⁻ - decay} & \\
\text{³⁵.⁹⁴ %} & \rightarrow \text{³⁶.⁰⁶ %} \\
\text{α - decay} & \\
\text{β⁻ - decay} &
\end{align*} \]
### Fabrication Technology for (Th-U\(^{233}\)) MOX

<table>
<thead>
<tr>
<th>Sr. No</th>
<th>Technique</th>
<th>Flow sheet and type of facility</th>
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</thead>
<tbody>
<tr>
<td>1</td>
<td>Powder-Pellet Route</td>
<td>Fully Shielded Facility</td>
</tr>
<tr>
<td>2</td>
<td>Sol-gel microsphere pelletization</td>
<td>Fully shielded facility</td>
</tr>
<tr>
<td>3</td>
<td>Impregnation (vacuum/microwave of gel/pellet)</td>
<td>Partially shielded facility</td>
</tr>
<tr>
<td>4</td>
<td>CAP process (Coated agglomerate pelletization)</td>
<td>Partially shielded facility</td>
</tr>
</tbody>
</table>
Impregnation Set-up Inside the Glove Box

Microstructure of ThO$_2$-4%UO$_2$
Fuel made by Impregnation
Flow-sheet for fabrication of ($\text{Th,U})\text{O}_2$ pellets

- Co- Milling (Planetary ball mill)
- Precompaction (140 MPa)
- Granulation
- Final Compaction (350 MPa)
- Sintering At 1923K in Ar-8%$\text{H}_2$
- Centreless grinding
- Encapsulation (in zircaloy cladding tube)

ThO$_2$ granules

- Coating with UO$_2$ powder
- Final Compaction (350 MPa)
- Sintering At 1673K in Air
- Centreless grinding
- Encapsulation (in zircaloy cladding tube)

IAEA TM, Fukui
Flow sheet for (Th-U\textsuperscript{233})O\textsubscript{2} pellet using CAP Process

- Fresh ThO\textsubscript{2} Granules (precompaction)
- Fresh ThO\textsubscript{2} Agglomerates (Extrusion)
- Fresh ThO\textsubscript{2} Microspheres (So-gel Technique)

Coating/blending with U\textsuperscript{233} oxide/
Co-ppt/Master mix

Compaction in rotary press

Sintering in air

Encapsulation

Co-ppted ThO\textsubscript{2}-
50% U\textsubscript{3}O\textsubscript{8} or
pure U\textsuperscript{233} oxide

Unshielded Facility

Shielded Facility
The interface between two colonies of fine grains indicating that diffusion is complete between two agglomerates
Uranium distribution

Elemental scan for U M\(\alpha\), Th M\(\alpha\), and O K\(\alpha\) for ThO\(_2\)-3.75% UO\(_2\) pellet

<table>
<thead>
<tr>
<th>Element</th>
<th>Fine grains Wt %</th>
<th>Coarse grains Wt %</th>
</tr>
</thead>
<tbody>
<tr>
<td>Th</td>
<td>84.770</td>
<td>84.236</td>
</tr>
<tr>
<td>U</td>
<td>3.120</td>
<td>3.656</td>
</tr>
<tr>
<td>O</td>
<td>12.110</td>
<td>12.108</td>
</tr>
</tbody>
</table>

Atom distribution in coarse grains and fine grains for ThO\(_2\)-4%UO\(_2\) pellet determined by EPMA
The strength of the pellet at room temperature is related to grain size by the Hall-Petch relation. Accordingly, the smaller grain sized pellets will have higher strength. But at high temperature, the grain boundaries become weaker than grain matrix. Since the pellets of smaller grain size have larger grain boundary areas, these pellets become softer than pellets with larger grain size. Also as the grain size decreases, the creep rate of the fuel increases. Therefore, pellets with smaller grain size have higher creep rate and better plasticity at high temperatures. These pellets will reduce the PCMI.

The basic requirements for the high performance of a fuel are:

a) "Soft pellets" – To reduce Pellet clad mechanical interaction (PCMI).

b) Large grain size – To reduce fission gas release (FGR)
Thermal Conductivity Data of (Th,U)O$_2$
Generated at RMD

Thermal Conductivity (W/mK)

- ThO$_2$ RMD
- ThO$_2$ +2% UO$_2$ RMD
- ThO$_2$ +4% UO$_2$ RMD
- UO$_2$ RMD

Temperature (K)
Recycle Fuels in INDIA

Pu Bearing MOX Fuel Fabrication for Thermal & Fast Reactors

**Scope of work**

**First stage**
- (U~4%Pu) MOX for BWRs (TAPS)
- (U~0.4%Pu) MOX for PHWRs (KAPS)

**Second stage**
- (U-45%Pu) MOX for FBTR
- (U-29%Pu) MOX for PFBR
- (U-70%Pu) MC for FBTR
- (U-Pu-Zr) Metal for FBR

**Third stage**
- (Th-3%Pu) MOX - AHWR
- (Th-3%U233)MOX -AHWR

Dec.1-2, 2008
IAEA TM, Fukui
Development of Nuclear Energy based on a closed fuel cycle enabling fuller use of Uranium & Thorium is the only way to meet the development aspirations of over a billion people while meeting the low emission norms.
Thank You