ICTP Workshop
Modelling and Quality Control for
Advanced & Innovative Fuel Technologies

Trieste, 14 November, 2005

Opening Address

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**IAEA**

**NUCLEAR FISSION**

- **Fission Chain Reaction**

  Neutrons → **U - 235** → Fission → **Light Fission Product** (mass number 95) → + 200 MeV Energy + Neutrons → **U - 235**

- **Production of Fissile Isotopes Pu**$^{239}$ & **U**$^{233}$

  Neutrons → **U**$^{238}$ → **U**$^{239}$ → **Th**$^{232}$ → **Th**$^{233}$ → **Pa**$^{233}$ → **Pu**$^{239}$ → β$^{-}$ → β$^{-}$ → β$^{-}$ → β$^{-}$ → **U**$^{233}$
FISSILE & FERTILE ISOTOPES

URANIUM
- 4 ppm in earth’s crust, Klaproth, 1789
- Natural Uranium:
  - $^{238}\text{U}$ (fertile): 99.3%; 4.5x10^9 y
  - $^{235}\text{U}$ (fertile): 0.7%; 8.0x10^8 y

\[
^{235}\text{U} \xrightarrow[n^1]{\text{fissile}} ^{236}\text{U} + ^{n}\text{M} M_1(85) + M_2(139) + (2-3)n + 200 \text{ MeV}
\]

\[
^{238}\text{U} \xrightarrow[n^1]{\text{fertile}} ^{239}\text{Th} \xrightarrow[23.5\text{ min}]{\beta^-} ^{239}\text{Np} \xrightarrow[2.35\text{ days}]{\beta^-} ^{240}\text{Pu (fissile)}
\]

THORIUM
- 12 ppm in earth’s crust, Berzelius, 1828
- Natural Thorium: No fissile isotope,
  - $^{232}\text{Th}$ (fertile): 1.4x10^10 y

\[
^{232}\text{Th} \xrightarrow[n^1]{\text{fertile}} ^{233}\text{Pa} \xrightarrow[23.5\text{ min}]{\beta^-} ^{233}\text{U (fissile)}
\]

PLUTONIUM
- Does not occur in nature; man-made, Seaborg, Wahl and Kennedy, 1941
  - $^{239}\text{Pu}$: $\alpha$; 2.4 x 10^4 y
  - $^{240}\text{Pu}$: $\alpha, \beta^-$; 6.58 x 10^3 y
  - $^{241}\text{Pu}$: $\beta^-$; 13 y, Am241 strong $\gamma$ emitter
  - $^{242}\text{Pu}$: $\alpha, \beta^-$; 3.79 x 10^5 y
  - $^{238}\text{Pu}$: $\alpha, \beta^-$; 86.4 y

- Maximum limits
  - body burden: 0.18 – 0.65 $\mu$g
  - concentration in air: 2 x 10^{-12} $\mu$ curie per cc
  - concentration in water: 10^{-4} $\mu$ curie per cc
NUCLEAR REACTOR

REACTOR CORE MATERIALS
- Fuel (Plate or Pin)
  - Fissile (U²³⁵, Pu²³⁹ or U²³³) & Fertile (U²³⁸ or Th²³²) as metals, alloys, composites, oxide, carbide or nitride.
- Fuel Cladding
  - Al or Al alloys for research reactors.
  - Zircaloy for LWR & PHWR.
  - SS 316, D-9, HT-9 for LMFBR.
- Coolant
  - H₂O/D₂O: Water cooled reactor.
  - CO₂/He: Gas cooled reactor.
  - Na: LMFBR
- Moderator (Only for Thermal Reactor)
  - H₂O/D₂O or graphite
- Control Rod
  - B, Cd, Hf, Gd.

TYPE OF REACTORS
- Research Reactors
  - Not for generating electricity but as source of neutron for radioisotope production, materials testing, neutron radiography & diffraction and basic studies.
- Power Reactors
  - Used as a heat source for generation of power/electricity.

Diagram:
- Source of heat energy
- Source of neutrons
- Control Rod
- Secondary Coolant
- Generator
- Primary Coolant
- PUMP
- Heat Exchanger
- Fissile
  - U²³⁵ + n → 0²⁹² U²³⁶
  - Fission
  - Fragment A: Sr, Zr, W, Mo, etc. (95)
  - Fission Fragment B: Xe, Cs, Ba, La, etc. (139)
  - n = 200 MeV
- Fertile
  - U²³⁸ + n → 0²⁹² U²³⁹
  - Fission
  - Pu²³⁹
  - 94
- Natural Uranium
  - 99.3% U²³⁵ fertile
  - 0.7% U²³³ fertile
- Natural Thorium
  - Th²³² fertile
  - Th²³² + n → 0²⁹² Th²³³
  - 233 days
- Fertile
  - U²³³ fertile

Nucuyên Reacțor

MATERIALELE DE CORECTOREA REACTORULUI
- Combustibil (Platou sau Roșu)
  - Fissil (U²³⁵, Pu²³⁹ sau U²³³) și Fertil (U²³⁸ sau Th²³²) ca metale, aliaje, compoziții, oxiu, carbida sau nitrid.
- Cladare de combustibil
  - Al sau aliaje de Al pentru reacțori de cercetare.
  - Zircaloy pentru LWR & PHWR.
  - SS 316, D-9, HT-9 pentru LMFBR.
- Încalzitor
  - H₂O/D₂O: Reacțori cu încalzire cu apă.
  - CO₂/He: Reacțori cu încalzire cu gaz.
  - Na: LMFBR
- Moderator (Numai pentru Reactor Termic)
  - H₂O/D₂O sau grafit.
- Pompa de control
- Cuvânt pe care nu apare: Primary Coolant

TIPI DE REACTOARE
- Reactoare de cercetare
  - Nu pentru producerea de electricitate, ci ca sursă de neutron pentru producerea de izotopi radioactivi, testarea materialelor, radiografia și difractometrie și studii de bază.
- Reactoare de putere
  - Utilizate ca sursă de caldura pentru generația de putere/elecție. 
NUCLEAR REACTORS & THEIR APPLICATIONS

Source of intense heat energy & Source of neutrons

- Power Reactors:
  - Generation of Electricity
  - Production of Hydrogen
  - Desalination of Sea Water
  - Marine Propulsion
  - District Heating

- Non-Power Reactors:
  - Production of Radioisotopes
    - Nuclear Medicine, Radiopharmaceuticals
    - Radioimmunotherapy - $^{99m}$Tc, I $^{131}$, Sm $^{153}$, P $^{32}$, etc.
    - Cancer Diagnosis & Therapy
    - Sterilisation of Medical Kits, Hospital Wastes & Sewage
    - Food Irradiation & Preservation
    - $\gamma$-radiography - Co $^{60}$, Ir $^{192}$ & Cs $^{137}$
  - Production of Fissile Isotopes (Pu $^{239}$ and U $^{235}$)
  - Neutron Radiography, Neutron Diffraction & Neutron Activation Analysis
  - Irradiation - Testing of Materials
  - Training, Education & Basic Research
Nuclear Energy for Tomorrow –
to Combat CO₂ emission & Global Warming

ENERGY PRESENTLY UTILISED:

- Of today's 6 billion people in the world, one third has no access and another one third has limited access to electricity.
- 66% global electricity is generated from fossil fuel, including 40% from coal, causing large release of CO₂ greenhouse gas to the environment and in turn global warming.
- Fossil-fuel would be significantly depleted by the end of this Century.

TRANSPORTATION: 75%

- Cars, trucks, trains and planes use 'fossil fuels': gasoline, diesel, jet fuel, etc. releasing CO₂ to the environment.
- Only 12% of the world's population has access to automobiles - when the other 88% decides to drive, the present 'fossil-fuelled' vehicle, imagine the CO₂ emission to the environment.

Fossil Fuels release 70 million tons CO₂ every day or 800 tons/second to the environment causing Global Warming.

In the next 50 years - as world population expands to 9 billions - global energy consumption will double. How do we meet the ever-increasing demand minimizing CO₂ emission that cause global warming?

NUCLEAR ENERGY IS THE INEVITABLE OPTION!

- NUCLEAR FISSION HEAT ENERGY should be used for generation of electricity, desalination of sea water, district heating in cold countries & for production of hydrogen economically by electrolysis of water or by cracking of hydrocarbon.
- HYDROGEN, instead of GASOLINE, should be the energy source for 'land transportation' in tomorrow's mega cities for minimising CO₂ greenhouse gas emissions & other automobile pollution.
By the end of 1960s and early 1970s, PWRs, BWRs and RBMK were commercialized when several 1000 MWe units were commissioned in USA and Russia. The VVER-440 reactors were also commercialized during this period.

The Nuclear power programme reached the commercial phase all over the world in the 1970s and 1980s when a large number of Light Water Reactors (LWR), including PWRs, BWRs, VVERs and RBMK type reactors, CANDU-PHWRs and Gas-Cooled Reactors (both Magnox and Advanced Gas-cooled Reactors) were commissioned in USA, Canada, USSR, Europe, Japan, South Korea and India. Several prototype and commercial LMFBRs were also commissioned in USSR, France and UK.

As on March 2005, there are some 440 commercial nuclear reactors operating in 31 countries with over 360 GWe total capacity. They supply 16% of world’s electricity.
NUCLEAR POWER REACTORS
(July 2005)

441 Reactors in 30 countries
3,674,966 MWe - 16% global electricity

26 Reactors under construction - 20,826 MWe
Pressure Vessel and Pressure Tube Type of Nuclear Power Reactors

Pressure Vessel Type
(eg. Boiling Water Reactors at Tarapur)

Pressure Tube Type
[Pressurised Heavy Water Reactor (PHWR), eg. PHWR 220 at Rawatbhatta, Narora, Kaiga, etc.]
TYPICAL CALANDRIA VESSEL
• **Light Water Reactor (LWR):** 87%
  - Pressurised Water Reactor (PWR). Russian type known as VVER
  - Boiling Water Reactor (BWR)
    USA, France, Germany, Sweden, Belgium, Russia & CIS (WER),
    Japan, South Korea, Brazil & China
• **Pressurised Heavy Water Reactor (PHWR):** 6%
  (also known as CANDU) Canada, India, South Korea, Romania, Argentina, Pakistan & China
• **Light Water Graphite moderated Reactor (LWGR):** 3%
  Russia and CIS - known as RBMK type reactor.
• **Gas Cooled Graphite Moderated Reactor (GCR):** 3%
  Popular only in UK.
• **Liquid Metal Fast Reactor (LMFR):** ~1%
  Presently, only one prototype and commercial LMFR is in operation: Phenix 250 MWe (France) & BN-600 MWe (Russia). In India, a Fast Breeder Test Reactor (FBTR) is in operation. Prototype Fast Breeder Reactor is under construction in Japan (Monju) and India (PFBR)
On-going International Programmes:

- **IAEA-initiated** INNOVATIVE NUCLEAR REACTORS & FUEL CYCLE PROGRAMME (INPRO)

- **US-initiated** Generation IV International Forum (GIF)

In both INPRO and GIF, nuclear power reactors have been perceived to be utilised for the following purpose:

- Generation of Electricity.
- Production of Hydrogen Fuel
- Desalination of Sea Water
- District Heating
UNIQUE FEATURES OF NUCLEAR ENERGY & POWER

• Relatively New

"Nuclear Fission Energy" is very recent (< 60 years) in the time scale of human civilisation.
First Nuclear Power Reactor in the world : 1955

• High Energy Density :

1 atom of ‘C’ on combustion releases ~ 4 eV
1 atom of U^{235} on fission release ~200 MeV

Annual Fuel Requirement of a 1000 MWe Power Station:
Nuclear : 30 tons; Coal : 2.6 million tons; Oil : 2.0 million tons

• Environment Friendly:

Zero emission of CO₂, SO₂ & NOₓ – no global warming and acid rain
Annual Discharge from 1000 MWe Power Station:
Nuclear : 3.5 tons spent fuel; Coal Fired : 6.5 million tons CO₂

CO₂ emission per kWh: 0.967 kg Denmark (82% Coal, 0% Nuclear)
0.63 kg UK (49.5% Coal, 28% Nuclear)
0.064 kg France (77.36% Nuclear)

• Generates man-made ‘fissile’ isotopes or fuels:

Nuclear fuel is made up of ‘fissile’ (U^{235}, Pu^{239} & U^{233}) and ‘fertile’ (U^{238} & Th^{232}) isotopes. U^{235} is
the only ‘fissile’ isotope occurring in nature. The ‘fission’ process splits up ‘fissile’ nuclei,
releases very high heat energy and generates extra neutrons which could convert naturally
occurring ‘fertile’ isotopes, U^{238} & Th^{232} to man-made ‘fissile’ isotopes Pu^{239} & U^{233} respectively.

From radiation safety point of view, natural uranium (U^{235} & U^{238}) and thorium (Th^{232})
are mildly radioactive and have very little hazard from external radiation. However,
Pu^{239} (always associated with Pu^{240}, Pu^{241}, Pu^{242} & Pu^{243}), U^{233} (always associated with
U^{235}) and fission products are highly radioactive and health hazardous & require
proper containment, beta, gamma neutron shieldings and remote handling. In order
to ensure safety from any ‘criticality accident’ (uncontrolled nuclear fission chain
reaction), only limited and controlled quantity of ‘fissile ’ (U^{235}, Pu^{239} or U^{233}) materials
are permitted to handle at a time. The radioactive waste has to be properly treated,
fixed and stored or disposed. From security point of view, physical protection of
‘fissile’ material is essential to avoid proliferation risk for non-peaceful purpose.
Comparative Hazards associated with external dose from radioactive material

**Alpha Radiation:** (practically no hazard from external radiation)
Can be stopped completely by a sheet of paper – may just penetrate the surface of the skin.

**Beta Radiation:** (minimum hazard from external radiation)
Can be stopped by a sheet of aluminium a few mm in thickness.

**Gamma:** (hazardous – requires proper shielding) & X Rays
Are very penetrating and can pass right through human body – mostly absorbed by heavy elements like lead, which is normally used as shielding material for gamma x-rays.

**Neutrons:** (hazardous – requires proper shielding)
Are very penetrating – in general, efficient shielding against neutron can be provided by water, perspex, etc. Concrete shielding (sometimes up to 1 m thickness) is used for neutron and gamma shielding.

Remote Operations using manipulators for handling highly radioactive materials (mainly high gamma and high neutron dose) inside concrete hot cells.
## CRITICAL MASS OF URANIUM & PLUTONIUM

<table>
<thead>
<tr>
<th>Fissile Material</th>
<th>Approximate Critical Mass (kg)</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>Bare</td>
</tr>
<tr>
<td><strong>URANIUM</strong></td>
<td></td>
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<tr>
<td>Density 18.8 g/cm³</td>
<td></td>
</tr>
<tr>
<td>Natural ‘U’</td>
<td></td>
</tr>
<tr>
<td>U²³⁵</td>
<td>47</td>
</tr>
<tr>
<td>U²³³</td>
<td>17</td>
</tr>
<tr>
<td>90% enriched U²³⁵</td>
<td>53</td>
</tr>
<tr>
<td>20% enriched U²³⁵</td>
<td>750</td>
</tr>
<tr>
<td><strong>PLUTONIUM</strong></td>
<td></td>
</tr>
<tr>
<td>Density 19.85 g/cm³</td>
<td></td>
</tr>
<tr>
<td>Pu²³⁹</td>
<td>10</td>
</tr>
<tr>
<td>PuO₂</td>
<td>24.5</td>
</tr>
<tr>
<td>PuC</td>
<td>18</td>
</tr>
<tr>
<td>PuF₄</td>
<td>56</td>
</tr>
</tbody>
</table>

### Max. Permissible Conct. Of ‘Pu’ radionuclides

<table>
<thead>
<tr>
<th>Plutonium Radionuclides</th>
<th>Body (MPBB) (µg)</th>
<th>Air (µ Curie/cm³)</th>
<th>Water (µ Curie/cm³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu²³⁸</td>
<td>2.4 x 10⁻³</td>
<td>2 x 10⁻¹²</td>
<td>10⁻⁴</td>
</tr>
<tr>
<td>Pu²³⁹</td>
<td>0.65</td>
<td>2 x 10⁻¹²</td>
<td>10⁻⁴</td>
</tr>
<tr>
<td>Pu²⁴⁰</td>
<td>0.18</td>
<td>2 x 10⁻¹²</td>
<td>10⁻⁴</td>
</tr>
<tr>
<td>Pu²⁴¹</td>
<td>8.2 x 10⁻³</td>
<td>9 x 10⁻¹¹</td>
<td>7 x 10⁻³</td>
</tr>
<tr>
<td>Pu²⁴²</td>
<td>12.8</td>
<td>2 x 10⁻¹²</td>
<td>10⁻⁴</td>
</tr>
</tbody>
</table>
MAJOR NUCLEAR FUEL FORMS

A: Pellet-pin

Zircaloy clad, Natural UO₂ Fuel

VVER-1000 Zr-1%Nb clad Slightly Enriched Uranium (<5% U²³⁵)UO₂

Ceramic Fuel Pellets

Fast Breeder Reactor FUEL Stainless Steel clad, (U, Pu)O₂ (20-25% Pu O₂)

17x7 PWR

9x9 BWR

37 Fuel Elements CANDU/PHWR Zircaloy clad, Natural UO₂ Fuel

312 Fuel Elements

CANDU PHWR Fuel Elements

37 Fuel Elements

Stainless Steel clad, (U, Pu)O₂

(20-25% Pu O₂)

312 Fuel Elements

17x7 PWR

9x9 BWR

Zircaloy clad Slightly Enriched Uranium (<5% U²³⁵)UO₂

(Vibratory compacted fuel pins have been utilized on a very limited scale for experimental purpose)
**FUEL FORMS**

**B: Coated Fuel Particles**

1. Coated fuel particles for HTGR
   - **Prismatic block**
     - US, Japan, Russia and France

2. Fuel particles (dry or wet route) for vibratory compacted fuel pins

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**Pebble Bed**
- coated particle fuels embedded in spherical shape
- Germany, South Africa, China

- Pebble fuel element
- Pebble has diameter of 60 mm

- Triso coated particle

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**COATED RTICLES**
- Pyrolytic Carbon
- Silicon Carbide
- Porous Carbon Buffer
- UCO Kernel

- TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right)

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**COMPACTS**

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**FUEL ELEMENTS**
For nuclear energy to be sustainable as a global source of emission – free energy, the reactor fuel cycle must also remain sustainable (DG-IAEA Scientific Forum 2004)
Workshop Structure – 3 Modules & Topics

Module I: Nuclear Power Reactors and Fuels
- Reactor systems
- Fuel cycle options
- INPRO and GIF
- Conventional, advanced and innovative fuels

Module II: Fuel Design, Fabrication, QC, Modelling, Irradiation – Testing & PIE
- Fuel rod and assembly design
- Fabrication & QC of fuel pellets, rods and assembly
- Irradiation – testing in research and power reactors and
- results of post irradiation examination (PIE)
- Fuel modelling and different codes

Module III: Advanced Fuel Management
- Spent fuel management
- Use of plutonium and uranium-233 based fuels
- Proliferation-resistant fuels
- Minor Actinide Incineration
ICTP Workshop
Lecture 2
Nuclear Fuel Cycle Options

Chaitanyamoy Ganguly

Trieste, 15 November, 2005

Chaitanyamoy GANGULY

Head, Nuclear Fuel Cycle & Materials Section

IAEA International Atomic Energy Agency
For nuclear energy to be sustainable as a global source of emission-free energy, the reactor fuel cycle must also remain sustainable (DG-IAEA Scientific Forum 2004)
Develop a fuel cycle that is economically viable, environmentally benign, proliferation-resistant, safe and sustainable
Permanent Repositories for Safe Disposal of Nuclear Wastes

Monitored Geological Repository for High Level Radioactive Waste at Yucca Mountains, Nevada

Basis of Selection:

i) its remote location and long distance from a large population center—100 miles from Las Vegas, Nevada;

ii) its very dry climate—less than 6 inches of rainfall a year

iii) its extremely deep water table—800 to 1,000 feet below the level of the potential repository

Yucca Mountains, USA

Four Permanent Deep Repositories in Finland for Safe Disposal of Highly Radioactive Spent Fuels: 2 at Nuclear Power Plant sites, namely Olkiluoto at Eurojoki & Lovisa and other 2 at Kuhmo at Aanekoski.

Olkiluoto, FINLAND

The ONKALO facility at Olkiluoto is under construction at depths of 300, 400 and 500 meters with an access tunnel and an associated ventilation tunnel. The bed-rock at Olkiluoto is suitable for safe disposal. The radioactive waste would be packed in copper based canisters, which will be surrounded by compacted bentonite in the repositories. The Permanent Repository at Olkiluoto would be operational in 2010.
Figure 3. integrated Nuclear Fuel Cycle Information System (iNFCIS) maintained in NFC&MS.
International initiatives on innovative reactors and fuel cycle

- **INPRO**: International Projects on Innovative Nuclear Reactors and Fuel Cycle
- **GIF**: Generation IV International Forum and Advanced Fuel Cycle Initiative (AFCI)
CURRENT TRENDS IN URANIUM DEMAND & SUPPLY

Demand: 66,815 tons ‘U’

Production: 40,251 tons ‘U’ (60% of demand)

The shortfall of 40% is met from the following Secondary Supplies:

- Excess Commercial Inventories
- LEU from Ex-Military HEU
- Re-enrichment of DU-Tailings (enrichment plants) & REU (reprocessing plants)
- Ex-Military & Civil Pu (to be used in the form of Mixed Uranium Plutonium Oxide (MOX))

THERE IS NO SHORTAGE OF URANIUM RESOURCES FOR NEXT 50 YEARS EVEN FOR ONCE-THROUGH FUEL CYCLE
## Countries with Major Uranium Resources & No. of Nuclear Reactors & Countries with Major Nuclear Power Programme & their U-Sources

[World Uranium Resources (RAR<130US$/kg ‘U’): 3.17 million tons ‘U’]


<table>
<thead>
<tr>
<th>Country</th>
<th>Uranium Resources (Tons ‘U’)</th>
<th>Percentage of world resource</th>
<th>No. of Nuclear Power Reactors (% Electricity)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Australia</td>
<td>735,000</td>
<td>23%</td>
<td>Nil</td>
</tr>
<tr>
<td>Kazakhstan</td>
<td>530,460</td>
<td>17%</td>
<td>Nil</td>
</tr>
<tr>
<td>Namibia</td>
<td>170,532</td>
<td>5%</td>
<td>Nil</td>
</tr>
<tr>
<td>Niger</td>
<td>102,227</td>
<td>3%</td>
<td>Nil</td>
</tr>
<tr>
<td>Uzbekistan</td>
<td>79,620</td>
<td>2.5%</td>
<td>Nil</td>
</tr>
<tr>
<td>Mongolia</td>
<td>46,200</td>
<td>1.5%</td>
<td>Nil</td>
</tr>
<tr>
<td>USA</td>
<td>345,000</td>
<td>11%</td>
<td>104 (20%)</td>
</tr>
<tr>
<td>Canada</td>
<td>333,834</td>
<td>10.5%</td>
<td>20 (~12%)</td>
</tr>
<tr>
<td>South Africa</td>
<td>315,330</td>
<td>10%</td>
<td>2 (5.9%)</td>
</tr>
<tr>
<td>Russian Fed.</td>
<td>143,020</td>
<td>4.5%</td>
<td>30 (16%)</td>
</tr>
<tr>
<td>Brazil</td>
<td>86,190</td>
<td>3%</td>
<td>2 (4%)</td>
</tr>
<tr>
<td>France</td>
<td>100% from overseas sources</td>
<td></td>
<td>59 (78%)</td>
</tr>
<tr>
<td>Germany</td>
<td>100% from overseas sources</td>
<td></td>
<td>18 (30%)</td>
</tr>
<tr>
<td>Japan</td>
<td>100% from overseas sources</td>
<td></td>
<td>53 (39%)</td>
</tr>
<tr>
<td>Korea (R.O.)</td>
<td>100% from overseas sources</td>
<td></td>
<td>19 (39%)</td>
</tr>
<tr>
<td>China (excl. Taiwan)</td>
<td>35,060</td>
<td>1%</td>
<td>9 (1.4%)</td>
</tr>
<tr>
<td>India</td>
<td>40,980 (&gt;130$/kg)</td>
<td></td>
<td>14 (3%)</td>
</tr>
</tbody>
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Uranium Resources Worldwide

1. **Conventional Resources**

   - **Reasonably Assured Resources (RAR)**
   - **Estimated Additional Resources (EAR) category I & II**
   - **Speculative Resources (SR)**

   a) Conventional Resources (RAR and EAR-1):
      \[
      \leq \text{US$ 80 / kg U} : 3,537,000 \text{ tons}; \leq \text{US$ 130 / kg U: 4,589,000 tons U}
      \]

   b) Undiscovered Conventional Resources (EAR-II and SR): 9,794,000 tons U

   **Total Conventional Resource (a + b): 14,383,000 tons U**

2. **Total Unconventional Resource**

   - **Uranium in Phosphates** 22,000,000 tons U
   - **Uranium in Sea Water** 4,000,000,000 tons U

   (with improved adsorbent material, the latest estimated price of U from sea-water reported by Japanese is in the range of US$ 300/kgU – IAEA Red Book 2001).
World Thorium Resources - economically extractable

<table>
<thead>
<tr>
<th>Country</th>
<th>Reserves / t</th>
</tr>
</thead>
<tbody>
<tr>
<td>Australia</td>
<td>300 000</td>
</tr>
<tr>
<td>India</td>
<td>290 000</td>
</tr>
<tr>
<td>Norway</td>
<td>170 000</td>
</tr>
<tr>
<td>USA</td>
<td>160 000</td>
</tr>
<tr>
<td>Canada</td>
<td>100 000</td>
</tr>
<tr>
<td>South Africa</td>
<td>35 000</td>
</tr>
<tr>
<td>Brazil</td>
<td>16 000</td>
</tr>
<tr>
<td>Other Countries</td>
<td>95 000</td>
</tr>
</tbody>
</table>
URANIUM MINING TRENDS

Rossing Open Cast Mine

Uranium Deep Mining

Mining Techniques

Production per Method (%) as in 2003
(total 35 772 t U)

- Open pit mining: 28%
- Deep underground mining: 41%
- ISL mining: 20%
- Co-/By-product: 11%
FROM ZIRCON SAND TO ZIRCONIUM ALLOY INGOTS AT NFC, HYDERABAD

1. **Zircon Sand**
2. **Hf-free ZrO₂ Powder**
3. **Nuclear Grade Zr Sponge**
4. **Compaction of Zr Sponge + alloying elements Briquettes**

- **Zirconium Alloy Ingot**
  - Max. size: 350 mm dia x 2 m height

- **Vacuum Arc Melting Furnace using Consumable Electrode**

- **Electron Beam Welding of Briquettes to form Consumable Electrode**
Major Activities of Zirconium Alloy Fabrication Plant at NFC

- Hot Extrusion of Seamless Zirconium Alloy Tubes
- Pilot Hole Expansion Press
- Pilger Mill for Production of Zirconium Alloy Fuel Tubes
- Array of Zirconium Alloy Fuel Tubes
- Hot Rolling of Zirconium Alloy Sheets
- Cold Swaging of Zirconium Alloy Bar
Production of Natural Uranium Oxide Fuel Bundles for PHWR
### FUEL ASSEMBLY FOR 540 MWe PRESSURISED HEAVY WATER REACTORS

#### ONE FUEL ASSEMBLY CONTAINS

- **ZIRCALOY-4 COMPONENTS**
  - **FUEL TUBES** (13.08 od x 0.38 t x 485.8 l) ... 37 Nos
  - **END CAPS** (13.20 φ x 5.3 t) ..... 74 Nos
  - **SPACER PADS** (8.6 l x 2.5 w x 3.22 t) ..... 12 Nos
    - (8.6 l x 2.5 w x 0.87 t) ..... 144 Nos
  - **BEARING PADS** (33.5 l x 2.5 w x 1.3 t) ..... 36 Nos
    - (32.5 l x 2.5 w x 1.0 t) ..... 18 Nos
  - **END PLATES** (90.98 φ x 1.60 t) ..... 2 Nos

- **NATURAL URANIUM DIOXIDE**
  - **PELLETS** (12.20 φ x 13.29 l) ..... 1295 Nos.
  - **WEIGHT OF PELLETS** ..... 21.65 kg

#### ONE PHWR 540 CORE CONTAINS

- **FUEL ASSEMBLIES** ..... 5096 Nos
- **URANIUM DIOXIDE** ..... 110 Tons
- **FUEL Assly. WELD JOINTS** ..... 31,69,712 Nos

**ELECTRIC POWER FROM ONE FUEL ASSEMBLY** .... 10,44,000 units

---

A typical Calandria Vessel for CANDU-PHWR 500 Unit (392 horizontal fuel channels - each channel contain 13 nos. of 37 element fuel bundles)
### Main long-lived or parent Radionuclides present in Irradiated Fuel

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Half-life (Year)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Uranium</strong></td>
<td></td>
</tr>
<tr>
<td>U-234</td>
<td>$2.46 \times 10^5$</td>
</tr>
<tr>
<td>U-235</td>
<td>$7.04 \times 10^8$</td>
</tr>
<tr>
<td>U-236</td>
<td>$2.34 \times 10^7$</td>
</tr>
<tr>
<td>U-238</td>
<td>$4.47 \times 10^9$</td>
</tr>
<tr>
<td><strong>Actinides (α emitters)</strong></td>
<td></td>
</tr>
<tr>
<td>Pu-238</td>
<td>87.7</td>
</tr>
<tr>
<td>Pu-239</td>
<td>24,100</td>
</tr>
<tr>
<td>Pu-240</td>
<td>6,560</td>
</tr>
<tr>
<td>Pu-241</td>
<td>14.35</td>
</tr>
<tr>
<td>Pu-242</td>
<td>$3.74 \times 10^5$</td>
</tr>
<tr>
<td>Np-237</td>
<td>$2.14 \times 10^6$</td>
</tr>
<tr>
<td>Am-241</td>
<td>432.7</td>
</tr>
<tr>
<td>Am-243</td>
<td>7,368</td>
</tr>
<tr>
<td>Cm-245</td>
<td>$8.5 \times 10^3$</td>
</tr>
<tr>
<td>Cm-246</td>
<td>$4.73 \times 10^3$</td>
</tr>
<tr>
<td><strong>Fission Products (β/γ emitters)</strong></td>
<td></td>
</tr>
<tr>
<td>Sc-79</td>
<td>$6.5 \times 10^5$</td>
</tr>
<tr>
<td>Zr-93</td>
<td>$1.5 \times 10^6$</td>
</tr>
<tr>
<td>Tc-99</td>
<td>$2.13 \times 10^5$</td>
</tr>
<tr>
<td>Pd-107</td>
<td>$6.5 \times 10^6$</td>
</tr>
<tr>
<td>Sn-126</td>
<td>$1 \times 10^6$</td>
</tr>
<tr>
<td>I-129</td>
<td>$1.57 \times 10^7$</td>
</tr>
<tr>
<td>Cs-135</td>
<td>$2.3 \times 10^6$</td>
</tr>
<tr>
<td><strong>Activation Products (β/γ emitters)</strong></td>
<td></td>
</tr>
<tr>
<td>C-14</td>
<td>5,715</td>
</tr>
<tr>
<td>Ni-59</td>
<td>$7.6 \times 10^4$</td>
</tr>
<tr>
<td>Ni-63</td>
<td>100</td>
</tr>
<tr>
<td>Zr-93</td>
<td>$1.53 \times 10^6$</td>
</tr>
<tr>
<td>Nb-94</td>
<td>$2.03 \times 10^4$</td>
</tr>
</tbody>
</table>
Closing the fuel cycle facilitates:

a) Waste minimization [in terms of Pu & Minor Actinides (MA)]

b) Resource (natural uranium) utilization

Radiotoxicity of wastes after 1000 years

- Pu: ~1%
- MA: ~0.1%
- FP: 3-4%

Values:

- Reactor: 94-96% U, 1% Pu, 3-4% FP, 0.1% MA
- Spent Fuel: 96% valuable materials
- Wastes: 4% valuable materials

Graph: Evolution of the radiotoxicity over time.
<table>
<thead>
<tr>
<th><strong>Country</strong></th>
<th><strong>Name</strong></th>
<th><strong>Type</strong></th>
<th><strong>Power</strong></th>
<th><strong>Driver Fuel</strong></th>
</tr>
</thead>
<tbody>
<tr>
<td>United Kingdom</td>
<td>DFR</td>
<td>Experimental</td>
<td>↓</td>
<td>U-Mo (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>PFR</td>
<td>Prototype</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td>Germany</td>
<td>KNK-II</td>
<td>Experimental</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>SNR-300</td>
<td>Prototype (not operated)</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td>India</td>
<td>FBTR</td>
<td>Experimental</td>
<td>↓</td>
<td>(Pu₀.₇, U₀.₃)C ↓</td>
</tr>
<tr>
<td></td>
<td>PFBR</td>
<td>Prototype</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td>China</td>
<td>CEFR</td>
<td>CEFR (under construction)</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
<tr>
<td>Korea (Republic of)</td>
<td>KALIMER</td>
<td>Demonstration (under construction)</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
<tr>
<td>Russia</td>
<td>BR-5/BR-10</td>
<td>Experimental</td>
<td>↓</td>
<td>PuO₂/UC/UN ↓</td>
</tr>
<tr>
<td></td>
<td>BOR-60</td>
<td>Experimental</td>
<td>↓</td>
<td>UO₂ (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>BN-600</td>
<td>Commercial</td>
<td>↓</td>
<td>UO₂ (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>BN-350 (Kazakhstan)</td>
<td>Prototype</td>
<td>↓</td>
<td>UO₂/(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td></td>
<td>BN-800</td>
<td>Planned</td>
<td>↓</td>
<td></td>
</tr>
<tr>
<td>USA</td>
<td>EBR-1, EBR-II, FFTF</td>
<td>Experimental</td>
<td>↓</td>
<td>U-Fs &amp; U-Pu-Zr ↓</td>
</tr>
<tr>
<td></td>
<td>FFTF</td>
<td>Experimental</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td>France</td>
<td>Rapsodie</td>
<td>Prototype</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>Phenix</td>
<td>Commercial</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td></td>
<td>SuperPhenix-I</td>
<td>(shutdown)</td>
<td>↓</td>
<td>(U, Pu)O₂ ↓</td>
</tr>
<tr>
<td>Japan</td>
<td>JOYO</td>
<td>Experimental</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
<tr>
<td></td>
<td>Monju</td>
<td>Prototype</td>
<td>↓</td>
<td>(U, Pu)O₂ (HEU) ↓</td>
</tr>
</tbody>
</table>
Liquid Metal-cooled Fast Reactor Fuel Cycle with multiple recycling of U, Pu and Minor Actinides
NUCLEAR FUELS are:

- made of fissile isotopes $^{235}\text{U}$, $^{239}\text{Pu}$ or $^{233}\text{U}$ mostly with judicious combination of fertile isotopes $^{238}\text{U}$ or $^{232}\text{Th}$.
- used in the form of metals, alloys, oxides, carbides, nitrides, cer-met or cer-cer of the above fissile and fertile materials.
- used in the geometric shapes of rod, pin, plate or microsphere.
- encapsulated or clad with Zr-alloys (water-cooled reactor), SS (sodium-cooled fast reactor) or multi-layer coatings of pyrolytic carbon and silicon carbide (high temperature gas-cooled reactor).
# Conventional & Advanced Fuels For Nuclear Power Reactors

<table>
<thead>
<tr>
<th>Reactors</th>
<th>Conventional Fuels</th>
<th>Advanced/Alternative Fuels</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Light Water Reactor (LWR):</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR, PWR &amp; VVER</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel (pellets)</td>
<td>LEU (U-235 ≤ 5%) as UO₂</td>
<td>LEU (U-235 5-10%) Mixed Uranium Plutonium Oxide (≤10% PuO₂) [LEU+Minor Actinide (MA)] oxide for large grain size and controlled porosity 'Proliferation Resistant' spent fuel PuO₂ in Inert Matrix for burning ‘Pu’</td>
</tr>
<tr>
<td>Cladding</td>
<td>Zircaloy 2 (BWR)</td>
<td>Zr-Sn-Nb-Fe &amp; Zr-Nb-O alloys</td>
</tr>
<tr>
<td></td>
<td>Zircaloy 4 (PWR)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Zr-1% Nb (VVER)</td>
<td></td>
</tr>
<tr>
<td>Burning up</td>
<td>20 000-30 000 MWD/t</td>
<td>High : up to 60 000 MWD/t</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ultra High : up to 80 000 MWD/t</td>
</tr>
<tr>
<td><strong>Pressurized Heavy Water Reactor (PHWR)</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel (pellets)</td>
<td>Natural UO₂</td>
<td>REU, SEU in the form of UO₂, (U,Pu)O₂ (Th,Pu)O₂ &amp; (Th,U233)O₂, containing up to 2% fissile material. Large grain size and controlled porosity PuO₂ in Inert Matrix for burning ‘Pu’</td>
</tr>
<tr>
<td>Cladding</td>
<td>Zircaloy 4</td>
<td>Zircaloy 4</td>
</tr>
<tr>
<td>Burning up</td>
<td>6 700 MWD/t</td>
<td>15 000 – 20 000 MWD/t</td>
</tr>
</tbody>
</table>
## Conventional & Advanced Fuels For Nuclear Power Reactors

<table>
<thead>
<tr>
<th><strong>Liquid Metal-cooled Fast Breeder Reactor (LMFBR)</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel (pellets/particles/pins)</td>
<td>HEU in the form of UO2 &amp; (U,Pu)O2 (≤25% Pu) He-bonded pins</td>
</tr>
<tr>
<td>Cladding</td>
<td>Stainless Steel D-9</td>
</tr>
<tr>
<td>Burnup</td>
<td>100 000 MWd/t</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>1.0 – 1.2</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>High Temperature Gas Cooled Reactors (HTR) (coated microspheres)</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>Multi-layer (pyrolytical carbon &amp; SiC-coated) Uranium Oxide fuel particles (BISO or TRISO) embedded in graphite</td>
</tr>
<tr>
<td>Cladding</td>
<td>Stainless steel (type ferritic HT-9 or Oxide dispersed ODS)</td>
</tr>
<tr>
<td>Burnup</td>
<td>Up to 200 000 MWd/t</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>1.2 – 1.6</td>
</tr>
</tbody>
</table>

- Na-bonded (U,Pu)C, (U,Pu)N & U-Pu-Zr, (≤25% Pu) fuel with/without MA
- He-bonding also for carbide/nitride (PuO2+ThO2) for burning 'Pu'
- He-bonded vibratory compacted oxide, carbide and nitride fuel pins
- 'Pu and (Pu,MA) in inert matrix for burning (U/Th+MA) in blanket for 'Proliferation Resistance'' in irradiated blanket

HTR: High Temperature Gas Cooled Reactors
IAEA: International Atomic Energy Agency
Figure 9. Three-stage nuclear power programme in India involving ‘closed fuel cycles’
Process flowsheet followed in India for fabrication of plutonium rich mixed carbide fuel pellets for Fast Breeder Test Reactor at IGCAR, Kalpakkam

Analysis
- Specific surface area, chemical purity of oxide & graphite powders, U&Pu contents, isotopic analysis & O/M ratios

Analysis
- U, Pu (XRF) C (Combustion)

Analysis
- U,Pu (XRF), C (combustion), O & N (Inert gas fusion) MC, M₂C₂ & MO₂ (XRD)

UO₂
Powder

PuO₂
Powder

C
Powder

Weighing, Comilling, Blending & Tableting

Carbothermic Reduction

Crushing & Milling

Binder

Precompaction, Granulation, Final Compaction & Sintering

Final Process Control (U, Pu, C, O, MC, M₂C₂) linear mass, diameter & density

Controlled Oxidation

Reject recycle

Inspection (100 %) Visual, dimension & density

Reject recycle

ACCEPTED CARBIDE PELLETS FOR ENCAPSULATION IN SS 316 CLADDING TUBES

High Temperature High Vacuum Furnace for Carbothermic Synthesis of (U,Pu)C

Automatic Fuel Pellet Inspection Machine (Dimension & Density)
Key Features

- Thorium bearing fuel \([(\text{Th} + \text{Pu})\text{O}_2 \text{ Mox}, (\text{Th} + ^{233}\text{U})\text{O}_2 \text{ Mox}]\)
- Hollow cylindrical \((\text{ZrO}_2 - \text{Dy}_2\text{O}_3)\) displacer rod.
- Emergency core cooling water injected into the cluster through the holes in displacer rod.
- Low pressure drop design.

AHWR FUEL CLUSTER (D5)
"Pellet-Impregnation Techniques" for manufacturing high density (Th,U\(^{233}\))O\(_2\) fuel pellets

Th\(_2\)O\(_2\) microspheres (200-1000 µ) "porous" & free-flowing ("Sol-gel" or "powder" routes)

Th\(_2\)O\(_2\) Powder (MgO doped)

Cold Pelletisation

Solid Pellets or Annular Pellets

Partial Sintering (Low Temperature)

Low Density (<80% TD) Th\(_2\)O\(_2\) Pellets

Vacuum-Impregnation in Uranyl (U\(^{233}\)) Nitrate Solution - (Single or Multiple with or without Microwave Heating)

Drying, Calcination & High Temperature Sintering

High Density (Th,U\(^{233}\))O\(_2\) pellets (Solid or Annular)
Advanced fuel cycle schemes

PWR SF → OREOX → DUPIC
   ↓                  ↓
Reduction to Metal   Recovery of U, TRU-RE
(ER Process)         (Electrorefining, Reductive Extraction)
↓                       ↓
Storage/ Conditioning
↓
Disposal

U-TRU-RE
   ↓
Injection Casting

Electrorefining
   ↓
Chloride Volatility Process
   ↓
TRU Electrowinning
   ↓
ADS Transmuter Fuel Assembly Fabrication

ADS
   ↓
HYPER

IAEA
## COMMERCIAL SPENT URANIUM OXIDE FUEL REPROCESSING PLANTS IN OPERATION AND UNDER CONSTRUCTION IN THE WORLD

<table>
<thead>
<tr>
<th>Country / Company</th>
<th>Facility / Location</th>
<th>Fuel Type</th>
<th>Capacity (tHM/year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>France, COGEMA</td>
<td>UP2 and UP3, La Hague</td>
<td>LWR</td>
<td>1700</td>
</tr>
<tr>
<td>UK, BNFL</td>
<td>Thorp, Sellafield</td>
<td>LWR, AGR</td>
<td>1200</td>
</tr>
<tr>
<td>UK, BNFL</td>
<td>B205 Magnox</td>
<td>Magnox GCR</td>
<td>1500</td>
</tr>
<tr>
<td>Russian Federation, Minatom</td>
<td>RT-1 / Tcheliabinsk-65 Mayak 400</td>
<td>VVER</td>
<td>400</td>
</tr>
<tr>
<td>Japan, JNC</td>
<td>Tokai-Mura</td>
<td>LWR, ATR</td>
<td>90</td>
</tr>
<tr>
<td>Japan, JNFL</td>
<td>Rokkasho-Mura (under construction)</td>
<td>LWR</td>
<td>800</td>
</tr>
<tr>
<td>India, BARC</td>
<td>PREFRE-1, Tarapur</td>
<td>PHWR</td>
<td>100</td>
</tr>
<tr>
<td></td>
<td>PREFRE-2, Kalpakkam</td>
<td>PHWR</td>
<td>100</td>
</tr>
<tr>
<td>China, CNNC</td>
<td>Diowopu (Ganzu)</td>
<td>LWR</td>
<td>25-50</td>
</tr>
<tr>
<td>Country / Company</td>
<td>Facility / Location</td>
<td>Fuel Type</td>
<td>Capacity (tHM/year)</td>
</tr>
<tr>
<td>-------------------</td>
<td>---------------------</td>
<td>-----------------</td>
<td>---------------------</td>
</tr>
<tr>
<td>France, COGEMA</td>
<td>Cadarache</td>
<td>LWR, FBR</td>
<td>40</td>
</tr>
<tr>
<td>France, COGEMA</td>
<td>Marcoule-Melox</td>
<td>LWR</td>
<td>100</td>
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<td>Belgium, Belgonucleaire</td>
<td>Dessel</td>
<td>LWR</td>
<td>40</td>
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<tr>
<td>UK, BNFL</td>
<td>Sellafield SMP</td>
<td>LWR</td>
<td>120</td>
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<td>UK</td>
<td>Sellafield MDF</td>
<td>LWR</td>
<td>8</td>
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<tr>
<td>Russian Federation, Minatom</td>
<td>Chelyabinsk</td>
<td>FBR</td>
<td>60</td>
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<td>Japan, JNC</td>
<td>Tokai-Mura</td>
<td>ATR</td>
<td>10</td>
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<tr>
<td>Japan, JNFL</td>
<td>Rokkasho</td>
<td>LWR</td>
<td>130</td>
</tr>
<tr>
<td>India, AFFF, BARC</td>
<td>Tarapur</td>
<td>LWR, PHWR &amp; FBR</td>
<td></td>
</tr>
</tbody>
</table>
The GANEX concept: Group Actinides EXtraction

Fuel → Dissolution → Crystallization → U (80 %) → Fuel recycling

U, Pu, M.A. F.P. → HNO₃ → Co-extraction

Organic solvent → Co-extraction → An stripping → Ln stripping

Specific An complexant → Ln stripping

F.P. → Ln

Dilute acid → Ln recycling

Solvent recycling
Objectives of advanced methods of fabrication of ceramic nuclear fuel pellets

<table>
<thead>
<tr>
<th>Safety</th>
<th>Economics</th>
<th>Performance</th>
</tr>
</thead>
<tbody>
<tr>
<td>• Avoid generation and handling of powder of fuels for minimising:</td>
<td>• Minimise process steps</td>
<td>• Tailor make fuel microstructure for higher burn up</td>
</tr>
<tr>
<td>- radiotoxic dust hazard</td>
<td>• Reduce fuel synthesis &amp; sintering temperatures</td>
<td>- High density (≥96% T.D.), closed “porosity” and large (&gt;25µ) grain size for LWR &amp; PHWR</td>
</tr>
<tr>
<td>- fire hazard (for carbide &amp; nitride fuels)</td>
<td>• Reduce gas cost during synthesis and sintering</td>
<td>- Low density (&lt;85% T.D.) “open” porosity and small (&lt;5µ) grain size for LMFBR</td>
</tr>
<tr>
<td>• Fabrication flow sheet should be amenable to automation &amp; remotisation</td>
<td>- gas purification and recirculation</td>
<td>- Excellent micro-homogenity of fissile material in fuel</td>
</tr>
<tr>
<td>- for minimising personnel exposure to radiation</td>
<td>- alternative less expensive gas</td>
<td>- avoid fine pores (&lt;1µ) for minimising in-pile densification</td>
</tr>
<tr>
<td></td>
<td>• Reduce process losses and rejects</td>
<td></td>
</tr>
</tbody>
</table>
“Sol-Gel-Microsphere Pelletization (SGMP)” and “Vibro-Sol” Processes for Manufacturing Mixed Uranium Plutonium Oxide (MOX), Monocarbide (MC) and Mononitride (MN) Fuels for LMFR

Preparation of Sol-Gel-Microspheres of Oxide or Oxide+C: Ammonia External or Internal Gelation Process
Tailored Microstructure of Oxide Fuel Pellets prepared by Sol-Gel Microsphere Pelletisation Process

Porous Microspheres (easily crushable)

Non-Porous Microspheres (hard and not easily crushable)

Direct pelletisation followed by Sintering

Sintered Fuel Pellets of High Density & Uniformly distributed ‘closed’ pores
- suitable for PHWRs & LWRs

Sintered Fuel Pellets with black berry structure, low density and ‘open’ pores
- suitable for LMFBRs
Microstructures of high density oxide and mixed oxide fuel pellets fabricated by SGMP route for PHWR and LWR
Microstructure & Image Analysis of ThO$_2$-2%UO$_2$ prepared by Sol-Gel Microsphere Pelletisation (SGMP) Process

Porous Microsphere (SEM Picture)  As-Polished Microstructure  Etched Microstructure

Image Analysis of entire cut section of pellet (for determining undissolved microsphere boundary)

Image Analysis of selected areas (54µm x 54µm) (for average pore size & distribution)
OBJECTIVES OF INERT MATRIX FUEL

- Minimizing “proliferation risk” of plutonium (~ 200 tons of weapon-grade and ~ 1000 tons civilian grade) by using them in nuclear power reactors in operation

- Minimizing “Minor Actinides” (MA: Np, Am & Cm) and in turn radiotoxicity in waste

- In some cases minimizing ‘proliferation risk’ of weapon-grade (> 90 % $^{235}$U) uranium (though conventional process is down-blending)
**Inert Matrix**

- Neutron (very low capture and absorption cross-sections)
- Chemical compatibility with
  - Fuel
  - Cladding
  - Coolant
- Consideration of direct disposal after use

**Fuel**

- ‘Plutonium form’ – alloys and compounds
- Utilization of Minor Actinides together with plutonium
- Weapon-grade HEU \((^{235}\text{U} > 90\%)\) – alloys or compounds
# Examples of Inert Matrix

<table>
<thead>
<tr>
<th>Inert Matrix type</th>
<th>Inert Matrix formula</th>
</tr>
</thead>
<tbody>
<tr>
<td>Element</td>
<td>C, Mg, Al, Si, Cr, V, Zr, Mo, W</td>
</tr>
<tr>
<td>Inter-metallics</td>
<td>AlSi, AlZr, ZrSi</td>
</tr>
<tr>
<td>Alloy</td>
<td>Stainless steel, zirconium alloys</td>
</tr>
<tr>
<td>Carbide</td>
<td>SiC, TiC, ZrC</td>
</tr>
<tr>
<td>Nitrides</td>
<td>AlN, TiN, ZrN, CeN,</td>
</tr>
<tr>
<td>Binary oxide</td>
<td>MgO, Y$_2$O$_3$, ZrO$_2$, CeO$_2$</td>
</tr>
<tr>
<td>Ternary oxide</td>
<td>MgAl$_2$O$_4$, Y$_3$Al$<em>5$O$</em>{12}$, ZrSiO$_4$</td>
</tr>
<tr>
<td>Oxide solid solution</td>
<td>Y$<em>y$Zr$</em>{1-y}$O$<em>2$-$y$/2, Mg$</em>{(1-x)}$Al$<em>{(2+x)}$O$</em>{(4-x)}$</td>
</tr>
</tbody>
</table>
### Example of heterogenous materials as IMF’s

<table>
<thead>
<tr>
<th>Design</th>
<th>Composition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Solid solution</td>
<td>$A_{n_z}Y_{y}Pu_{x}Zr_{1-y}O_{2-y}$ *</td>
</tr>
<tr>
<td>Cercer</td>
<td>$MgAl_{2}O_{4} - Y_{y}Pu_{x}Zr_{1-y}O_{2-y/2}$ *</td>
</tr>
<tr>
<td>Cermet</td>
<td>$Zr - Y_{y}Pu_{x}Zr_{1-y-x}O_{2-y/2}$ *</td>
</tr>
<tr>
<td>Metmet</td>
<td>$PuAl_{4}*-Al$</td>
</tr>
</tbody>
</table>

### Examples of Inert Matrix design and additives

<table>
<thead>
<tr>
<th>Additive type</th>
<th>Additive formula</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Burnable poison</strong></td>
<td>B, Gd, Dy, Ho, Er, Eu</td>
</tr>
<tr>
<td><strong>Resonance additive</strong></td>
<td>Fe, W, Th, U</td>
</tr>
<tr>
<td><strong>Stabiliser</strong></td>
<td>$Y_{2}O_{3}$, CaO in $ZrO_{2}$</td>
</tr>
<tr>
<td></td>
<td>$Al_{2}O_{3}$ in $SiC$</td>
</tr>
</tbody>
</table>
Development of Nitride Pellet Synthesis at LANL

Old

- Am Oxide feed
- Pu Oxide feed
- Blend/Mill/Cold-press
- Sinter 1400-1800°C
- Green pellet

Modified

- Pu Oxide feed
- Am Oxide feed
- Zr Oxide feed
- Solutionize 1700°C
- Carbothermic reduction
- Solution (Pu,Am,Zr) Oxide feed
- Solution (Pu,Am,Zr) Nitride feed
- Mill/Cold-press
- Green pellet
- Sinter 1400-1800°C

Irradiation behavior of rock-like oxide fuels

### Dimensional variation and fractional gas release of ROX fuels

<table>
<thead>
<tr>
<th>Pin</th>
<th>Maximum temperature (K)</th>
<th>$\Delta \Phi / \Phi$ (%)</th>
<th>$\Delta V / V$ (%)</th>
<th>Xe FGR (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>ZM-7</td>
<td>1040</td>
<td>–</td>
<td>–</td>
<td>63</td>
</tr>
<tr>
<td>ZM-6</td>
<td>1270</td>
<td>–</td>
<td>–</td>
<td>61</td>
</tr>
<tr>
<td>ZM-4</td>
<td>980</td>
<td>10.8</td>
<td>–</td>
<td>63</td>
</tr>
<tr>
<td>SD</td>
<td>1850</td>
<td>2.7</td>
<td>5.5</td>
<td>38</td>
</tr>
<tr>
<td>SH</td>
<td>2080</td>
<td>5.0</td>
<td>10.2</td>
<td>22</td>
</tr>
<tr>
<td>Z</td>
<td>1580</td>
<td>2.0</td>
<td>&lt;4.0</td>
<td>2.2</td>
</tr>
<tr>
<td>CD</td>
<td>1930</td>
<td>2.1</td>
<td>4.3</td>
<td>22</td>
</tr>
<tr>
<td>CH</td>
<td>1830</td>
<td>2.8</td>
<td>5.7</td>
<td>7.8</td>
</tr>
</tbody>
</table>

### Test fuel matrix

<table>
<thead>
<tr>
<th>Pin</th>
<th>Composition (mol%)</th>
<th>YSZ inclusion size</th>
<th>Fissile density ($10^{20}/\text{cm}^3$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>YSZ$^a$</td>
<td>$\text{PuO}_2$$^b$</td>
<td>$\text{UO}_2$$^c$</td>
<td>$\text{MgAl}_2\text{O}_4$</td>
</tr>
<tr>
<td>ZM</td>
<td>16.7</td>
<td>11.1</td>
<td>–</td>
</tr>
<tr>
<td>SD</td>
<td>20.0</td>
<td>–</td>
<td>37.1</td>
</tr>
<tr>
<td>SH</td>
<td>20.0</td>
<td>–</td>
<td>37.1</td>
</tr>
<tr>
<td>Z</td>
<td>80.0</td>
<td>–</td>
<td>20.0</td>
</tr>
<tr>
<td>CD</td>
<td>16.5</td>
<td>–</td>
<td>30.6</td>
</tr>
<tr>
<td>CH</td>
<td>16.5</td>
<td>–</td>
<td>30.6</td>
</tr>
</tbody>
</table>

$^a$ YSZ = 79.9 mol% ZrO$_2$ + 20.1 mol% YO$_{1.5}$.

$^b$ Pu isotopic composition (at%) was 94.3, 5.3 and 0.4 for $^{239}$Pu, $^{240}$Pu and $^{241}$Pu, respectively.

$^c$ 19.6% enriched UO$_2$. 

R&D on Advanced LMFR Fuels and Advanced Methods of Fuel Fabrication

- **Ceramic Nuclear Fuels**
  - Conventional: \((U, Pu)O_2\)
  - Advanced: \((U, Pu)C\) & \((U, Pu)N\) with/without Minor Actinides

- **Advanced methods of fabrication of ceramic fuels:**
  - Dust-free advanced fabrication processes like vibratory compaction, vibro-sol & sol-gel microsphere pelletization

- **Metallic Fuels:**
  - \(U-Pu-Zr\), \(Th-U-Pu-Zr\) & \(U-Pu\) (for high breeding)

- **Fuel Cladding, Hexcans & Other Fuel Assembly components:**
  - Ferritic stainless steel HT9 & Oxide dispersed stainless steel with minimum radiation damage and void swelling

- **Advanced fabrication processes should be amenable to secured automated fabrication, real-time accounting of special nuclear material and proliferation resistance**
Conventional and Advanced Methods of Spent Fuel Reprocessing

AQUEOUS PROCESS:
Dissolution of spent fuel in Nitric acid followed by purification by solvent extraction by adapting the PUREX process, using TriButyl Phosphate (TBP) as solvent, is being used on an industrial scale for reprocessing of spent UO2 and MOX fuels. The PUREX process is not suitable for mixed carbide fuel but could be utilized for reprocessing mixed nitride and metallic fuels.

Modifications are being incorporated in PUREX process to make it proliferation resistant and economic.

PYROPROCESSING:
– Pyroprocessing involving electrolytic reduction
This route has been initially developed on a pilot plant scale for reprocessing of spent metallic fuels (U-Zr & U-Pu-Zr) in USA and was successfully extended on a laboratory scale for reprocessing of carbide and nitride fuels. The pyroprocessing route is yet to be adapted on an industrial scale.
In recent years, the Russian Federation has successfully demonstrated the pyroprocessing route for reprocessing of spent oxide fuels on a pilot plant scale.
– Pyroprocessing involving fluoride volatilization
The process includes fluorination followed by distillation. The method has so far been demonstrated on a laboratory scale only.