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

Trieste, 14 November , 2005

Opening Address

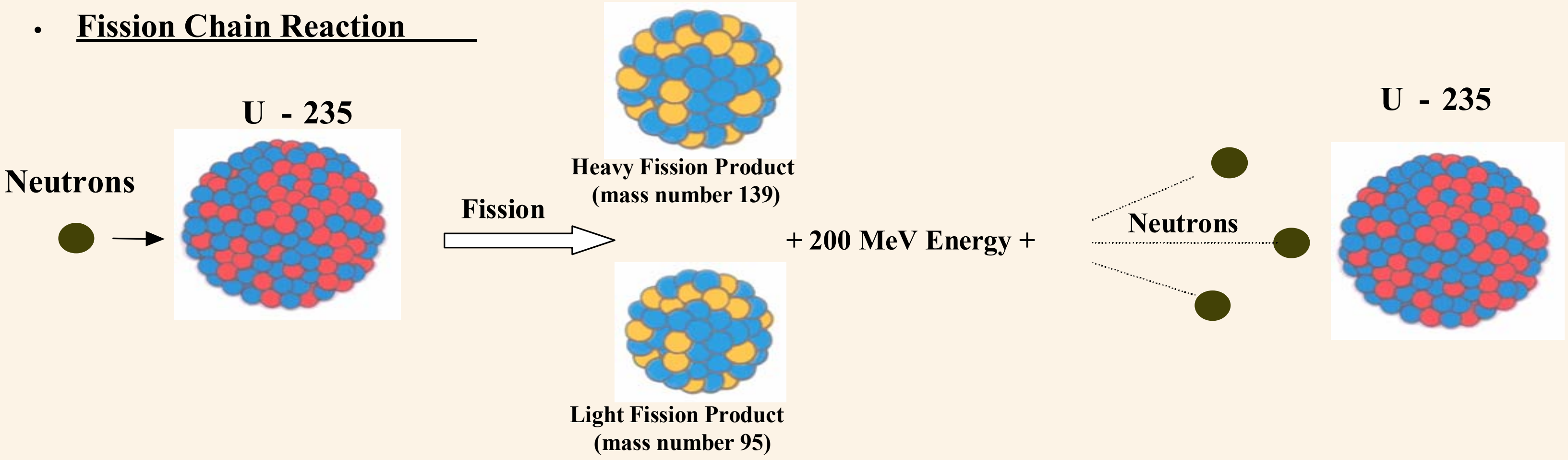
Chaitanyamoy Ganguly

Head, Nuclear Fuel Cycle & Materials Section

IAEA
International Atomic Energy Agency

NUCLEAR FISSION

- **Fission Chain Reaction**



Neutrons

U - 235

Fission

Heavy Fission Product
(mass number 139)

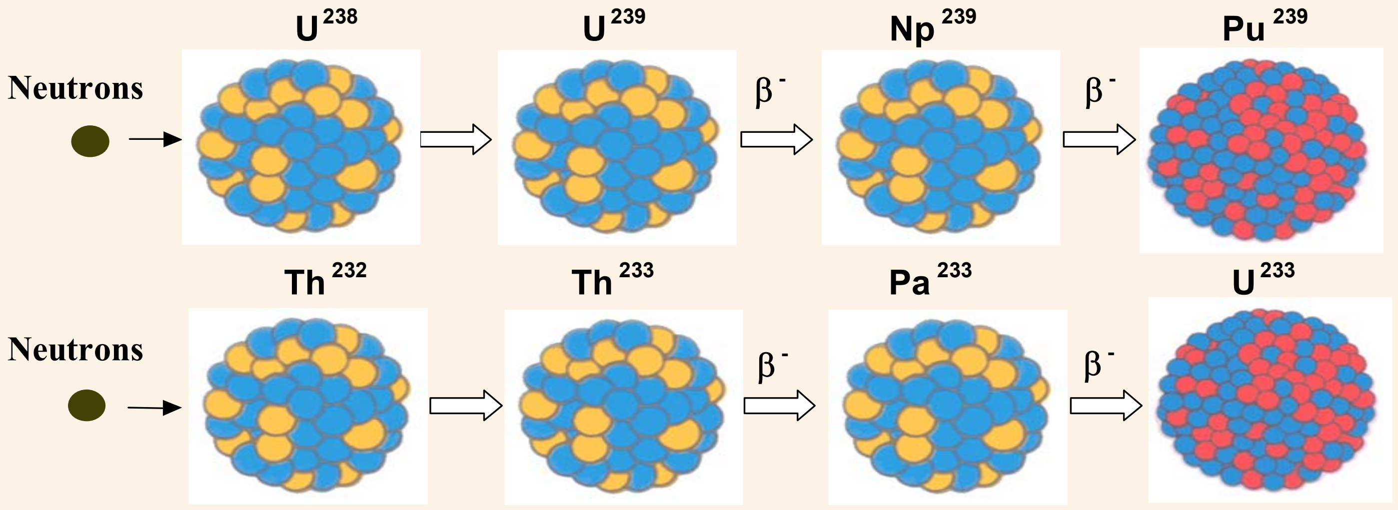
Light Fission Product
(mass number 95)

+ 200 MeV Energy +

Neutrons

U - 235

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



Neutrons

U^{238} ⇒ U^{239} $\xrightarrow{\beta^-}$ Np^{239} $\xrightarrow{\beta^-}$ Pu^{239}

Neutrons

Th^{232} ⇒ Th^{233} $\xrightarrow{\beta^-}$ Pa^{233} $\xrightarrow{\beta^-}$ U^{233}

FISSILE & FERTILE ISOTOPES

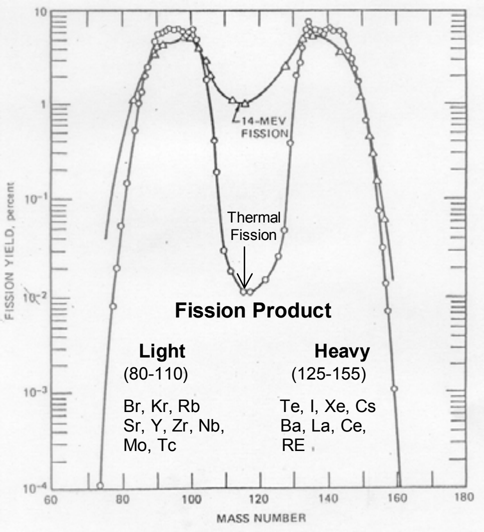
URANIUM

- 4 ppm in earth's crust, Klaproth, 1789
- Natural Uranium:

U^{238}(fertile): 99.3%; 4.5×10^{9}y
U^{235}(fertile): 0.7%; 8.0×10^{8} y

U^{235}_{92}(fissile) + n^{1}_{0} $\;U^{236}_{92}$ $\quad M_1(95)+M_2(139)+(2\text{-}3)n^1+200$ MeV

U^{238}_{92}(fertile) + n^{1}_{0} $\;U^{239}_{92} \xrightarrow[23.5\text{ m}]{\beta^-} Np^{239}_{93} \xrightarrow[2.35\text{ days}]{\beta^-} Pu^{239}_{94}$ (fissile)



THORIUM

- 12 ppm in earth's crust, Berzellius, 1828

Natural Thorium: No fissile isotope,

Fertile Th^{232}_{90} (fertile); 1.4×10^{10}y

Th^{232}_{90} (fertile) + $n^{1}_{0} \longrightarrow Th^{233}_{90} \xrightarrow[23.5\text{ min}]{} Pa^{233} \xrightarrow[27\text{ days}]{} U^{233}$ (fissile)

PLUTONIUM

- Does not occur in nature; man-made, Seaborg, Wahl and Kennedy, 1941
  - Pu^{239} : ∝ ; 2.4×10^{4} y
  - Pu^{240} : ∝, n ; 6.58×10^{3} y
  - Pu^{241} : β^- ; 13y, Am241 strong γ emitter
  - Pu^{242} : ∝, n ; 3.79×10^{5} y
  - Pu^{238} : ∝, n ; 86.4 y
- Maximum limits

body burden : 0.18 – 0.65 μ g
concentration in air : 2×10^{-12} μ curie per cc
concentration in water : 10^{-4} μ curie per cc

NUCLEAR REACTOR

REACTOR CORE MATERIALS

- **Fuel (Plate or Pin)**

  Fissile (U^{235}, Pu^{239} or U^{233} & Fertile (U^{238} or Th^{232}) 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**

  H_2O/ D_2O : Water cooled reactor.

  CO_2/He : Gas cooled reactor.

  Na : LMFBR

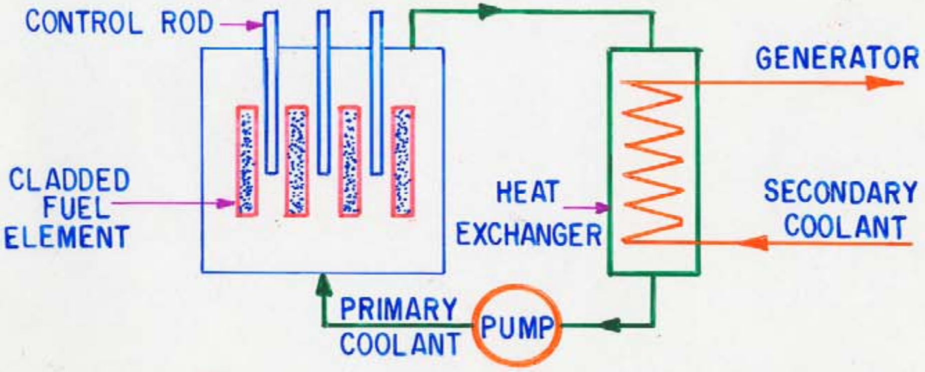
- **Moderator (Only for Thermal Reactor)**

  H_2O/D_2O or graphite

- **Control Rod**

  B, Cd, Hf, Gd.

- Source of heat energy
- Source of neutrons



$^{235}_{92}U + {}^{1}_{0}n = {}^{236}_{92}U =$ Fission Fragment 'A' (Sr, Y, Zr, Wb, Mo etc. (95)) + Fission Fragment 'B' (Xe, Cs, Ba, La etc. (139)) + (2-3) $^{1}_{0}n$ +~200 MeV

$^{238}_{92}U + {}^{1}_{0}n = {}^{239}_{92}U \xrightarrow[23.5\,min]{\beta^-} Np^{239} \xrightarrow[2.35\,days]{\beta^-} {}^{239}_{94}Pu$ (Fissile)

$^{232}_{90}Th + {}^{1}_{0}n = {}^{233}_{90}Th \xrightarrow[23.5\,min]{\beta^-} {}^{233}_{91}Pa \xrightarrow[27\,days]{} {}^{233}_{92}U$ (Fissile)

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.

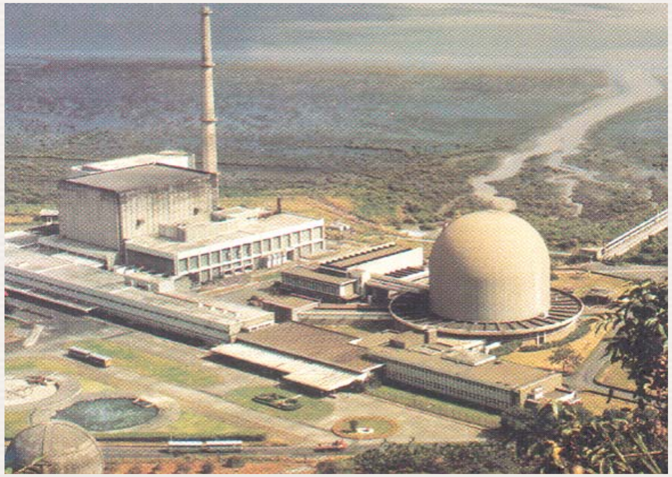
NUCLEAR REACTORS & THEIR APPLICATIONS

Source of intense heat energy & Source of neutrons

- **Power Reactors:**

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

- **Non-Power Reactors:**

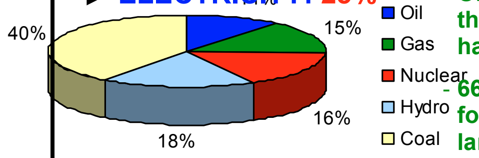


- Production of Radioisotopes
  - Nuclear Medicine, Radiopharmaceuticals Radioimmunotherapy - ^{99m}Tc, I ^{131}, Sm ^{153}, P ^{32}, etc.
  - Cancer Diagnosis & Therapy
  - Sterilisation of Medical Kits, Hospital Wastes & Sewage
  - Food Irradiation & Preservation
  - γ-radiography - Co ^{60}, Ir ^{192} & Cs ^{137}
- Production of Fissile Isotopes (Pu ^{239} and U ^{233})
- Neutron Radiography, Neutron Diffraction & Neutron Activation Analysis
- Irradiation-Testing of Materials
- Training, Education & Basic Research

Nuclear Energy for Tomorrow – to Combat CO_2 emission & Global Warming

ENERGY PRESENTLY UTILISED:

ELECTRICITY: 25%



- 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 CO_2 greenhouse gas to the environment and in turn global warming.
- Fossil-fuel would be significantly depleted by the end of this Century.

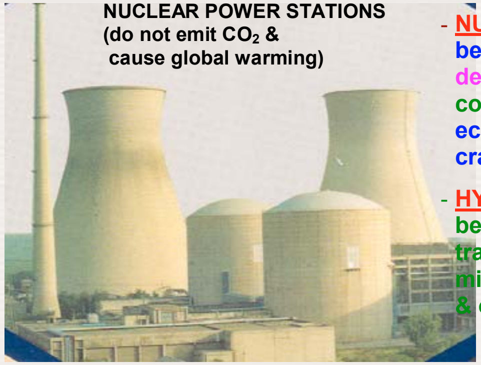
TRANSPORTATION: 75%

- Cars, trucks, trains and planes use 'fossil fuels': gasoline, diesel, jet fuel, etc. releasing CO_2 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 CO_2 emission to the environment.

Fossil Fuels release 70 million tons CO_2 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 CO_2 emission that cause global warming?

NUCLEAR ENERGY IS THE INEVITABLE OPTION !



NUCLEAR POWER STATIONS
(do not emit CO_2 &
cause global warming)

- 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 CO_2 greenhouse gas emissions & other automobile pollution.

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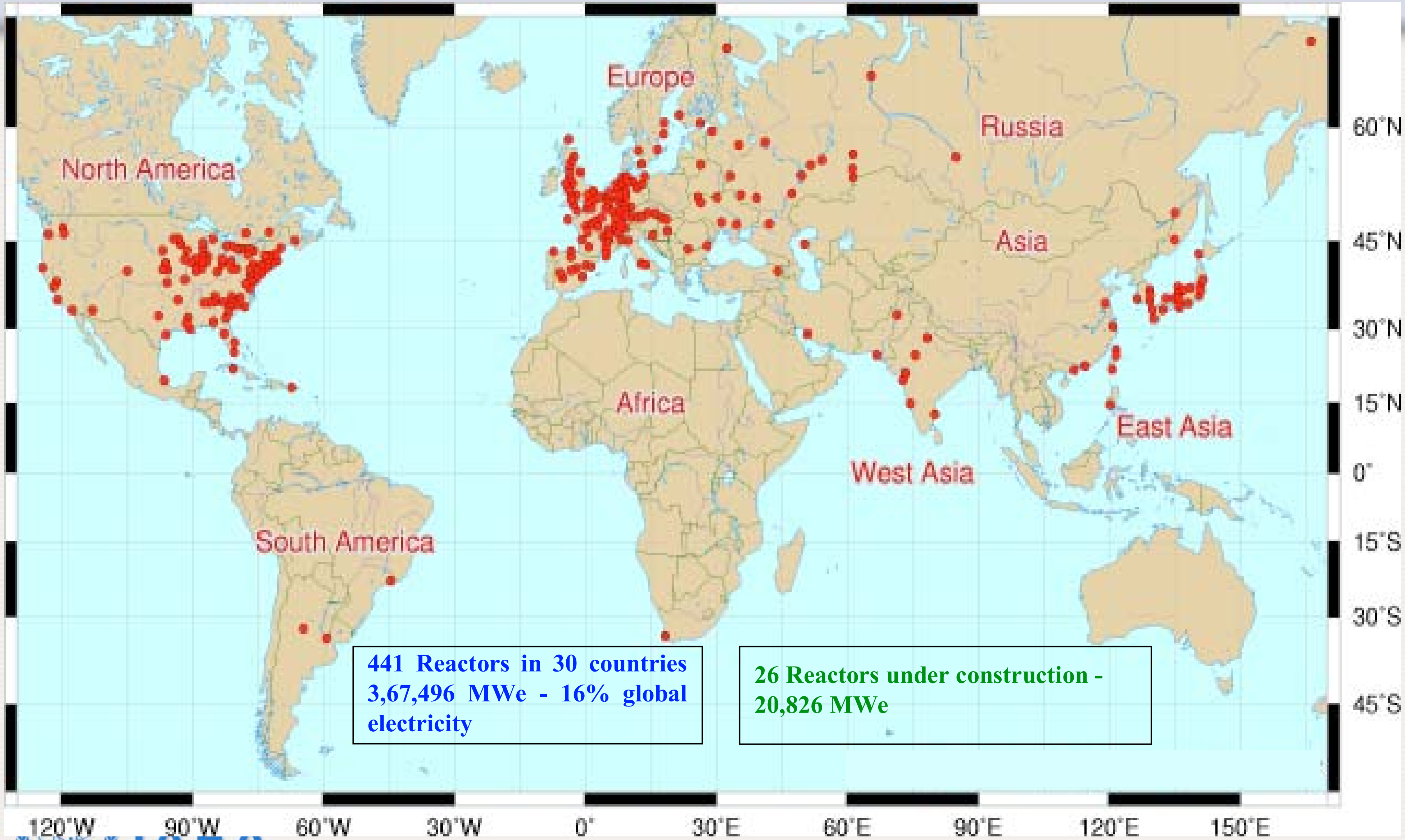
Atoms for Peace
– Civilian Nuclear Power programme

Commercial Phase

- 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

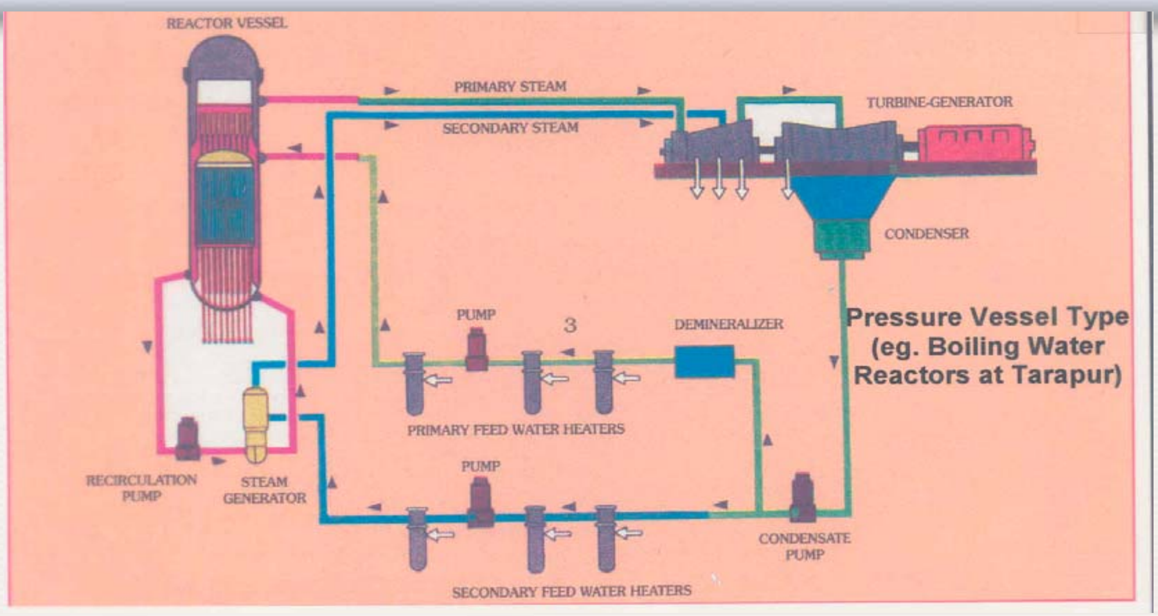
(July 2005)



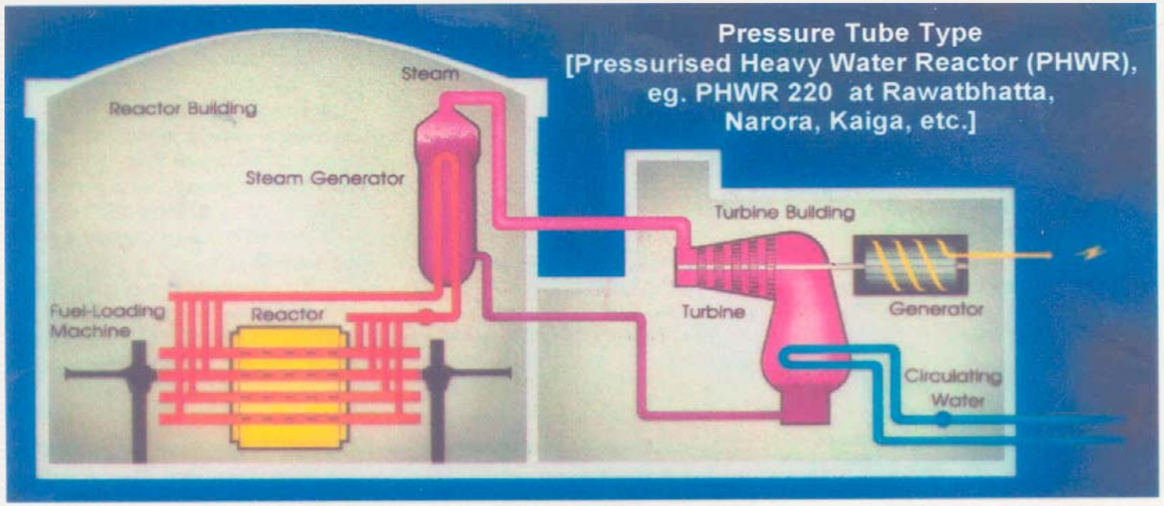
441 Reactors in 30 countries 3,67,496 MWe - 16% global electricity

26 Reactors under construction - 20,826 MWe

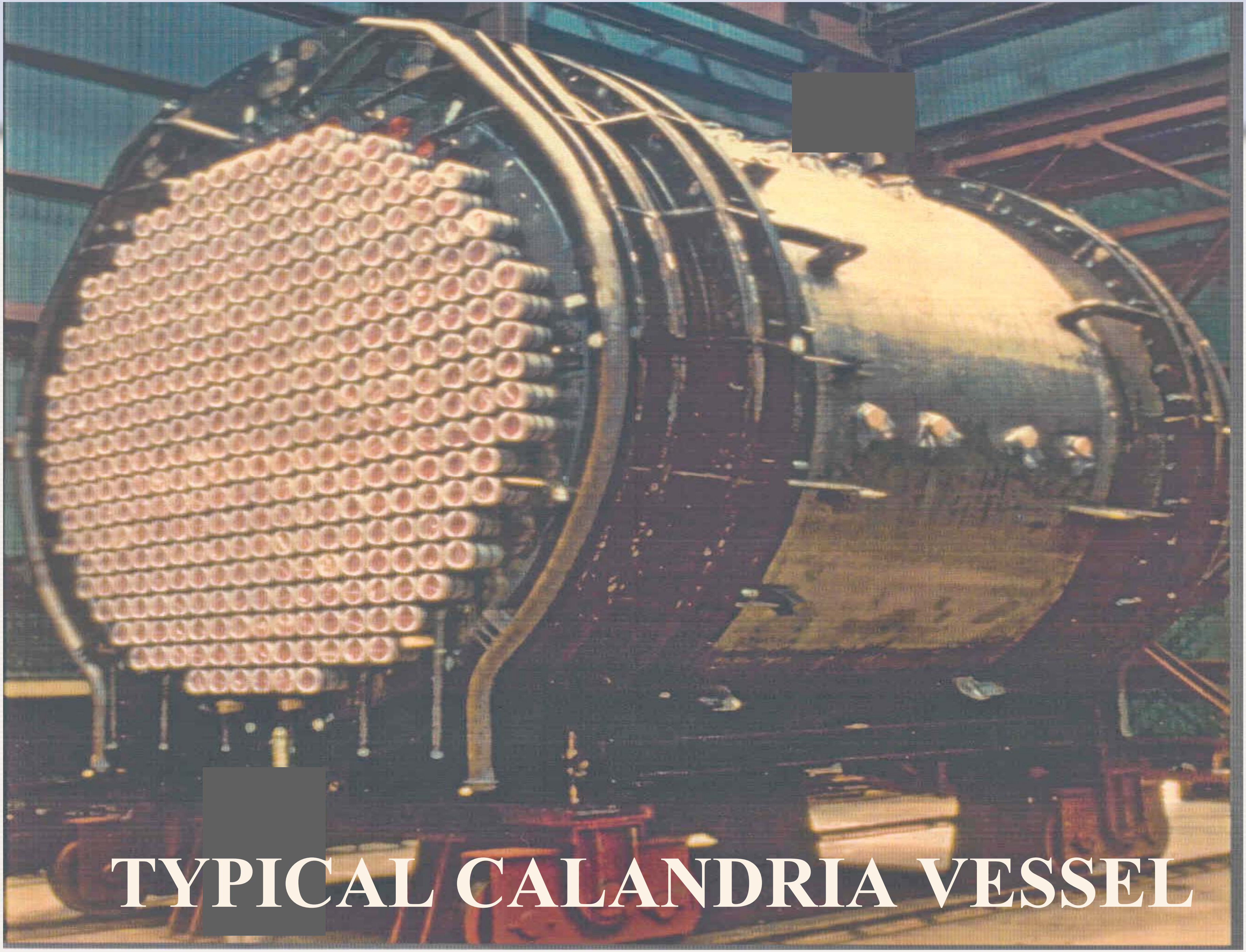
Pressure Vessel and Pressure Tube Type of Nuclear Power Reactors



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.]



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TYPICAL CALANDRIA VESSEL

NUCLEAR POWER REACTORS

- **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%**

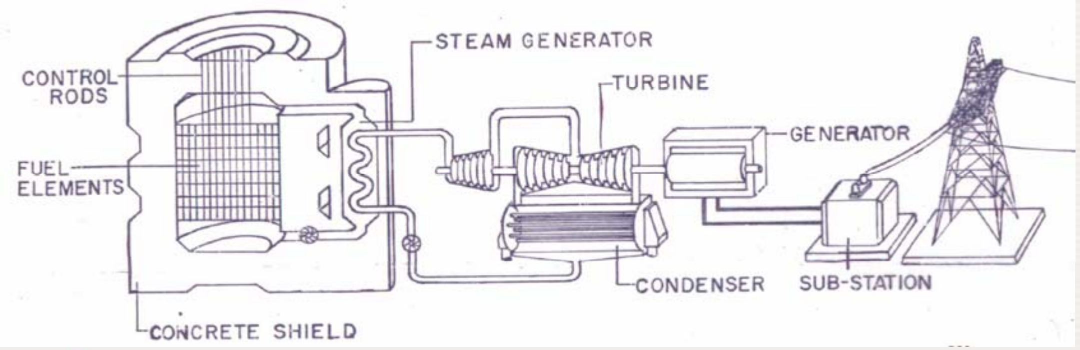
  Russia and CIS - known as RBMK type reactor.

- **Gas Cooled Graohite 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)



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REACTORS IN OPERATION
441 Nos. 3 67 496 MWe

Civilian Nuclear Power Programme in the 21st Century:

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

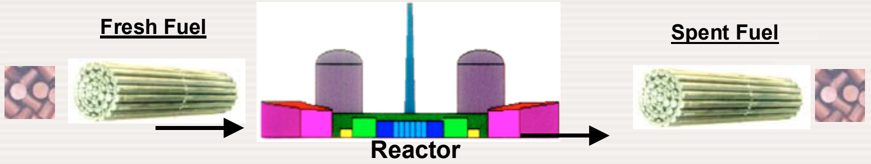
- **Environment Friendly**:

Zero emission of CO_2, SO_2 & NO_x – 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 CO_2

CO_2 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.



$$U^{235}, Pu^{239} \text{ or } U^{233} + {}_0n^1 \longrightarrow \text{Fission Products} + 3\,{}_0n^1 + 200\text{ MeV}$$

$$+\ U^{238} + {}_0n^1 \longrightarrow U^{239} \xrightarrow{\beta^-} Np^{239} \xrightarrow{\beta^-} Pu^{239}$$

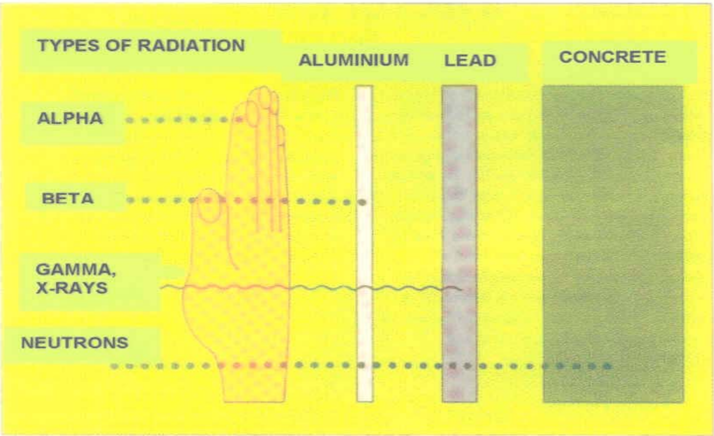
or

$$+\ Th^{232} + {}_0n^1 \longrightarrow Th^{233} \xrightarrow{\beta^-} Pa^{233} \xrightarrow{\beta^-} U^{233}$$

- **High Safety & Security**

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^{238}), U^{233} (always associated with U^{232}) 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.

Comparitive 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)

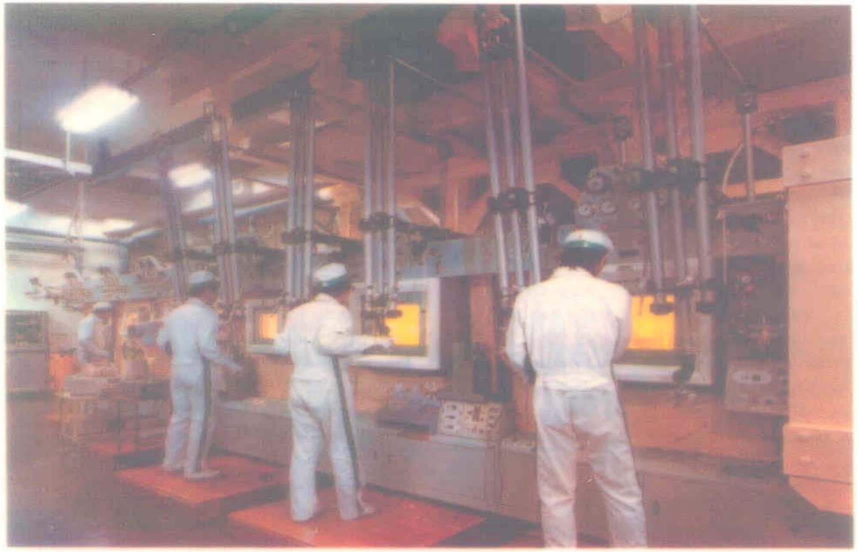
Can be stopped by a sheet of aluminium a few mm in thickness.

Gamma & X Rays: (hazardous–requires proper shielding)

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.

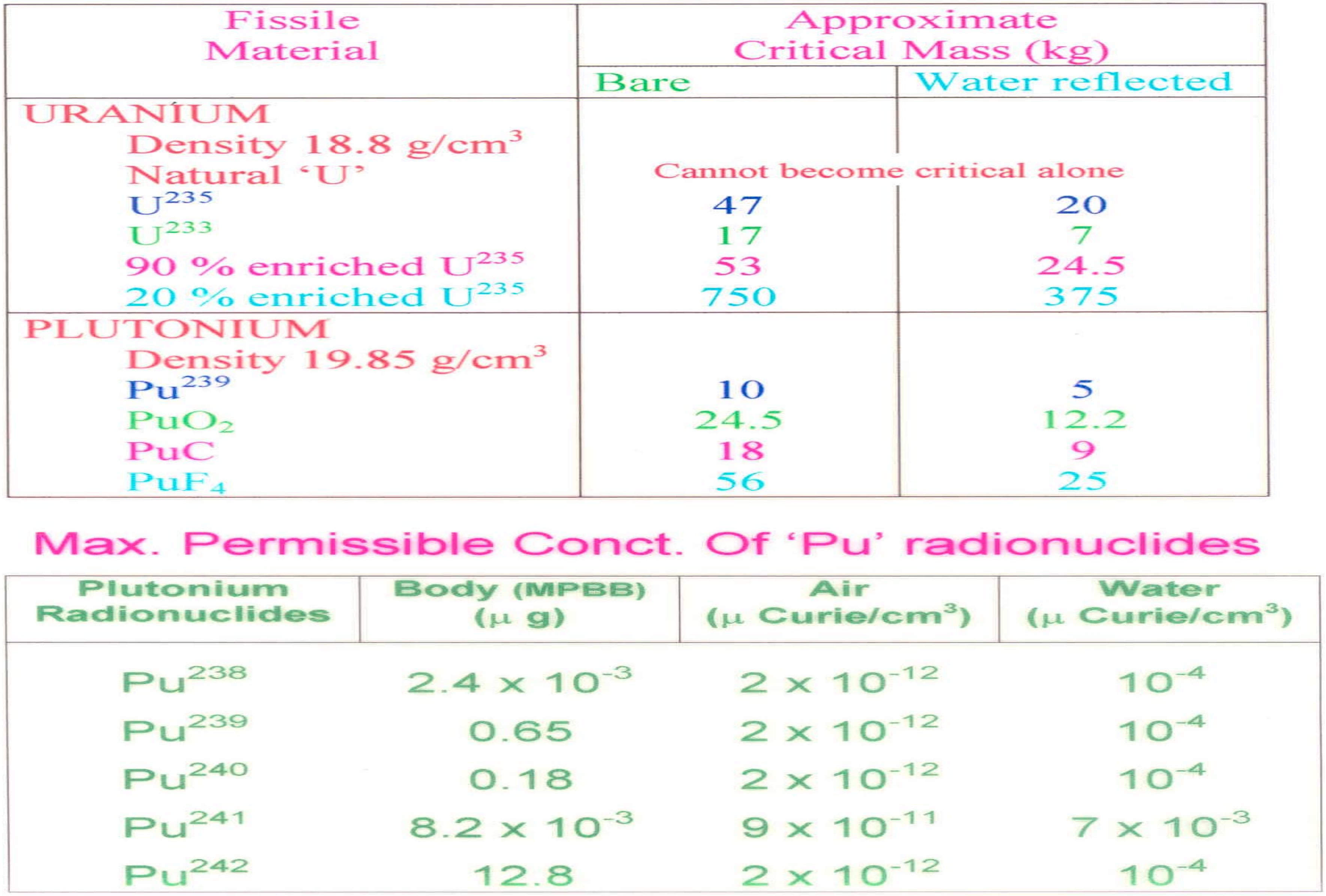
Neutrons:(hazardous – requires proper shielding)

Are very penetrating – in general, efficient shielding against neutron can be provided by water, perspex, etc. Concrete shielding (sometimes upto 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



Fissile Material	Approximate Critical Mass (kg)	
	Bare	Water reflected
URANIUM Density 18.8 g/cm^3		
Natural 'U'	Cannot become critical alone	
U^{235}	47	20
U^{233}	17	7
90 % enriched U^{235}	53	24.5
20 % enriched U^{235}	750	375
PLUTONIUM Density 19.85 g/cm^3		
Pu^{239}	10	5
PuO_2	24.5	12.2
PuC	18	9
PuF_4	56	25

Max. Permissible Conct. Of 'Pu' radionuclides

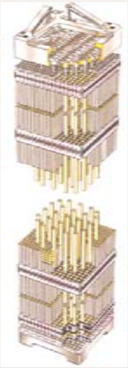
Plutonium Radionuclides	Body (MPBB) (μ g)	Air (μ Curie/cm^3)	Water (μ Curie/cm^3)
Pu^{238}	2.4×10^{-3}	2×10^{-12}	10^{-4}
Pu^{239}	0.65	2×10^{-12}	10^{-4}
Pu^{240}	0.18	2×10^{-12}	10^{-4}
Pu^{241}	8.2×10^{-3}	9×10^{-11}	7×10^{-3}
Pu^{242}	12.8	2×10^{-12}	10^{-4}

MAJOR NUCLEAR FUEL FORMS

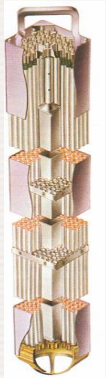
A: Pellet-pin

37 Fuel Elements
CANDU/PHWR
Zircaloy clad, Natural
UO_2 Fuel

Ceramic Fuel Pellets



17x7
PWR



9x9
BWR

Zircaloy clad
Slightly Enriched Uranium
$(<5\%U^{235})UO_2$

312 Fuel
Elements

VVER-1000
Zr-1%Nb clad
Sightly Enriched Uranium
$(<5\%U^{235})UO_2$

Fast Breeder Reactor FUEL
Stainless Steel clad, $(U, Pu)O_2$
(20-25% $Pu\,O_2$)

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

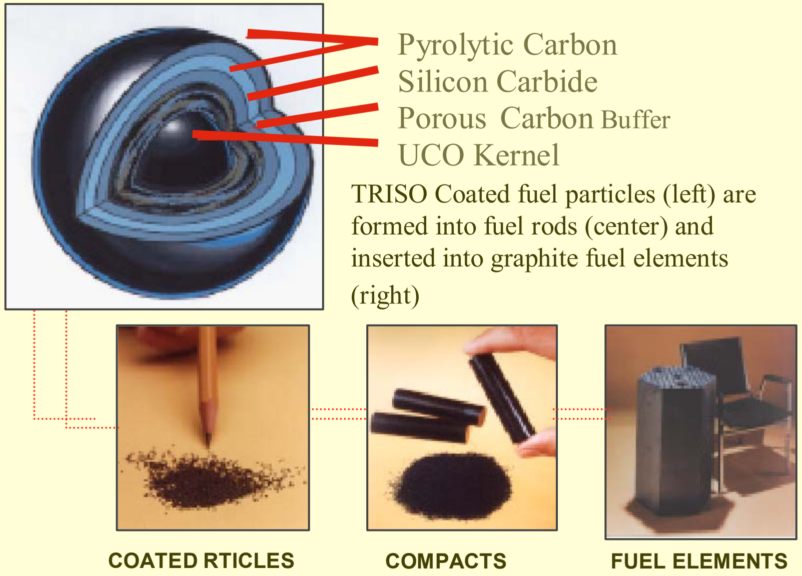
FUEL FORMS

B: Coated Fuel Particles

1. Coated fuel particles for HTGR

Prismatic block

US, Japan, Russia and France



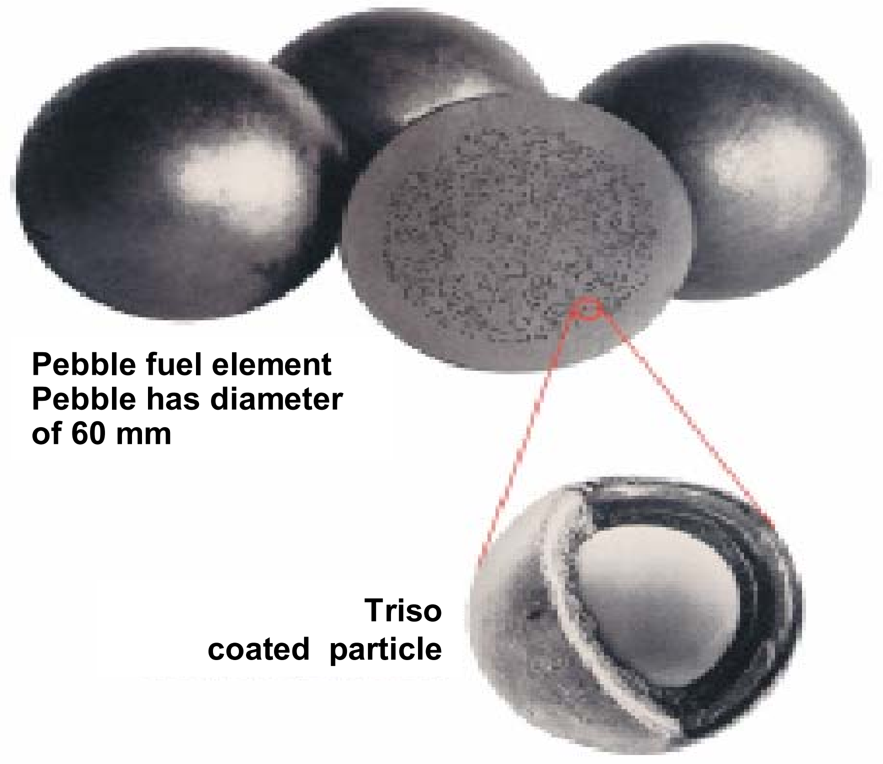
TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right)

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

Pebble Bed

coated particle fuels embedded in spherical shape

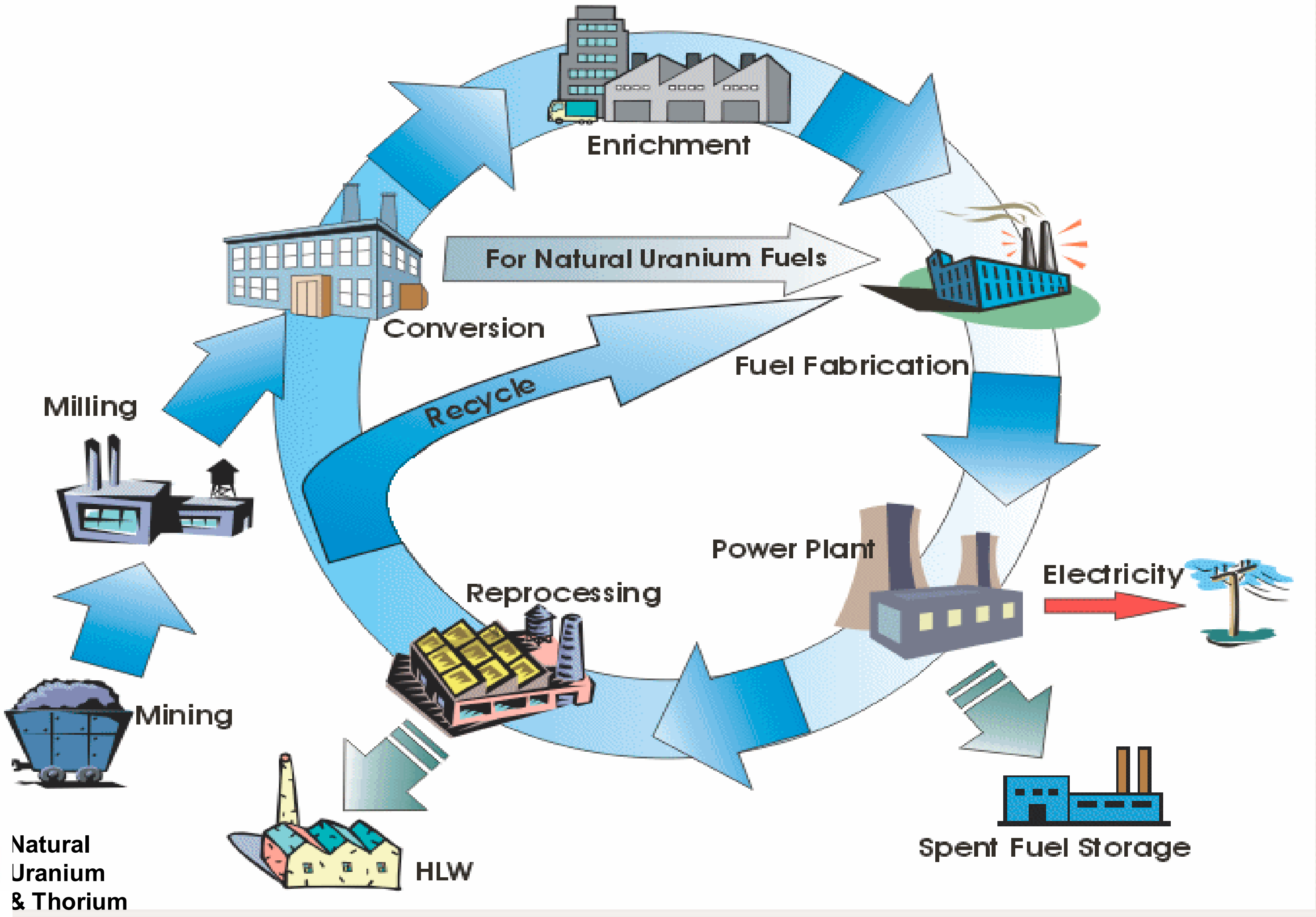
Germany, South Africa, China



Pebble fuel element
Pebble has diameter of 60 mm

Triso coated particle

NUCLEAR FUEL CYCLE



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

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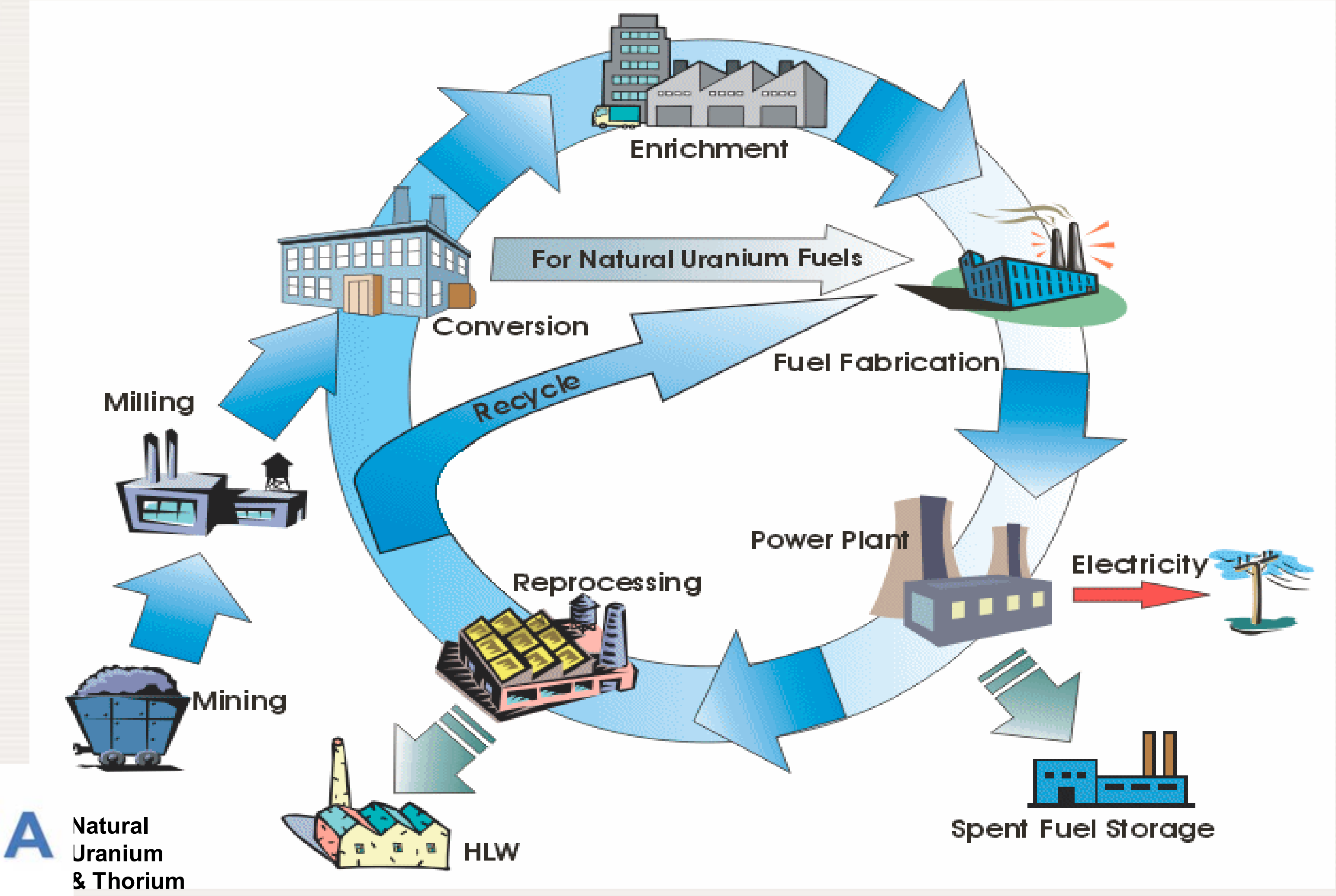
IAEA
International Atomic Energy Agency

NUCLEAR FUEL CYCLE

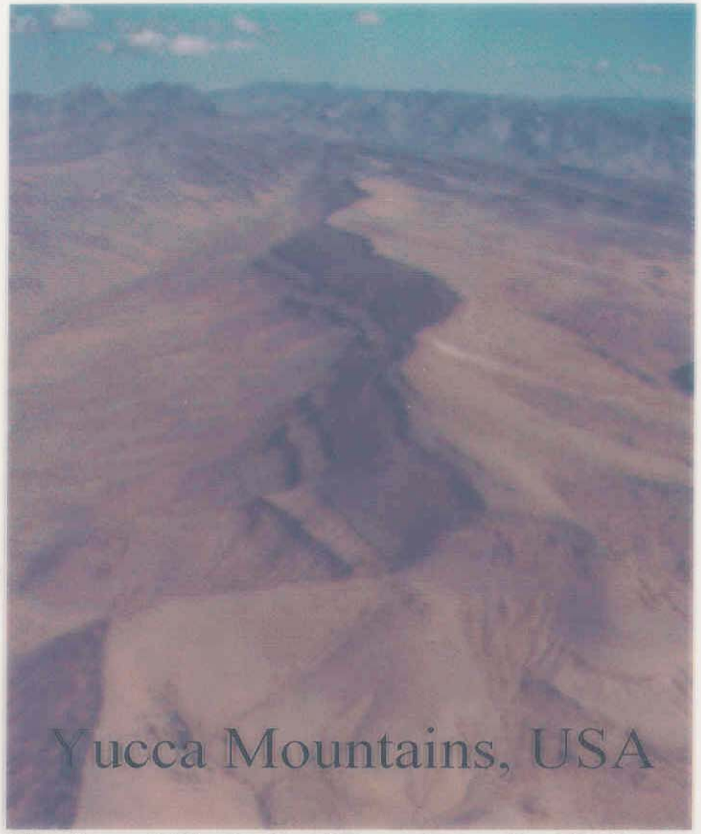


For nuclear energy to be sustainable as a global source of emission – free energy, the reactor fuel cycle must also remain sustainable (DG-IAEA Scientific Forum 2004)

MAJOR CHALLENGE IN NUCLEAR FUEL CYCLE

Develop a fuel cycle that is economically viable, environmentally benign, proliferation-resistant, safe and sustainable

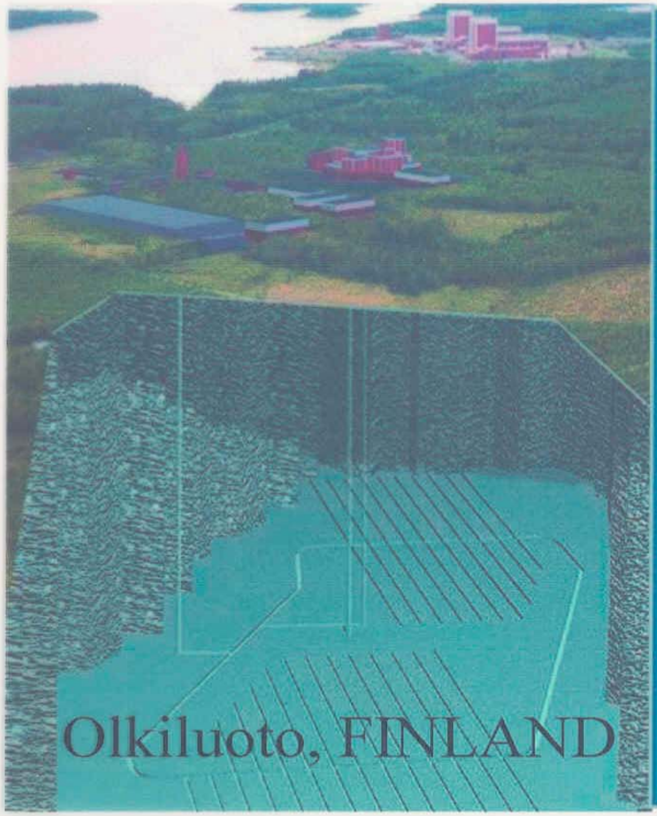
Permanent Repositories for Safe Disposal of Nuclear Wastes

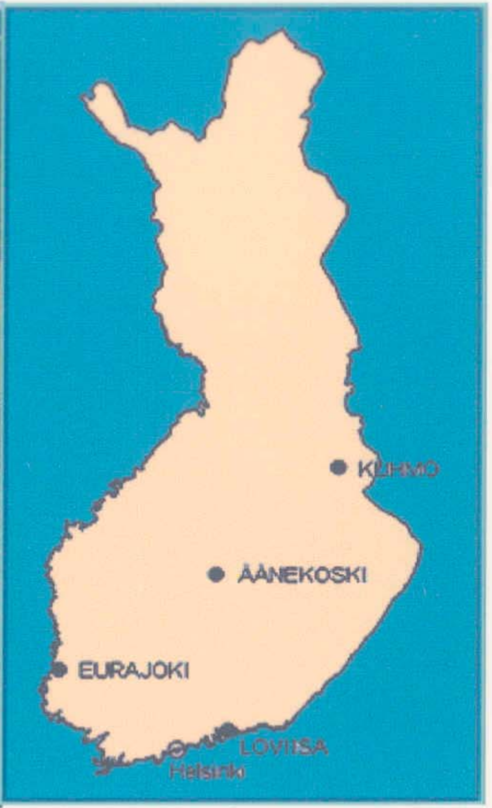


Monitored Geological Repository for High Level Radioactive Waste at Yucca Mountains, Nevada

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





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.

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.

ACTIVITIES (PROGRAMME B) OF NUCLEAR FUEL CYCLE & MAT0ERIALS SECTION (NFC&MS), IAEA

B1: Uranium Production Cycle and Environment

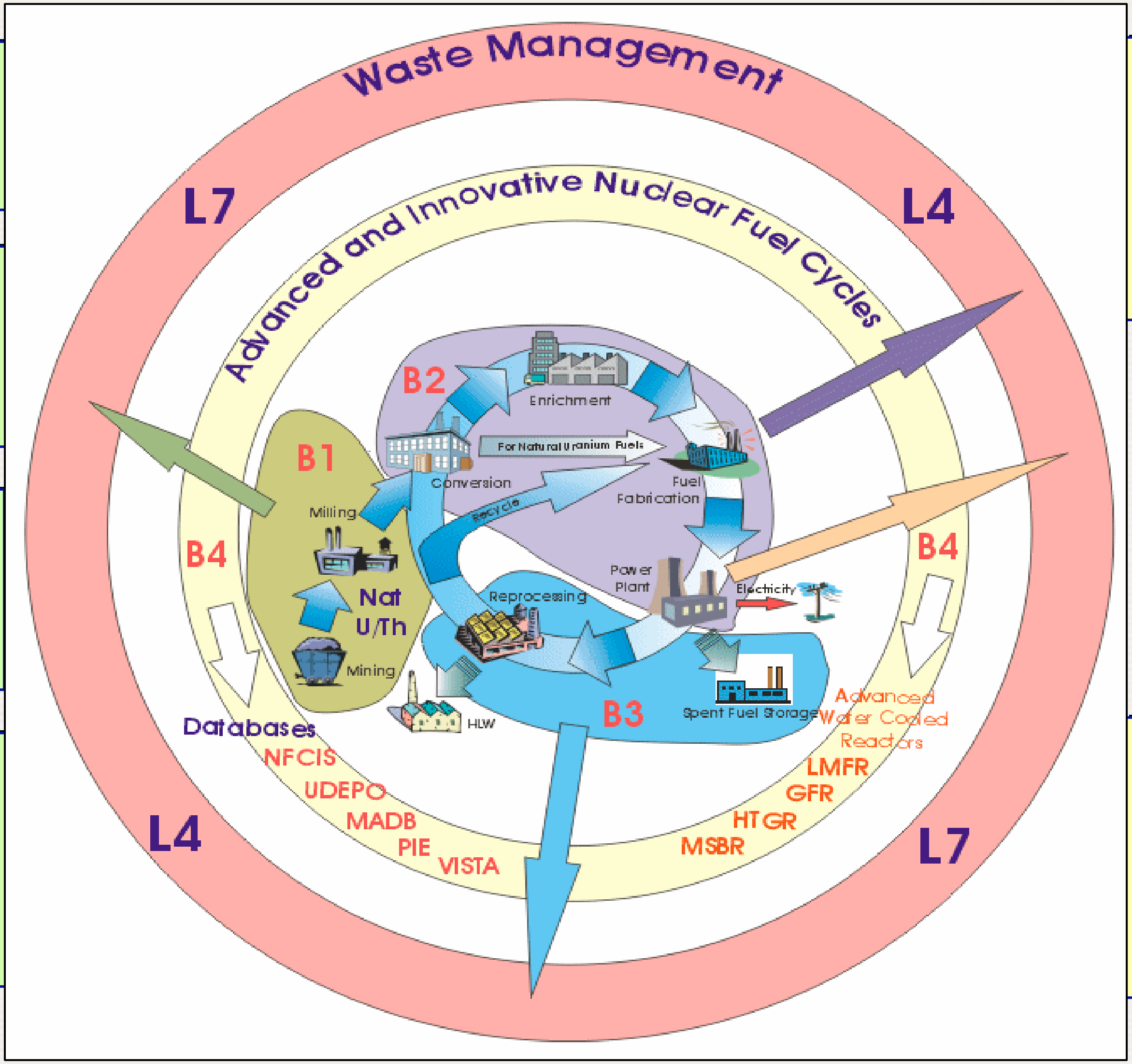
B2: Water-Cooled Fuel Performance and Technology

(Database: PIE)

B3: Management of Spent Fuel from Power Reactors

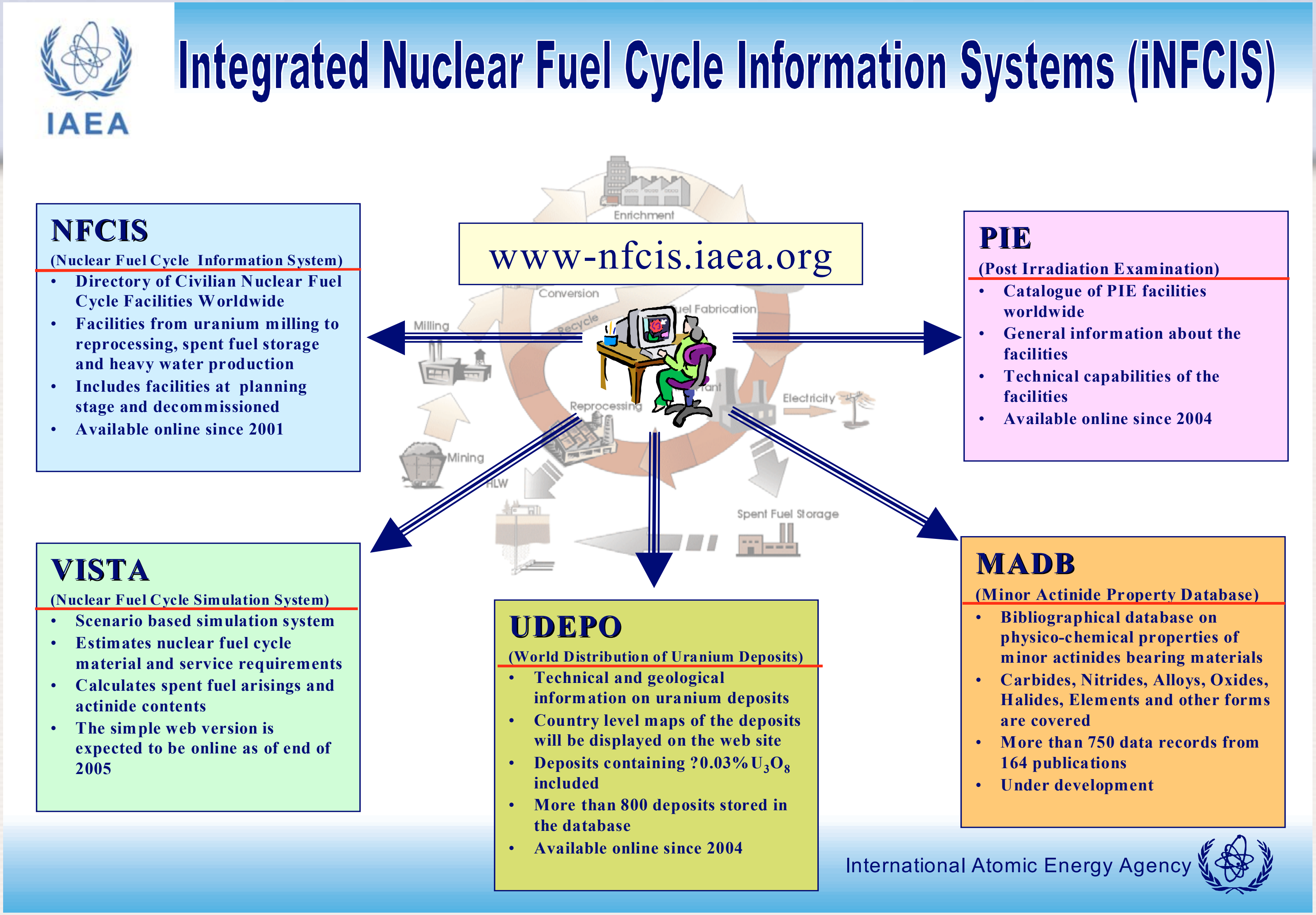
B4: Topical Nuclear Fuel Cycle Issues and Information Systems

(Databases: NFCIS, UDEPO, MADB, VISTA)



L4: Technologies for Disposable Radioactive Waste (Pre-disposal & Disposal)

L7: Technologies for the Decommissioning of Installations and Restoration of Sites



Integrated Nuclear Fuel Cycle Information Systems (iNFCIS)

www-nfcis.iaea.org

NFCIS

(Nuclear Fuel Cycle Information System)

- Directory of Civilian Nuclear Fuel Cycle Facilities Worldwide
- Facilities from uranium milling to reprocessing, spent fuel storage and heavy water production
- Includes facilities at planning stage and decommissioned
- Available online since 2001

PIE

(Post Irradiation Examination)

- Catalogue of PIE facilities worldwide
- General information about the facilities
- Technical capabilities of the facilities
- Available online since 2004

VISTA

(Nuclear Fuel Cycle Simulation System)

- Scenario based simulation system
- Estimates nuclear fuel cycle material and service requirements
- Calculates spent fuel arisings and actinide contents
- The simple web version is expected to be online as of end of 2005

UDEPO

(World Distribution of Uranium Deposits)

- Technical and geological information on uranium deposits
- Country level maps of the deposits will be displayed on the web site
- Deposits containing ?0.03% U_3O_8 included
- More than 800 deposits stored in the database
- Available online since 2004

MADB

(Minor Actinide Property Database)

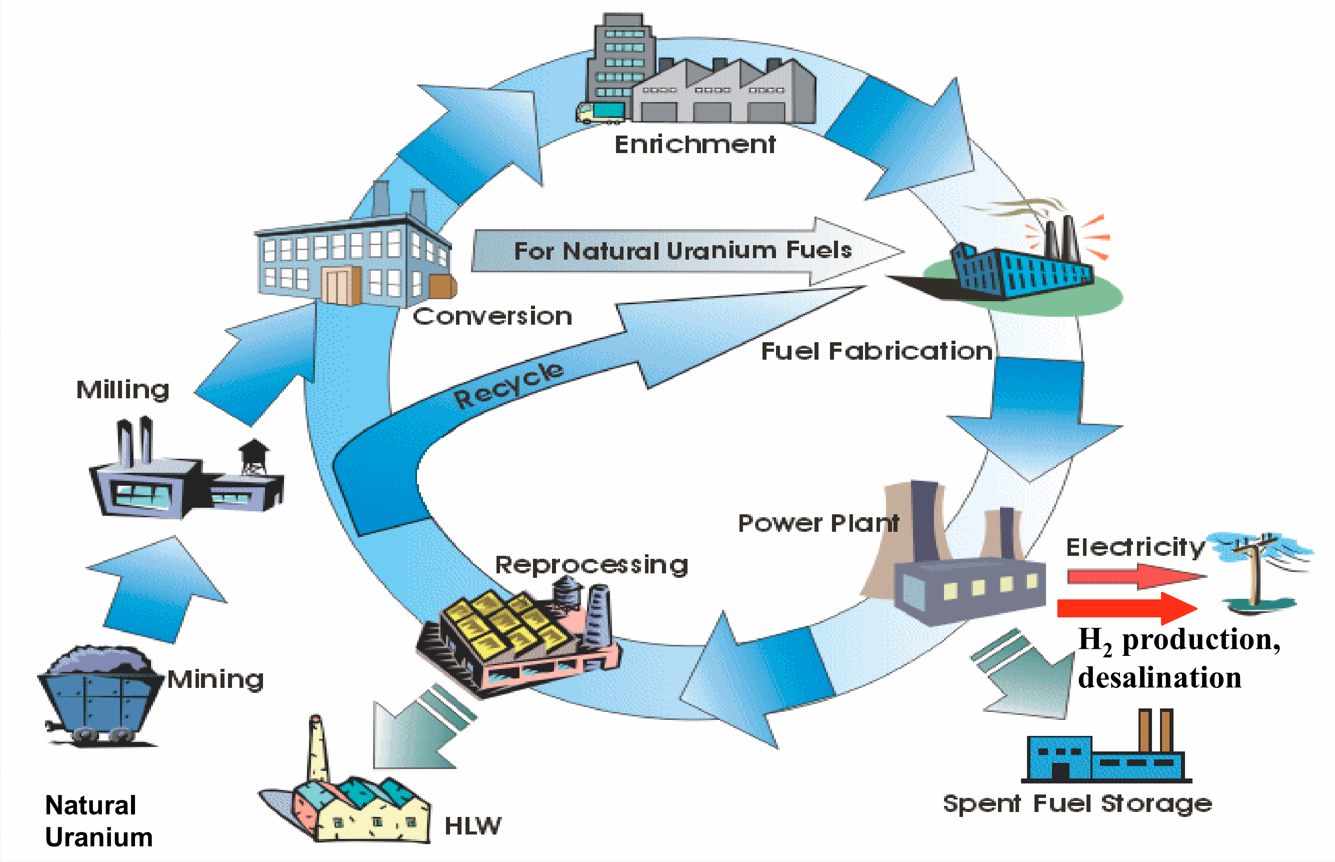
- Bibliographical database on physico-chemical properties of minor actinides bearing materials
- Carbides, Nitrides, Alloys, Oxides, Halides, Elements and other forms are covered
- More than 750 data records from 164 publications
- Under development

International Atomic Energy Agency

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: 40,251tons ‘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’]

(Ref: IAEA Red Book Dec. 2003)

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

India	40,980 (>130$/kg)	14 (3%)

Uranium Resources Worldwide

(Source: Uranium 2003 – 20th Edition of IAEA Red Book jointly with OECD-NEA)

1. Conventional Resources

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

a) Conventional Resources (RAR and EAR-1):
≤ US $ 80 / kg U : 3,537,000 tons; ≤ US $ 130 / kg U: 4,589,000 tons U

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

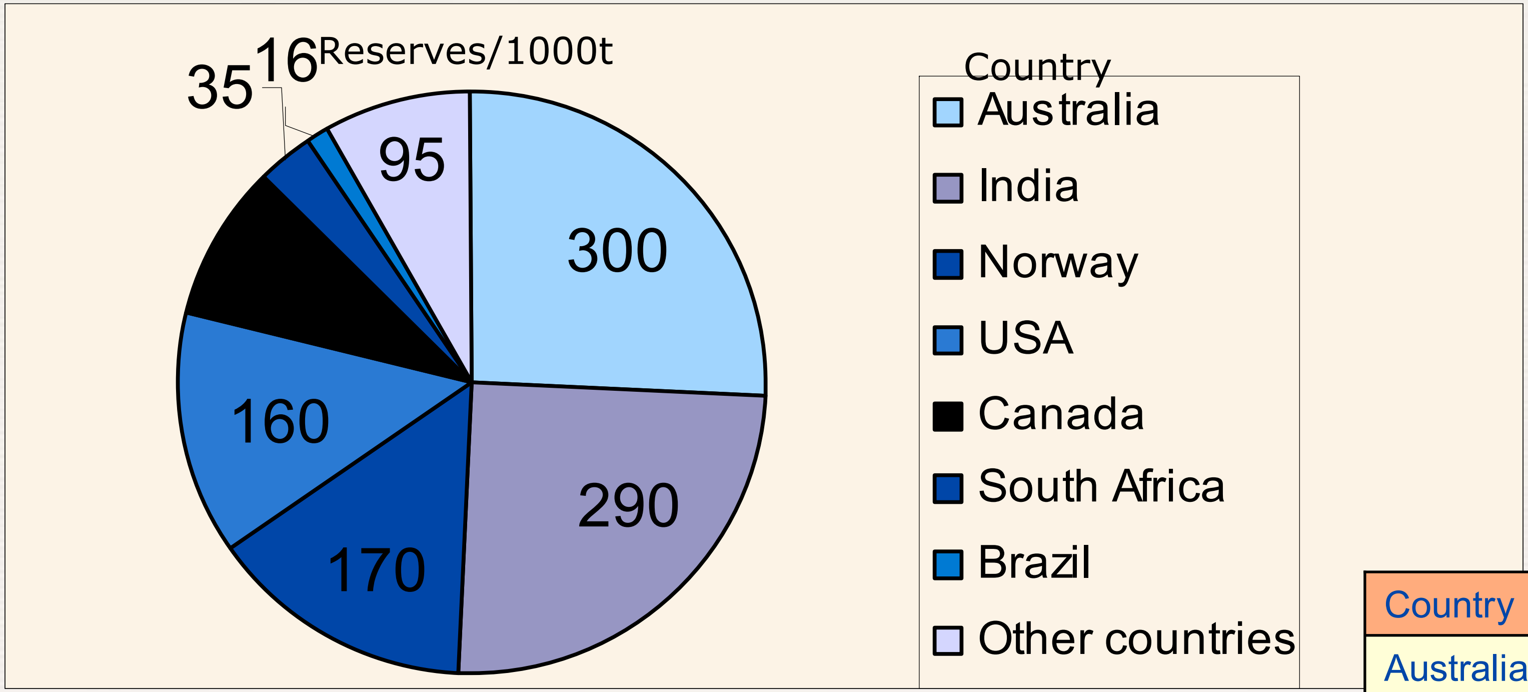
Total Conventional Resource (a + b) : 14,383,000 tons U

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

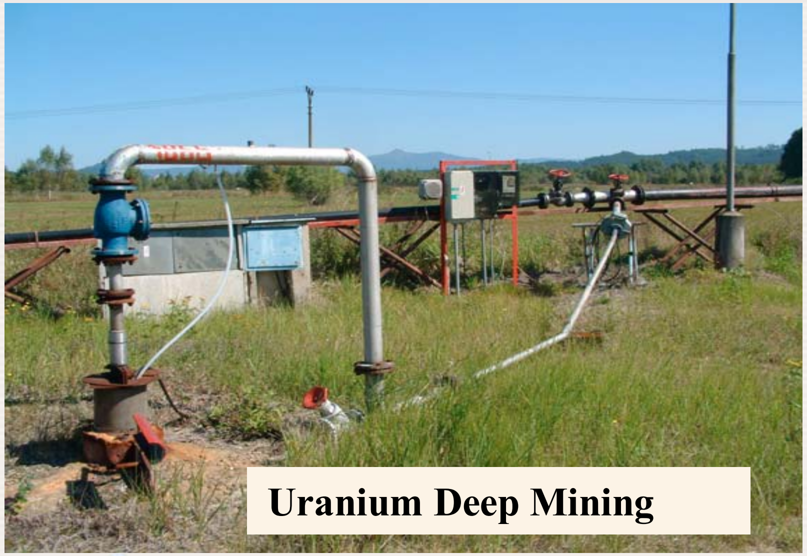


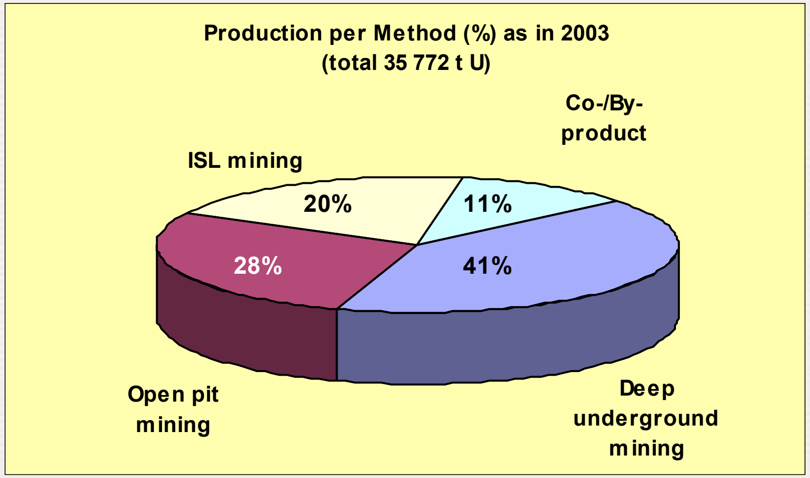
Country	Reserves / t
Australia	300 000
India	290 000
Norway	170 000
USA	160 000
Canada	100 000
South Africa	35 000
Brazil	16 000
Other Countries	95 000

URANIUM MINING TRENDS


Rossing Open Cast Mine


Mining Techniques


Uranium Deep Mining



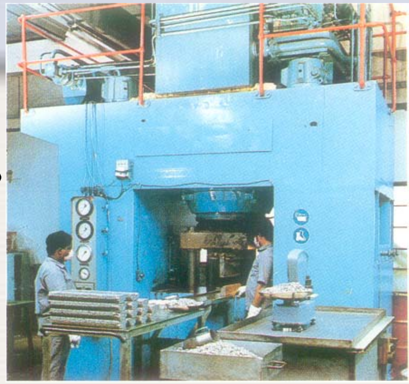
IAEA

FROM ZIRCON SAND TO ZIRCONIUM ALLOY INGOTS AT NFC, HYDERABAD

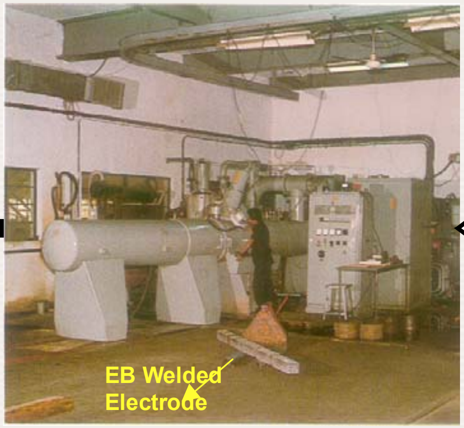
Zircon Sand

Hf-free ZrO_2 Powder

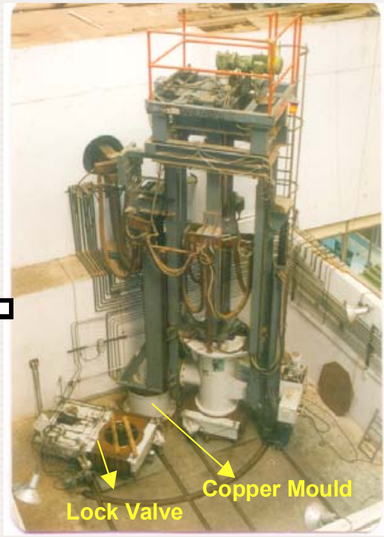
Nuclear Grade Zr Sponge



Compaction of Zr Sponge + alloying elements Briquettes



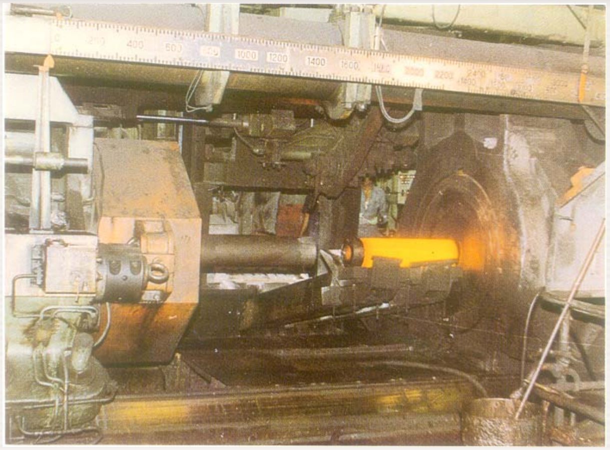
Electron Beam Welding of Briquettes to form Consumable Electrode



Vacuum Arc Melting Furnace using Consumable Electrode

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

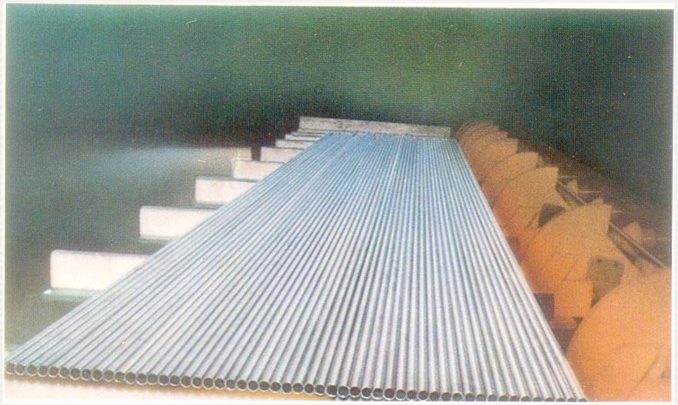
Major Activities of Zirconium Alloy Fabrication Plant at NFC



Hot Extrusion of
Seamless Zirconium Alloy Tubes

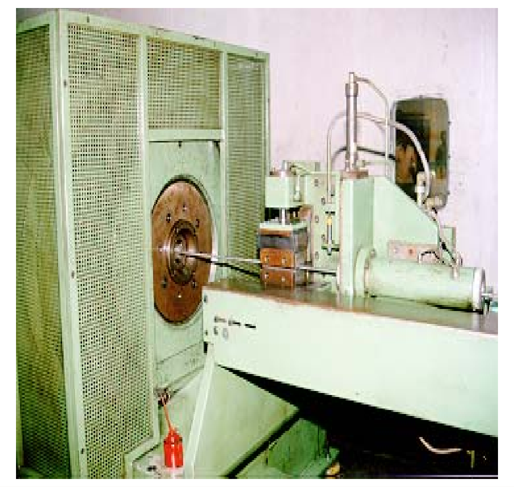
Pilot Hole Expansion Press

Pilger Mill for Production of
Zirconium Alloy Fuel Tubes



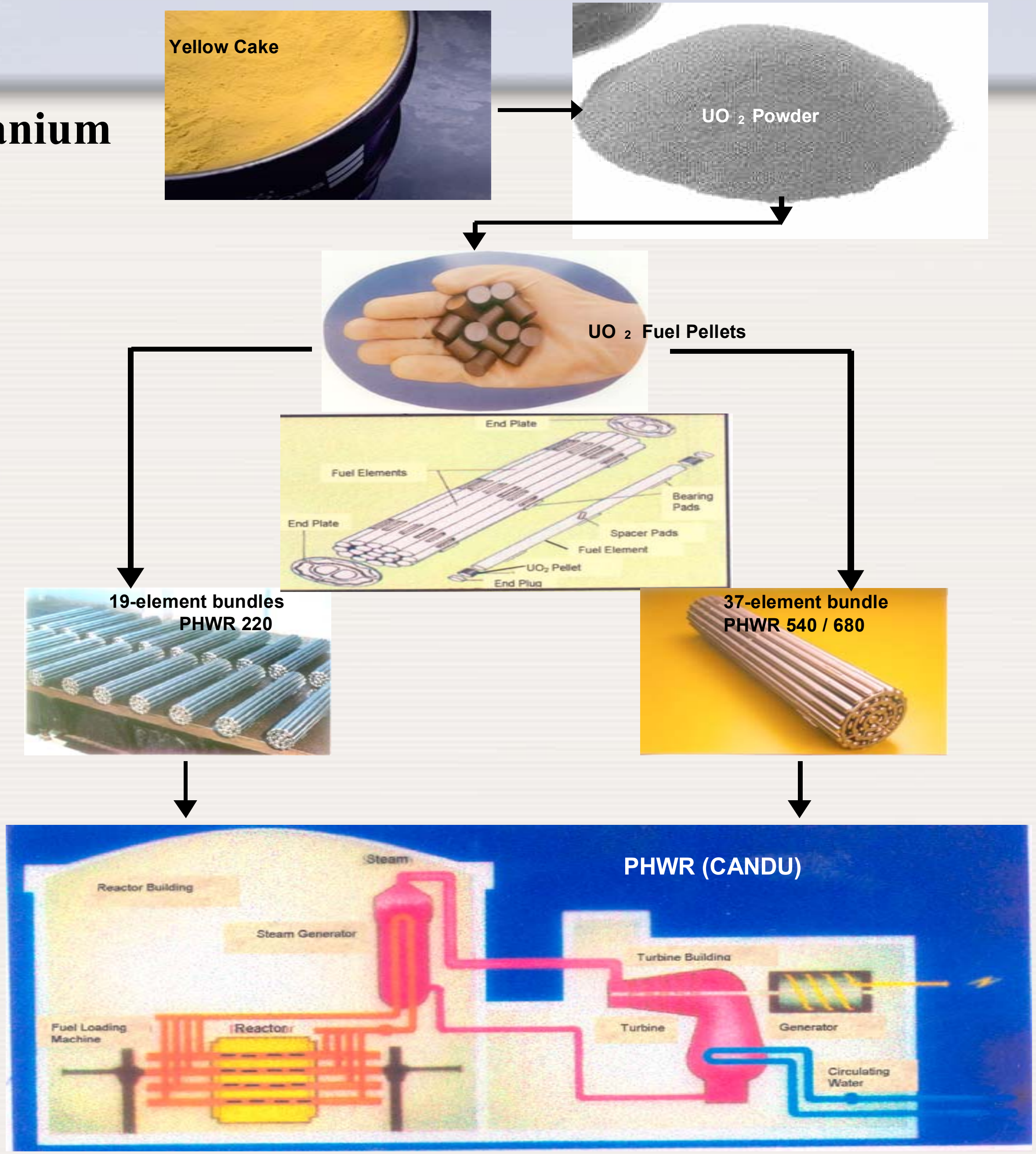
Array of Zirconium Alloy Fuel Tubes

Hot Rolling of Zirconium Alloy Sheets



Cold Swaging of Zirconium Alloy Bar

Production of Natural Uranium Oxide Fuel Bundles for PHWR



Yellow Cake

UO_2 Powder

UO_2 Fuel Pellets

19-element bundles
PHWR 220

37-element bundle
PHWR 540 / 680

PHWR (CANDU)

FUEL ASSEMBLY FOR
540 MWe PRESSURISED HEAVY WATER REACTORS

(**102.36** ϕ x 495.3 l)

➢ 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 ϕx 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 ϕx 1.60 t)	….. 2 Nos

- **NATURAL URANIUM DIOXIDE**

PELLETS	(12.20 ϕ x 13.29 l)	….. 1295 Nos.
WEIGHT OF PELLETS		….. 21.65 kg

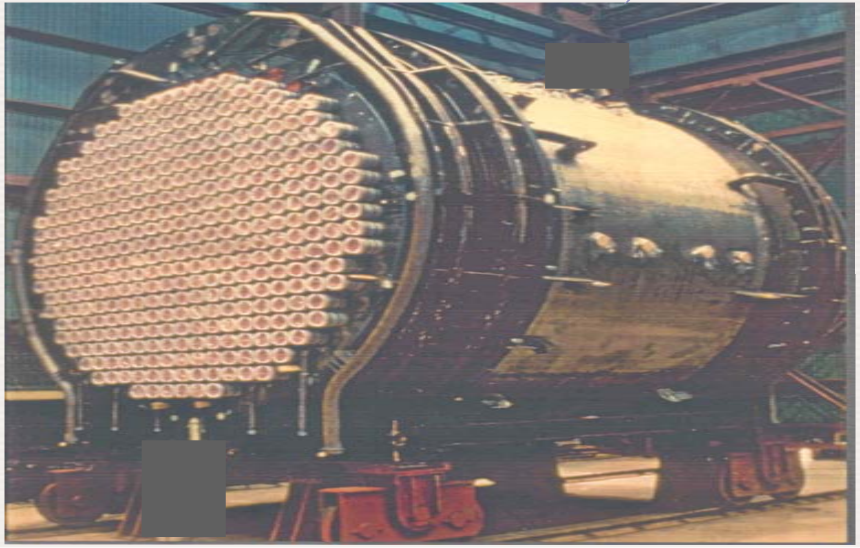
➢ ONE PHWR 540 CORE CONTAINS

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

EIECTRIC POWER FROM
ONE FUEL ASSEMBLY …. 10,44,000 units

DIMENSIONS IN MM

A typical Calandria Vessel for CANDU-PHWR 500 Unit (392 horizontal fuel channels - each channel contain 13 nos. of 37 element fuel bundles)



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Main long-lived or parent Radionuclides present in Irradiated Fuel

Radionuclide	Half-life (Year)
Uranium	
U-234	2.46 x 10^{5}
U-235	7.04 x 10^{8}
U-236	2.34 x 10^{7}
U-238	4.47 x 10^{9}
Actinides (α emitters)	
Pu-238	87.7
Pu-239	24,100
Pu-240	6,560
Pu-241	14.35
Pu-242	3.74 x 10^{5}
Np-237	2.14 x 10^{6}
Am-241	432.7
Am-243	7,368
Cm-245	8.5 x 10^{3}
Cm-246	4.73 x 10^{3}
Fission Products (β/γ emitters)	
Se-79	6.5 x 10^{5}
Zr-93	1.5 x 10^{6}
Tc-99	2.13 x 10^{5}
Pd-107	6.5 x 10^{6}
Sn-126	1 x 10^{5}
I-129	1.57 x 10^{7}
Cs-135	2.3 x 10^{6}
Activation Products (β/γ emitters)	
C-14	5,715
Ni-59	7.6 x 10^{4}
Ni-63	100
Zr-93	1.53 x 10^{6}
Nb-94	2.03 x 10^{4}

Closing the fuel cycle facilitates: a) Waste minimization [in terms of Pu & Minor Actinides (MA)]
b) Resource (natural uranium) utilization

Reactor

Fresh unraium fuel (as oxide)

SPENT FUEL

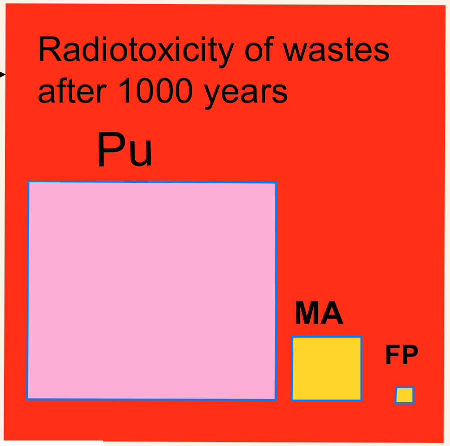
valuable materials
96%

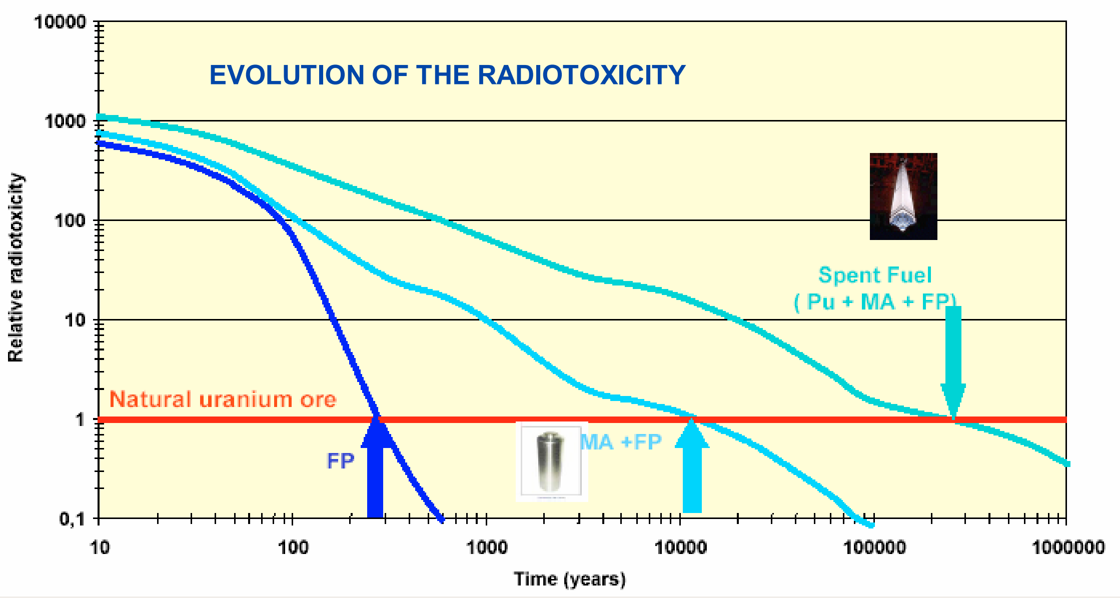
- U: 94-96%
- Pu: ~1%

+

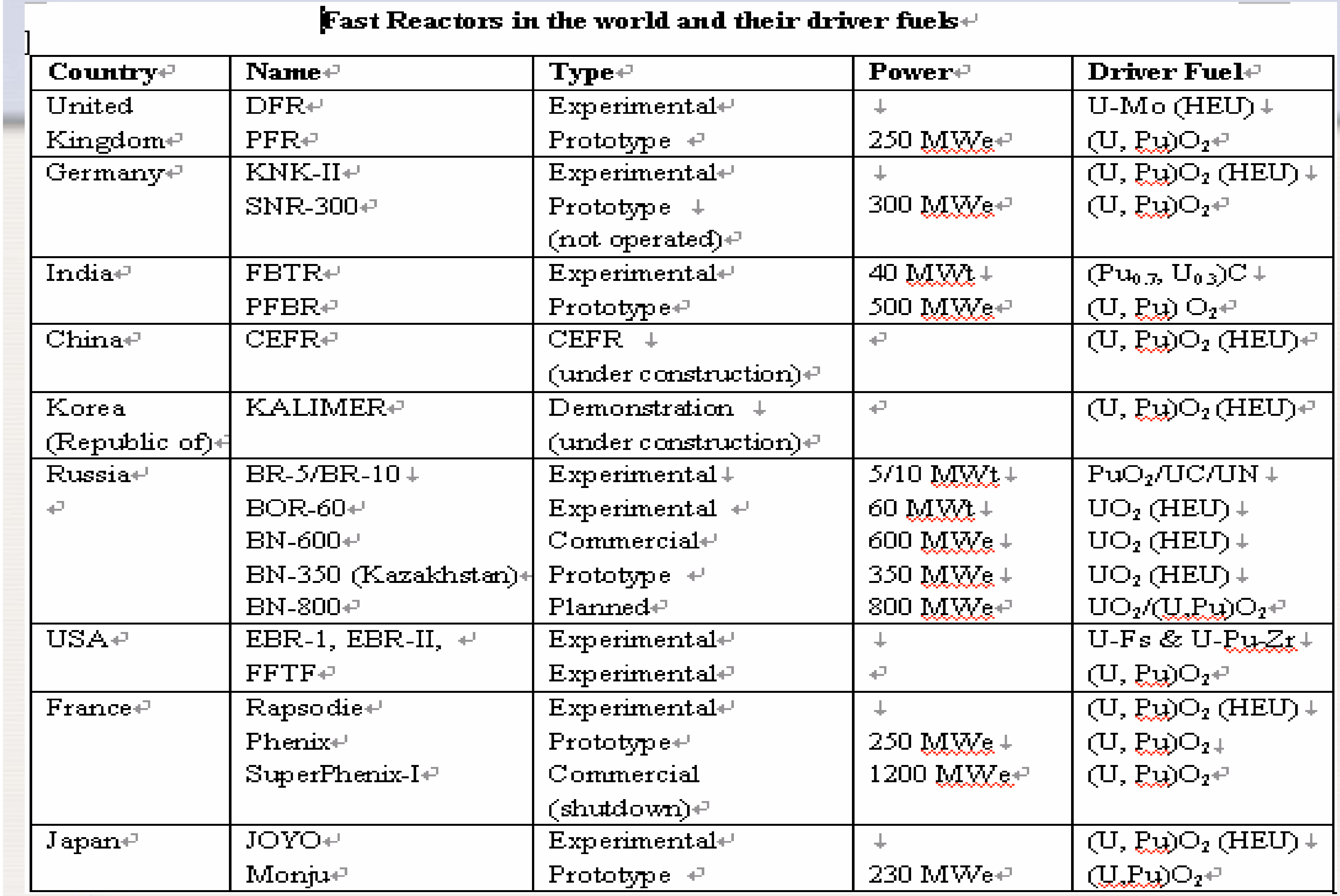
Wastes
4%

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





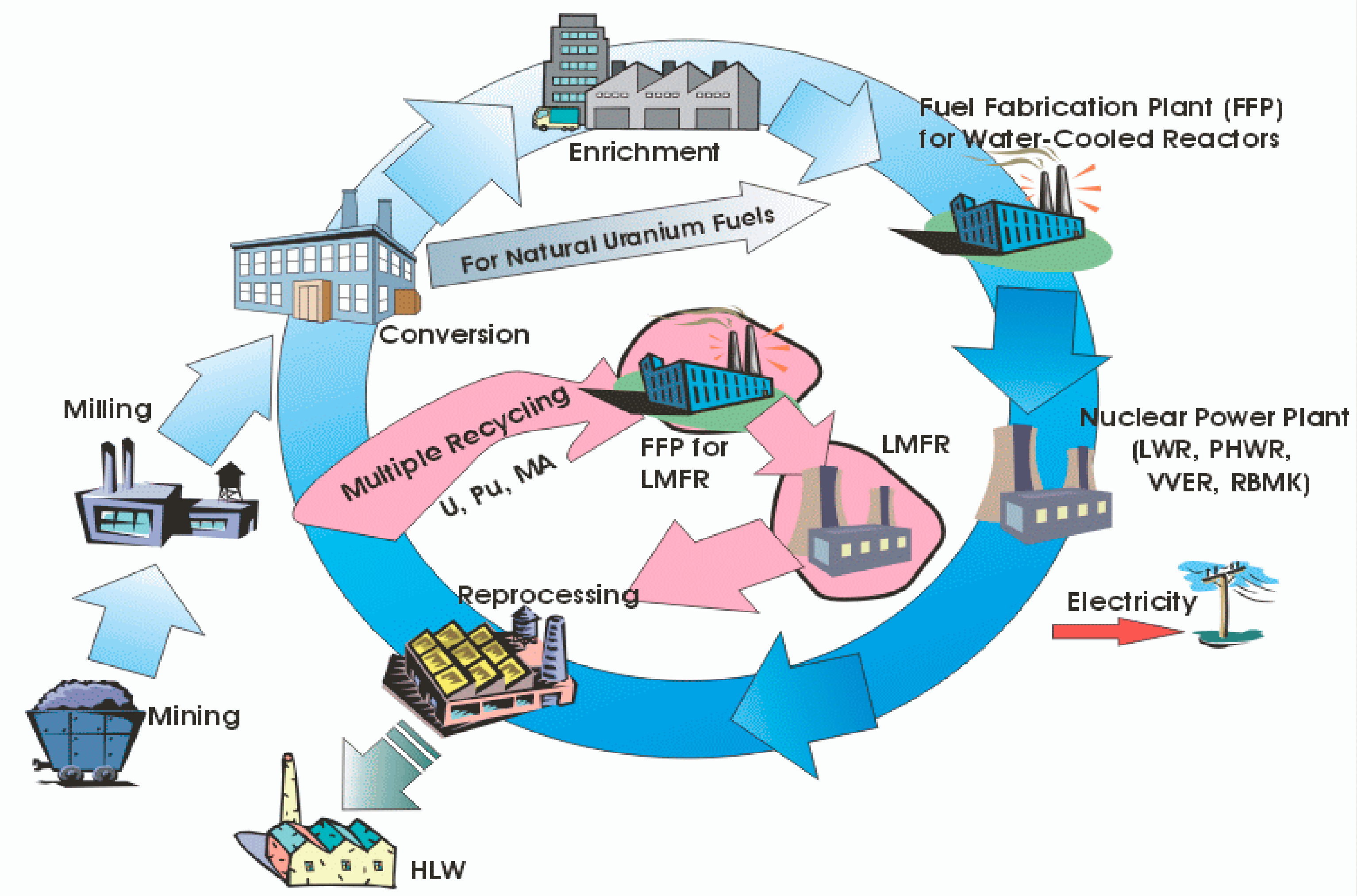
valuable materials
96%



Fast Reactors in the world and their driver fuels

Country	Name	Type	Power	Driver Fuel
United Kingdom	DFR PFR	Experimental Prototype	 250 MWe	U-Mo (HEU) $(U, Pu)O_2$
Germany	KNK-II SNR-300	Experimental Prototype (not operated)	 300 MWe	$(U, Pu)O_2$ (HEU) $(U, Pu)O_2$
India	FBTR PFBR	Experimental Prototype	40 MWt 500 MWe	$(Pu_{0.7}, U_{0.3})C$ $(U, Pu)\ O_2$
China	CEFR	CEFR (under construction)		$(U, Pu)O_2$ (HEU)
Korea (Republic of)	KALIMER	Demonstration (under construction)		$(U, Pu)O_2$ (HEU)
Russia	BR-5/BR-10 BOR-60 BN-600 BN-350 (Kazakhstan) BN-800	Experimental Experimental Commercial Prototype Planned	5/10 MWt 60 MWt 600 MWe 350 MWe 800 MWe	$PuO_2/UC/UN$ UO_2 (HEU) UO_2 (HEU) UO_2 (HEU) $UO_2/(U,Pu)O_2$
USA	EBR-1, EBR-II, FFTF	Experimental Experimental		U-Fs & U-Pu-Zr $(U, Pu)O_2$
France	Rapsodie Phenix SuperPhenix-I	Experimental Prototype Commercial (shutdown)	 250 MWe 1200 MWe	$(U, Pu)O_2$ (HEU) $(U, Pu)O_2$ $(U, Pu)O_2$
Japan	JOYO Monju	Experimental Prototype	 230 MWe	$(U, Pu)O_2$ (HEU) $(U,Pu)O_2$

Liquid Metal-cooled Fast Reactor Fuel Cycle with multiple recycling of U, Pu and Minor Actinides



ICTP Workshop
Lecture 3
Conventional & AdvancedFuels (including inert matrix fuels)

Chaitanyamoy Ganguly

Trieste, 16 November , 2005

Chaitanyamoy GANGULY

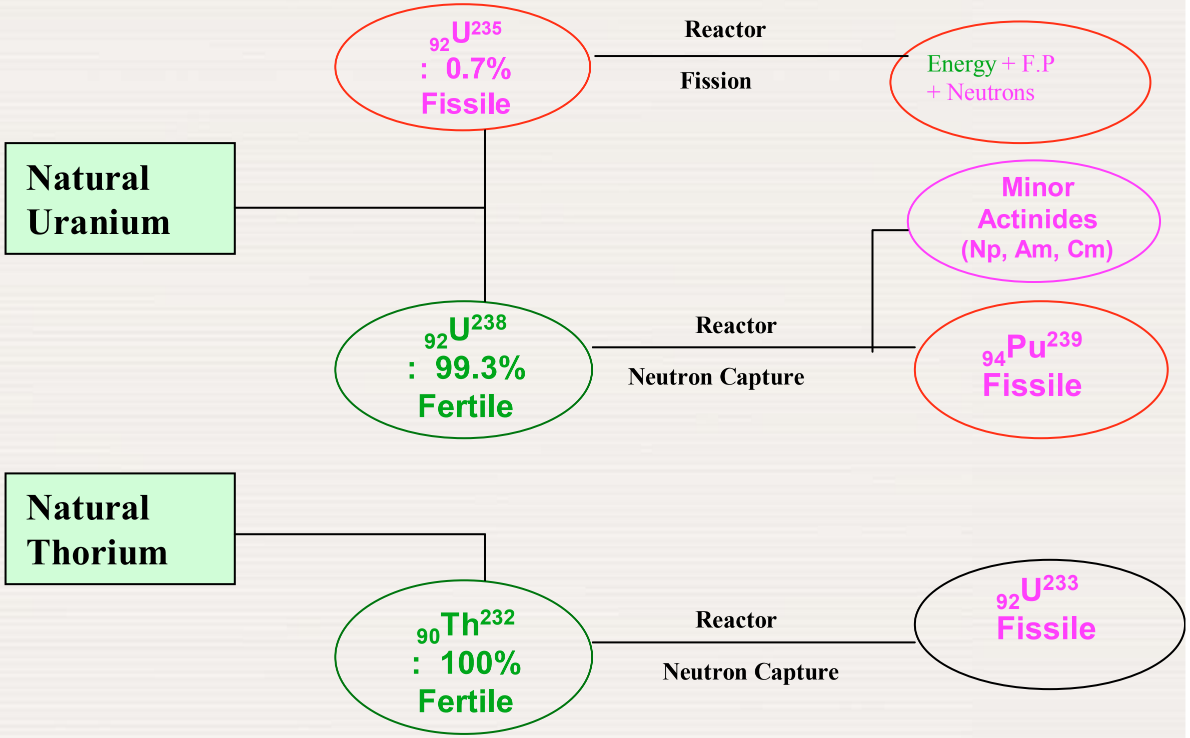
Head, Nuclear Fuel Cycle & Materials Section

IAEA
International Atomic Energy Agency

Natural Resources for Nuclear Fuel, Fissile and Fertile Materials

NUCLEAR FUELS are:

- made of fissile isotopes U^{235}, Pu^{239} or U^{233} mostly with judicious combination of fertile isotopes U^{238} or Th^{232}.
- 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).



Natural Uranium

$_{92}U^{235}$: 0.7% Fissile

Reactor Fission

Energy + F.P + Neutrons

$_{92}U^{238}$: 99.3% Fertile

Reactor Neutron Capture

Minor Actinides (Np, Am, Cm)

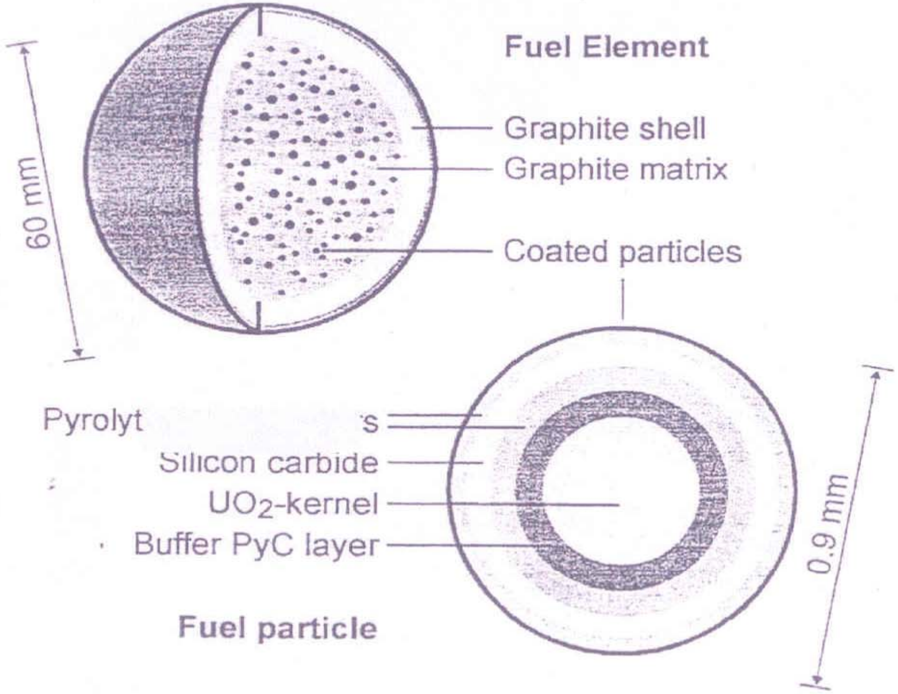
$_{94}Pu^{239}$ Fissile

Natural Thorium

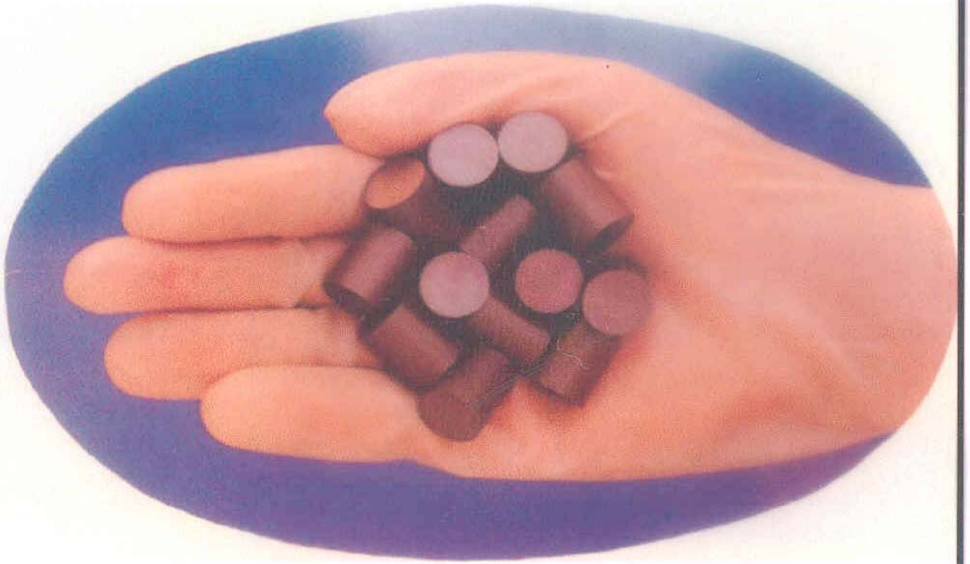
$_{90}Th^{232}$: 100% Fertile

Reactor Neutron Capture

$_{92}U^{233}$ Fissile

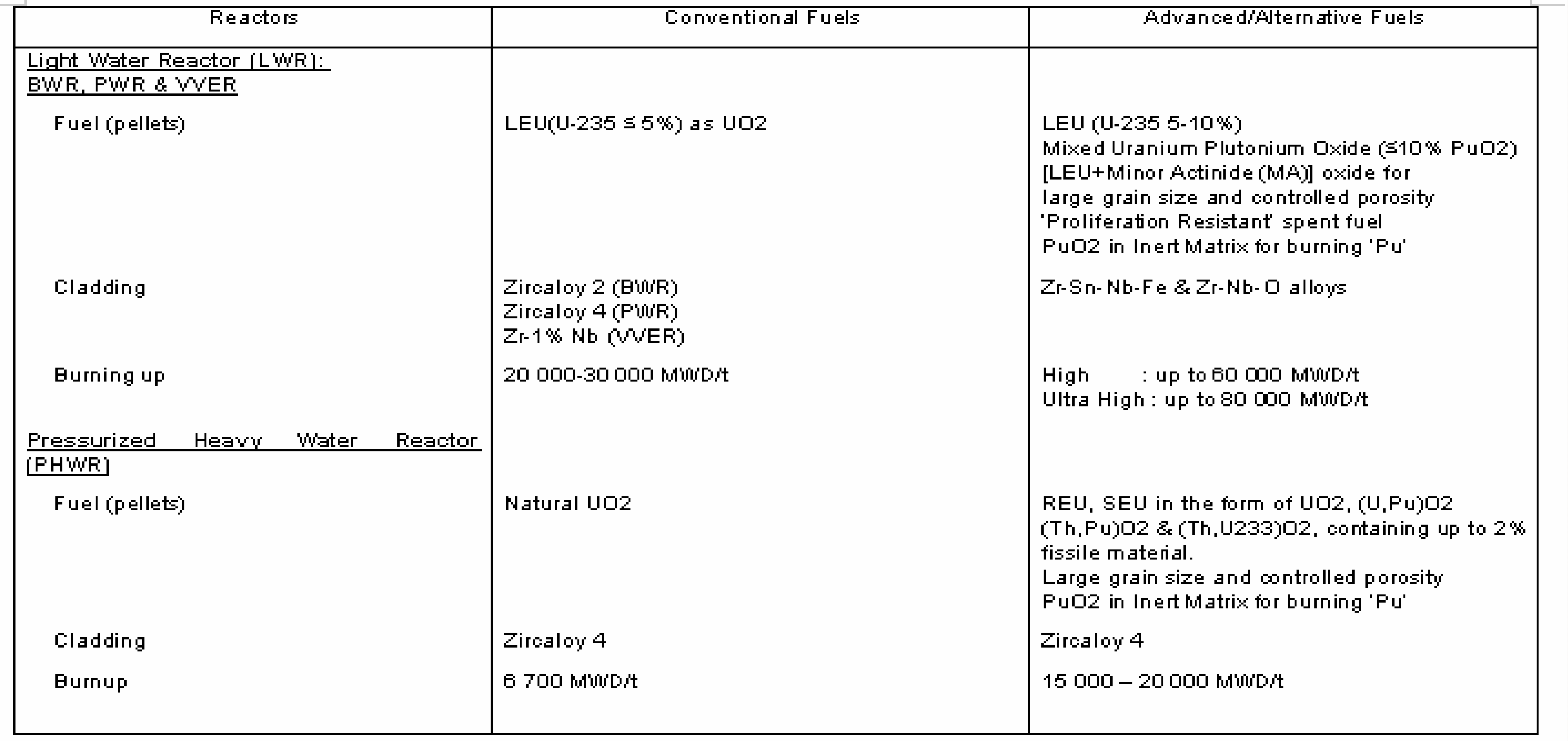


COATED FUEL PARTICLES



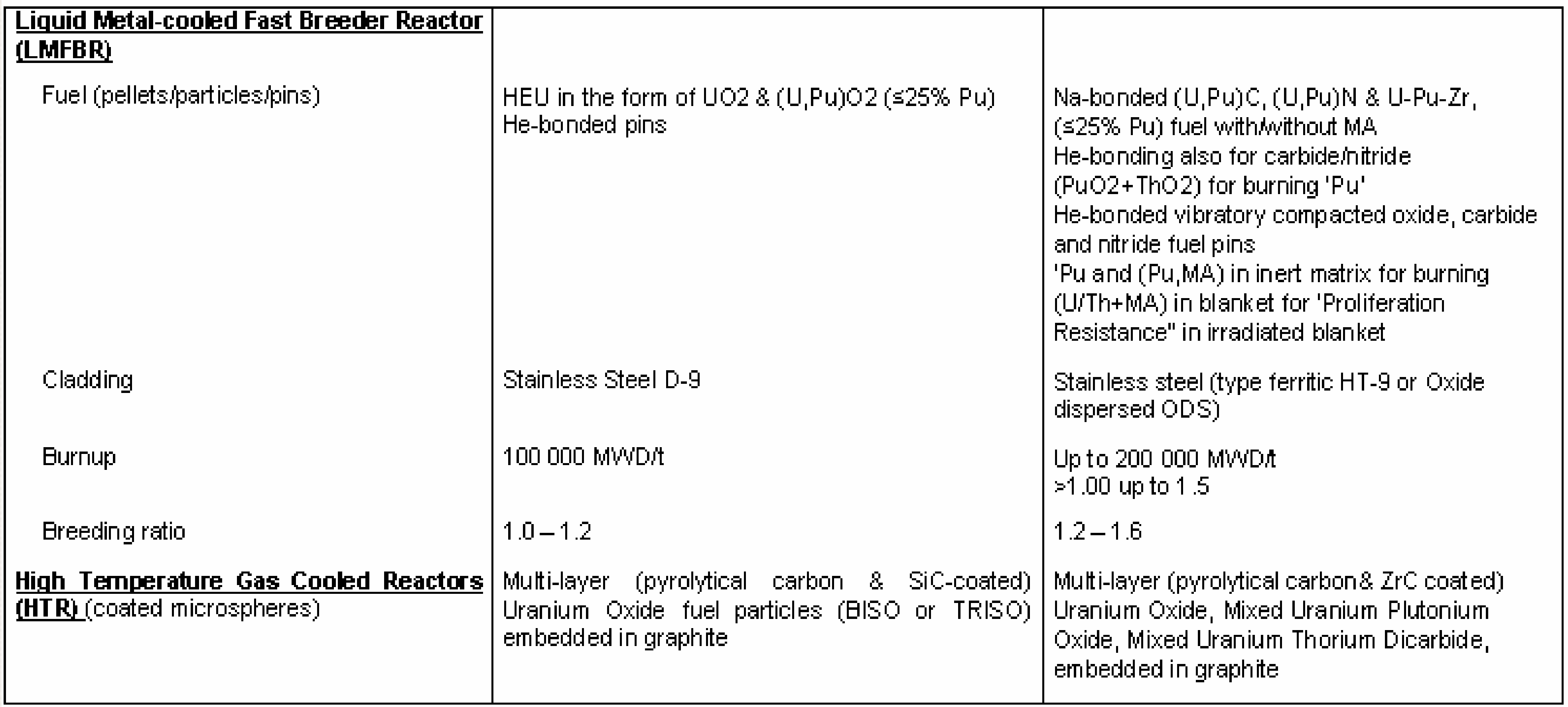
FUEL PELLETS

Conventional & Advanced Fuels For Nuclear Power Reactors

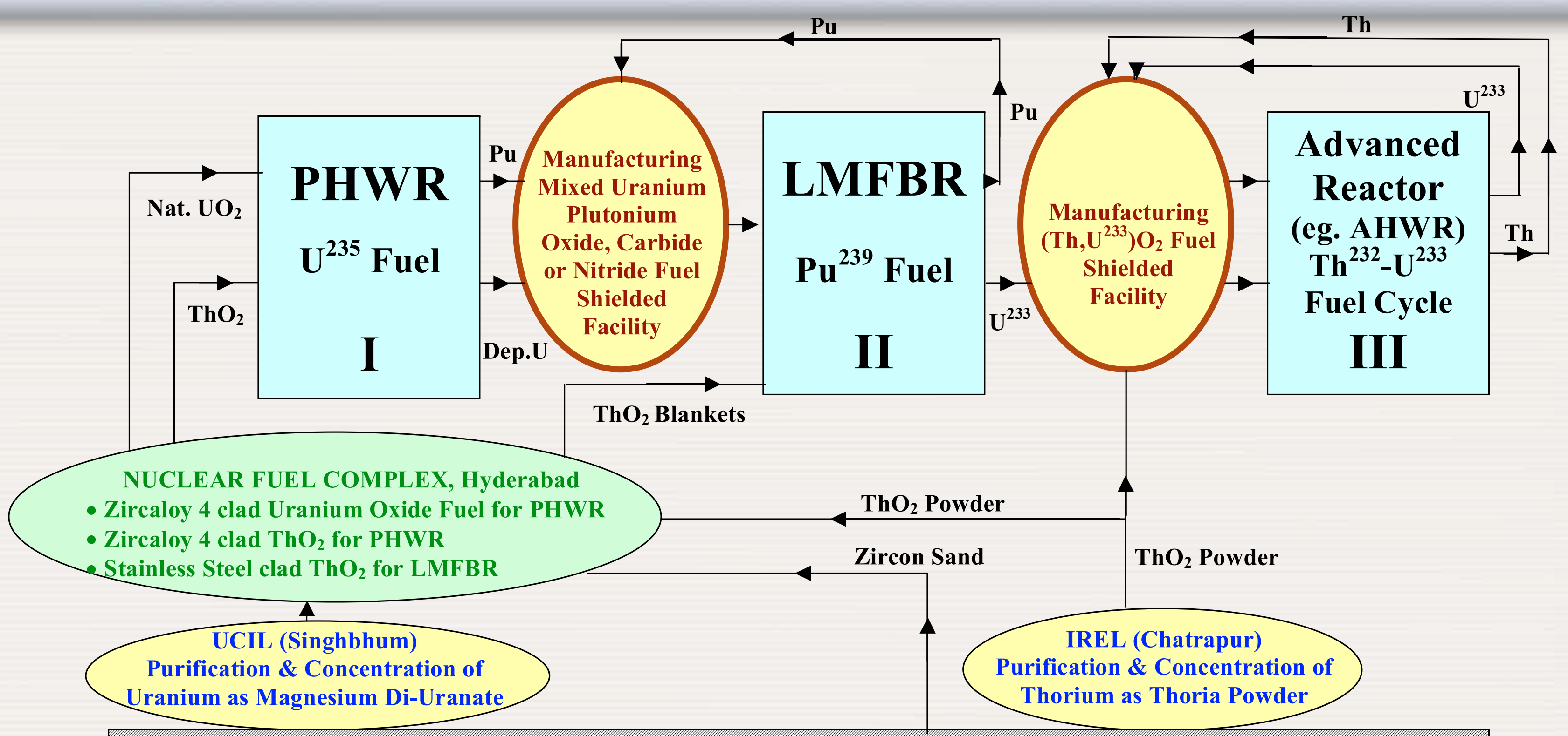


Reactors	Conventional Fuels	Advanced/Alternative Fuels
Light Water Reactor (LWR): BWR, PWR & VVER		
Fuel (pellets)	LEU(U-235 ≤ 5%) as UO2	LEU (U-235 5-10%) Mixed Uranium Plutonium Oxide (≤10% PuO2) [LEU+Minor Actinide (MA)] oxide for large grain size and controlled porosity 'Proliferation Resistant'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

Conventional & Advanced Fuels For Nuclear Power Reactors



Liquid Metal-cooled Fast Breeder Reactor (LMFBR)		
Fuel (pellets/particles/pins)	HEU in the form of UO2 & (U,Pu)O2 (≤25% Pu) He-bonded pins	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
Cladding	Stainless Steel D-9	Stainless steel (type ferritic HT-9 or Oxide dispersed ODS)
Burnup	100 000 MWD/t	Up to 200 000 MWD/t >1.00 up to 1.5
Breeding ratio	1.0 – 1.2	1.2 – 1.6
High Temperature Gas Cooled Reactors (HTR) (coated microspheres)	Multi-layer (pyrolytical carbon & SiC-coated) Uranium Oxide fuel particles (BISO or TRISO) embedded in graphite	Multi-layer (pyrolytical carbon& ZrC coated) Uranium Oxide, Mixed Uranium Plutonium Oxide, Mixed Uranium Thorium Dicarbide, embedded in graphite



PHWR
U^{235} Fuel
I

Nat. UO_2

ThO_2

Pu

Dep.U

Manufacturing Mixed Uranium Plutonium Oxide, Carbide or Nitride Fuel Shielded Facility

Pu

LMFBR
Pu^{239} Fuel
II

Pu

U^{233}

ThO_2 Blankets

Manufacturing $(Th,U^{233})O_2$ Fuel Shielded Facility

Th

U^{233}

Advanced Reactor (eg. AHWR) Th^{232}-U^{233} Fuel Cycle
III

Th

NUCLEAR FUEL COMPLEX, Hyderabad
- Zircaloy 4 clad Uranium Oxide Fuel for PHWR
- Zircaloy 4 clad ThO_2 for PHWR
- Stainless Steel clad ThO_2 for LMFBR

ThO_2 Powder

Zircon Sand

ThO_2 Powder

UCIL (Singhbhum)
Purification & Concentration of
Uranium as Magnesium Di-Uranate

IREL (Chatrapur)
Purification & Concentration of
Thorium as Thoria Powder

Mining of Uranium, Thorium & Zirconium

URANIUM CORPORATION OF INDIA LIMITED (UCIL)

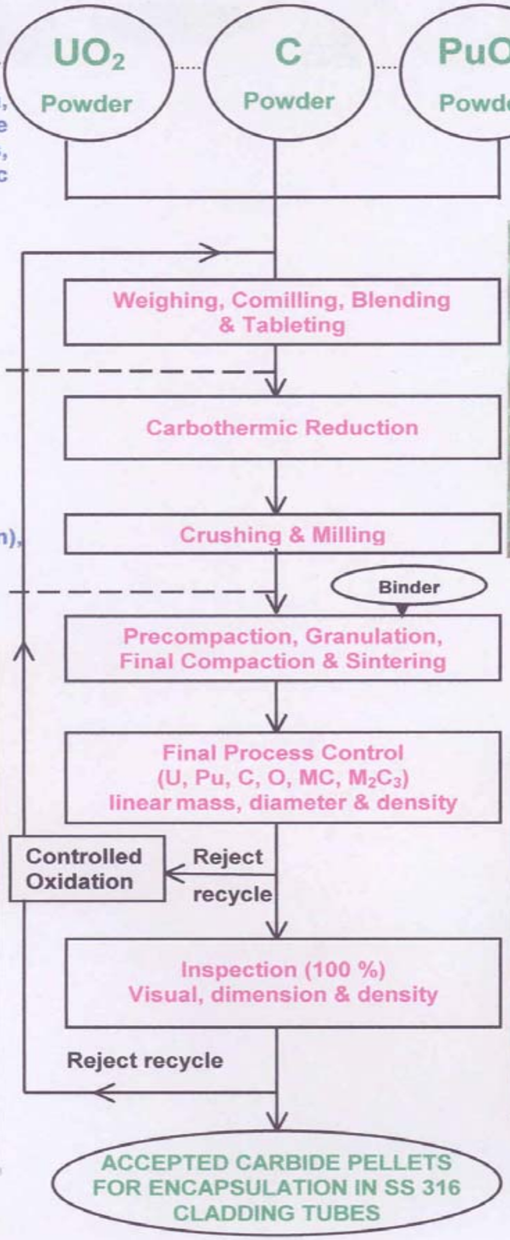
Operating Mines: Singhbhum, Jharkhand
Potential Mines: - Domiasiat & Wahkyn in Meghalaya
- Gogi in Bhima Basin in Karnataka
- Cuddapah Basin in Andhra Pradesh
- Rohil Ghat eswar in Rajasthan

INDIAN RARE EARTHS LIMITED (IREL)

Zircon & Monazite (for Thorium) from beach sands
Operating Mines: - Chavara, Kerala
- Manavalakurichi, Tamil Nadu
- Chatrapur, Orissa

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

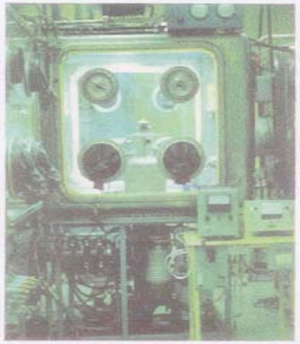
- Specific surface area, chemical purity of oxide & graphite powders, U&Pu contents, isotopic analysis & O/M ratios

- **Analysis**

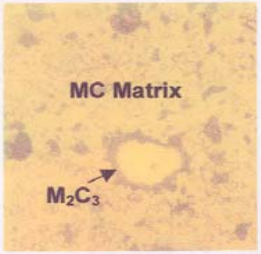
U, Pu (XRF)
C(Combustion)

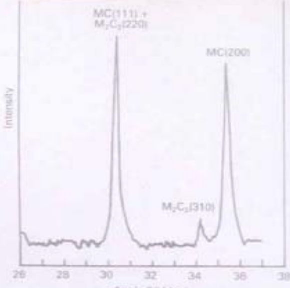
- **Analysis**

U,Pu (XRF),C (combustion),
O & N (Inert gas fusion)
MC, M_2C_3 & MO_2 (XRD)

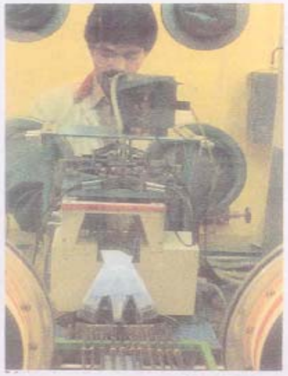


High Temperature High Vacuum Furnace for Carbothermic Synthesis of (U,Pu)C





X-ray diffraction pattern of carbide powder after sintering

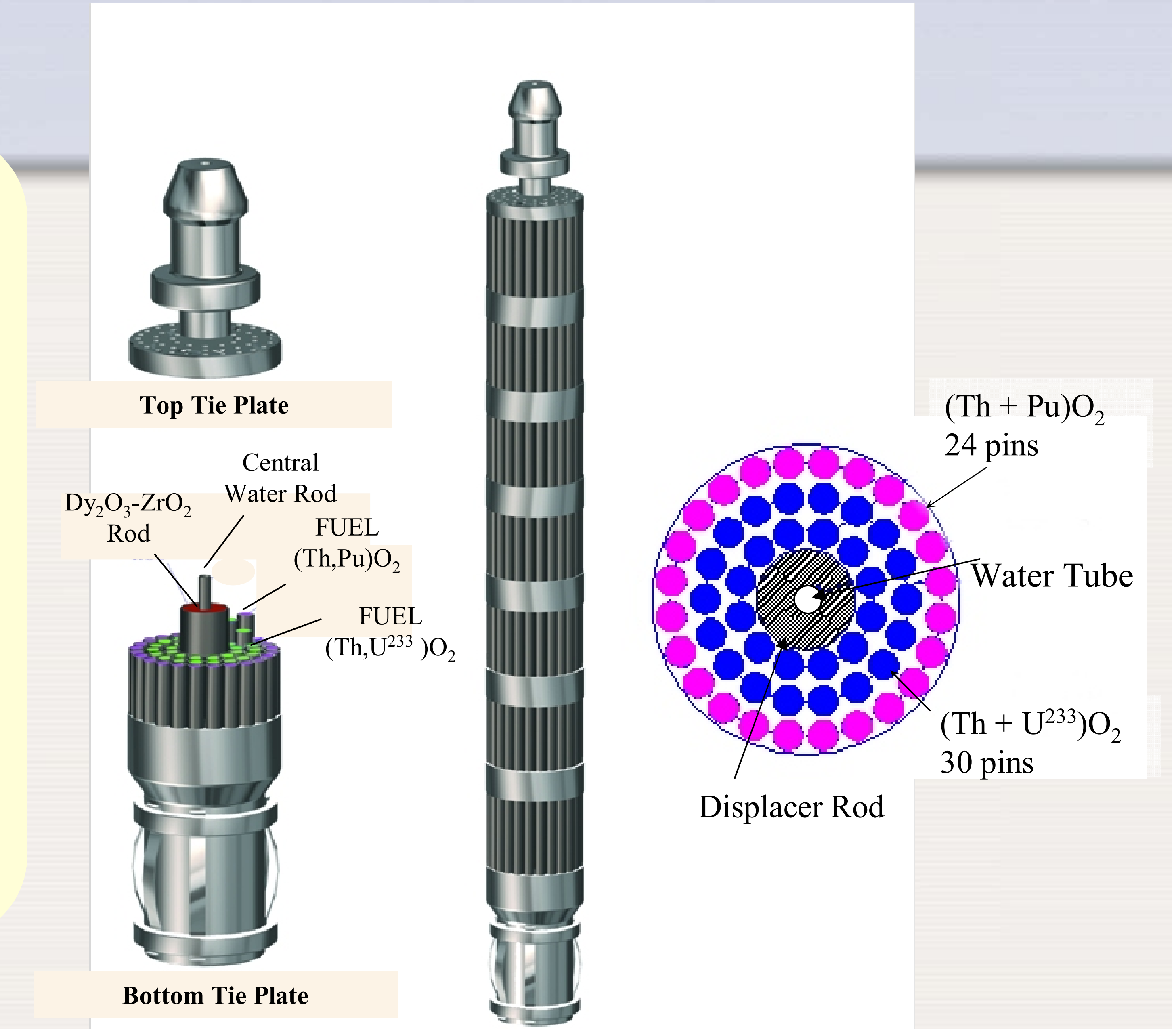


Automatic Fuel Pellet Inspection Machine (Dimension & Density)

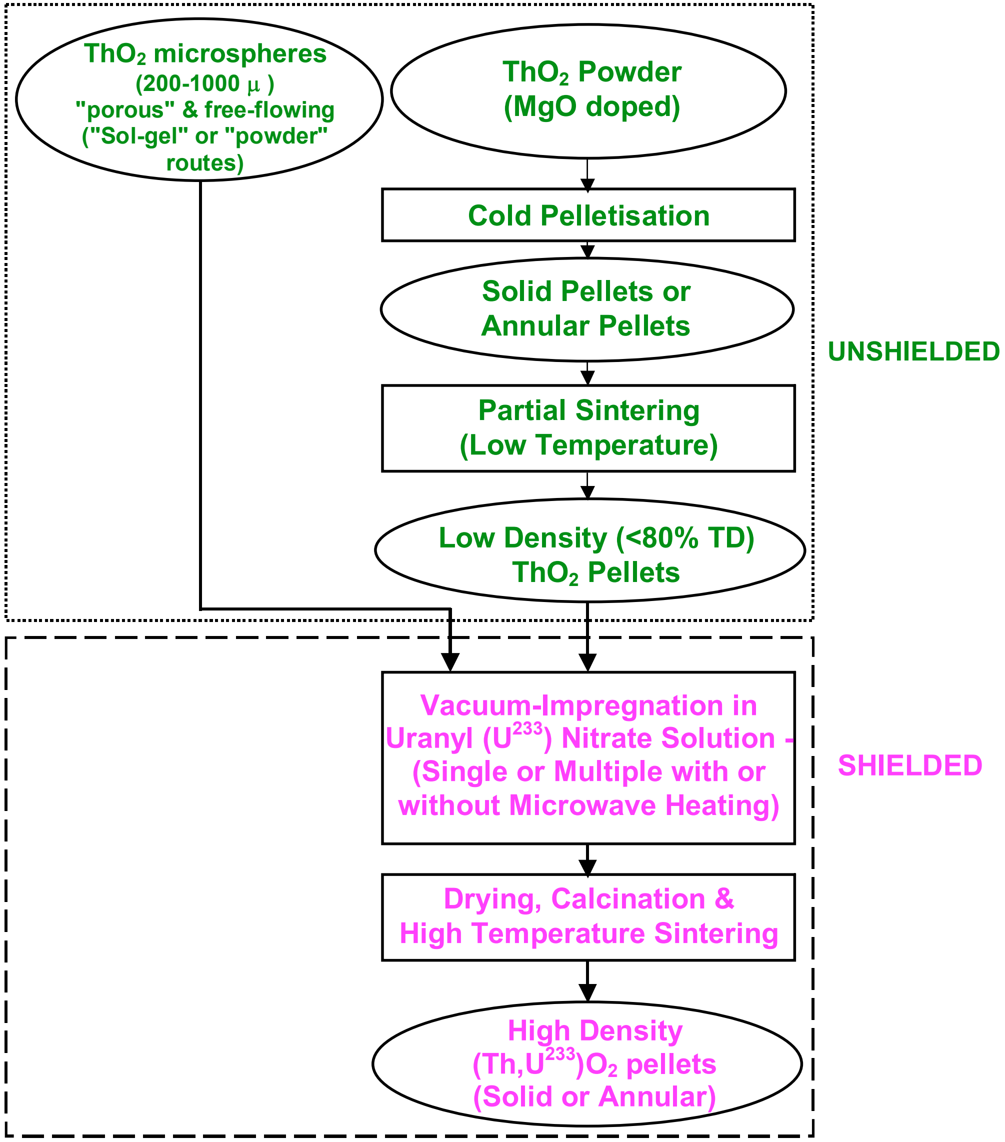
AHWR FUEL CLUSTER (D5)

Key Features

- Thorium bearing fuel [$(Th + Pu)O_2$ Mox, $(Th + {}^{233}U)O_2$ Mox]
- Hollow cylindrical $(ZrO_2\text{-}Dy_2O_3)$ displacer rod.
- Emergency core cooling water injected into the cluster through the holes in displacer rod.
- Low pressure drop design.



"Pellet-Impregnation Techniques" for manufacturing high density (Th,U^{233})O_2 fuel pellets



ThO_2 microspheres
(200-1000 μ)
"porous" & free-flowing
("Sol-gel" or "powder" routes)

ThO_2 Powder
(MgO doped)

Cold Pelletisation

Solid Pellets or Annular Pellets

Partial Sintering
(Low Temperature)

Low Density (<80% TD) ThO_2 Pellets

UNSHIELDED

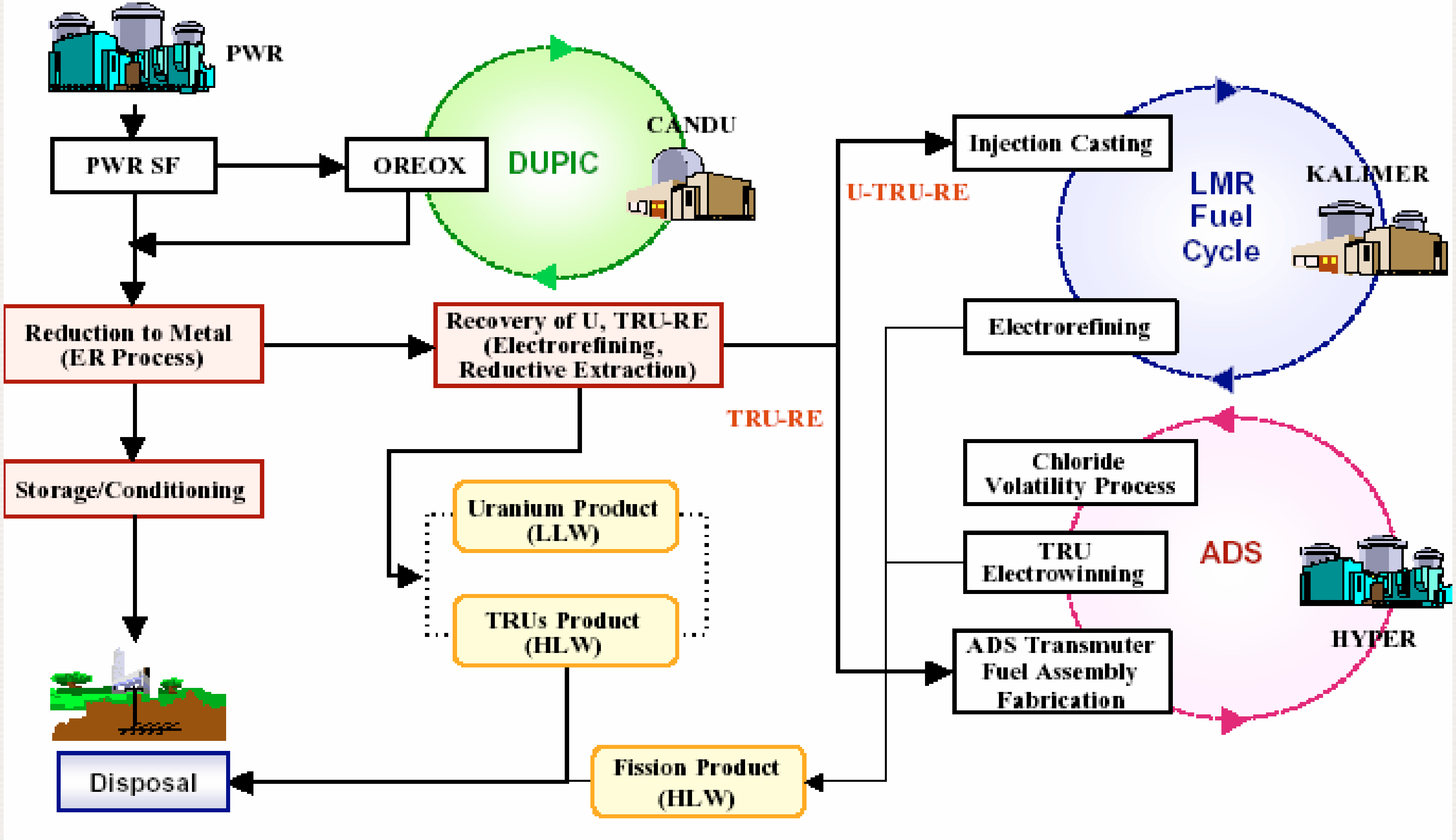
Vacuum-Impregnation in Uranyl (U^{233}) Nitrate Solution - (Single or Multiple with or without Microwave Heating)

Drying, Calcination &
High Temperature Sintering

High Density
(Th,U^{233})O_2 pellets
(Solid or Annular)

SHIELDED

Advanced fuel cycle schemes



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

Country / Company	Facility / Location	Fuel Type	Capacity (tHM/year)
France, COGEMA	UP2 and UP3, La Hague	LWR	1700
UK, BNFL	Thorp, Sellafield	LWR, AGR	1200
UK, BNFL	B205 Magnox	Magnox GCR	1500
Russian Federation, Minatom	RT-1 / Tcheliabinsk-65 Mayak 400	VVER	400
Japan, JNC	Tokai-Mura	LWR, ATR	90
Japan, JNFL	Rokkasho-Mura (under construction)	LWR	800
India, BARC	PREFRE-1, Tarapur PREFRE-2, Kalpakkam	PHWR PHWR	100 100
China, CNNC	Diowopu (Ganzu)	LWR	25-50

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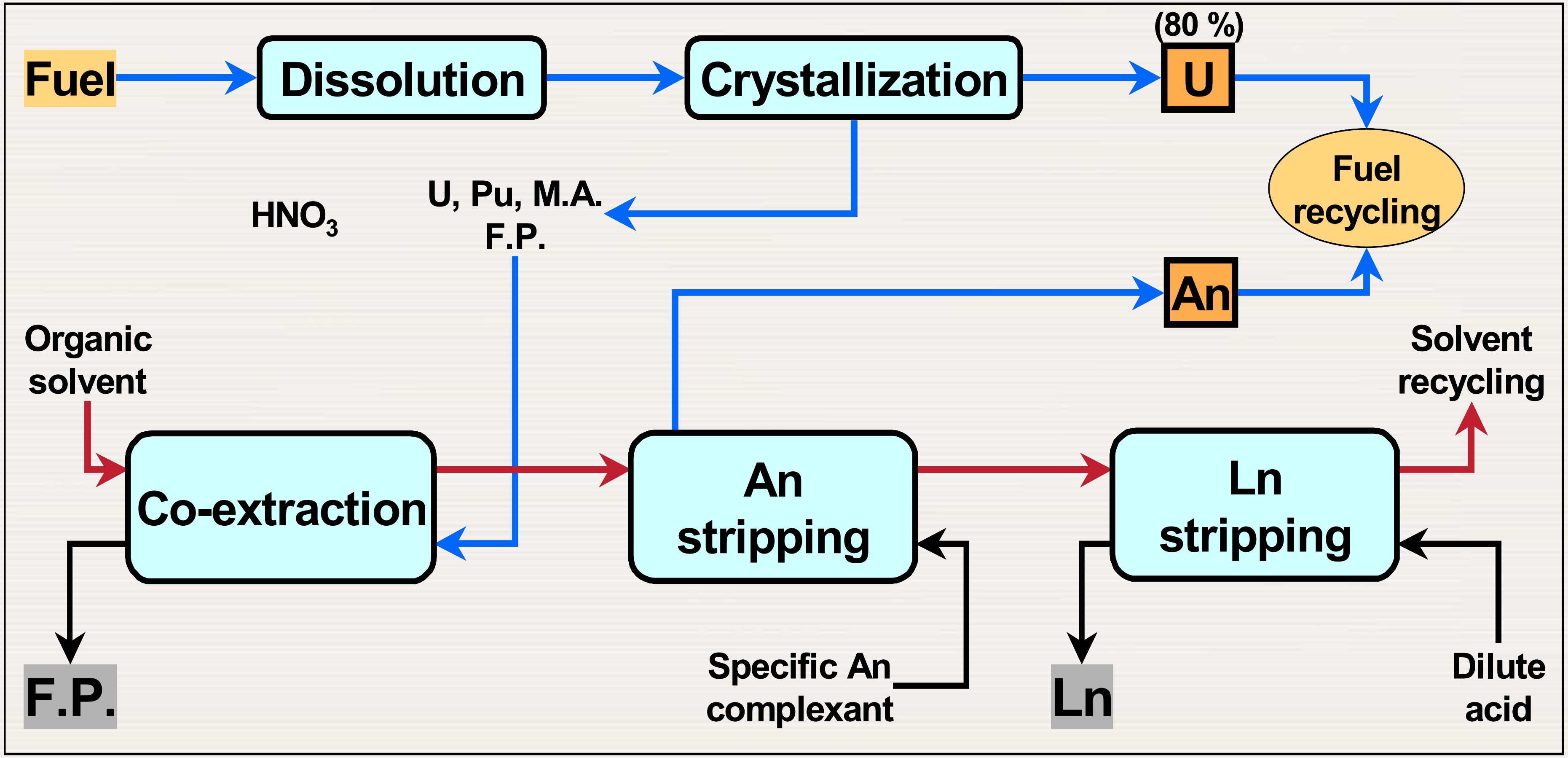
MIXED URANIUM PLUTONIUM OXIDE (MOX) FUEL FABRICATION FACILITIES

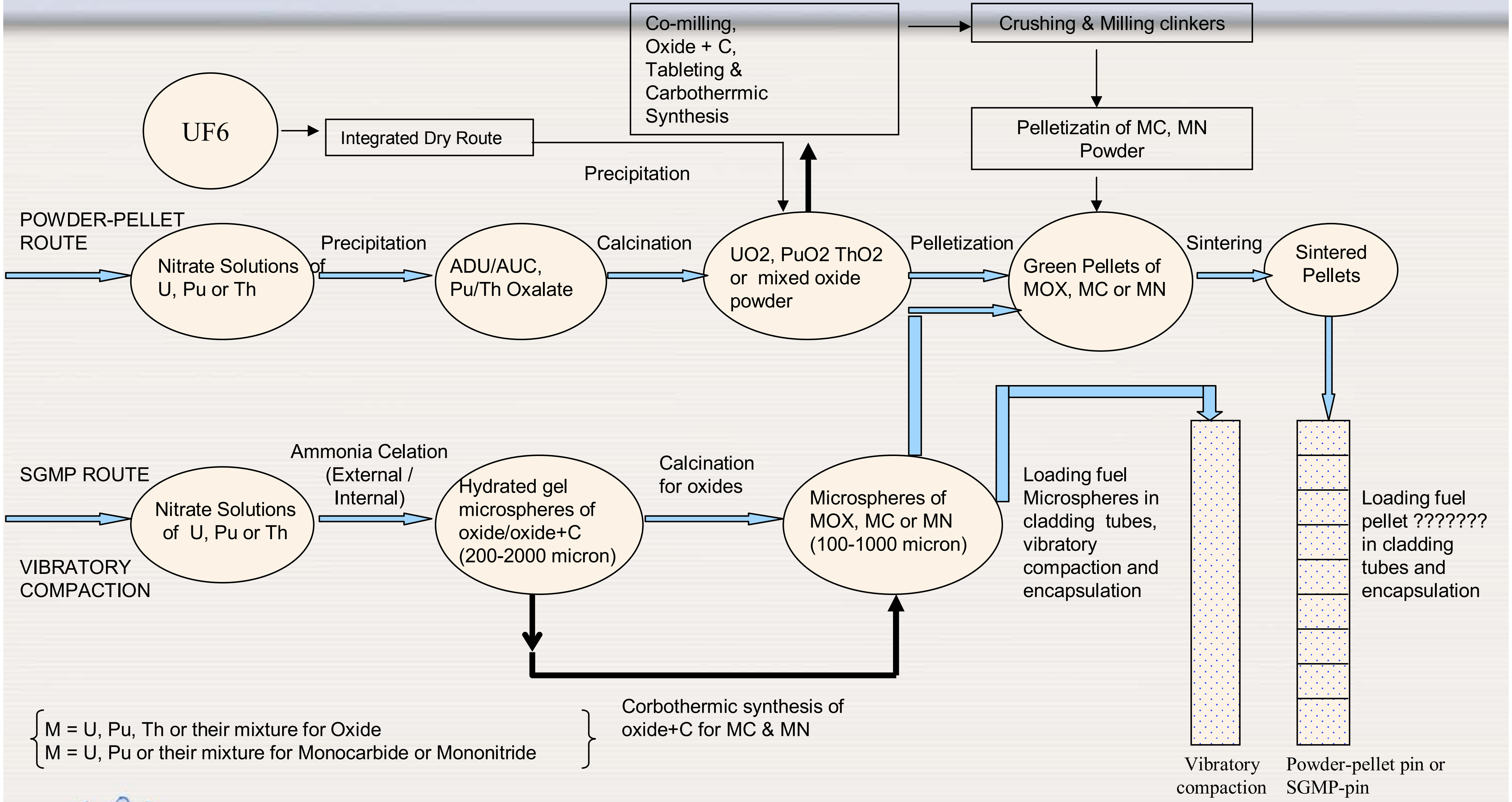
Country / Company	Facility / Location	Fuel Type	Capacity (tHM/year)
France, COGEMA	Cadarache	LWR, FBR	40
France, COGEMA	Marcoule-Melox	LWR	100
Belgium, Belgonucleaire	Dessel	LWR	40
UK, BNFL	Sellafield SMP	LWR	120
UK	Sellafied MDF	LWR	8
Russian Federation, Minatom	Chelyabinsk	FBR	60
Japan, JNC	Tokai-Mura	ATR	10
Japan, JNFL	Rokkasho	LWR	130
India, AFFF, BARC	Tarapur	LWR, PHWR & FBR	

The GANEX concept : Group ActiNides EXtraction



Objectives of advanced methods of fabrication of ceramic nuclear fuel pellets

Safety	Economics	Performance
• Avoid generation and handling of powder of fuels for minimising : - radiotoxic dust hazard - fire hazard (for carbide & nitride fuels) • Fabrication flow sheet should be amenable to automation & remotisation - for minimising personnel exposure to radiation	• Minimise process steps • Reduce fuel synthesis & sintering temperatures • Reduce gas cost during synthesis and sintering - gas purification and recirculation - alternative less expensive gas • Reduce process losses and rejects	• Tailor make fuel microstructure for higher burn up - High density (≥96% T.D.), closed “porosity” and large (>25μ) grain size for LWR & PHWR - Low density (<85% T.D.) “open” porosity and small (<5μ) grain size for LMFBR - Excellent micro-homogenity of fissile material in fuel - avoid fine pores (<1μ) for minimising in-pile densification



UF6

Integrated Dry Route

Co-milling,
Oxide + C,
Tableting &
Carbotherrmic
Synthesis

Crushing & Milling clinkers

Pelletizatin of MC, MN
Powder

Precipitation

POWDER-PELLET
ROUTE

Nitrate Solutions of
U, Pu or Th

Precipitation

ADU/AUC,
Pu/Th Oxalate

Calcination

UO2, PuO2 ThO2
or mixed oxide
powder

Pelletization

Green Pellets of
MOX, MC or MN

Sintering

Sintered
Pellets

SGMP ROUTE

VIBRATORY
COMPACTION

Nitrate Solutions
of U, Pu or Th

Ammonia Celation
(External /
Internal)

Hydrated gel
microspheres of
oxide/oxide+C
(200-2000 micron)

Calcination
for oxides

Microspheres of
MOX, MC or MN
(100-1000 micron)

Loading fuel
Microspheres in
cladding tubes,
vibratory
compaction and
encapsulation

Loading fuel
pellet ???????
in cladding
tubes and
encapsulation

Corbothermic synthesis of
oxide+C for MC & MN

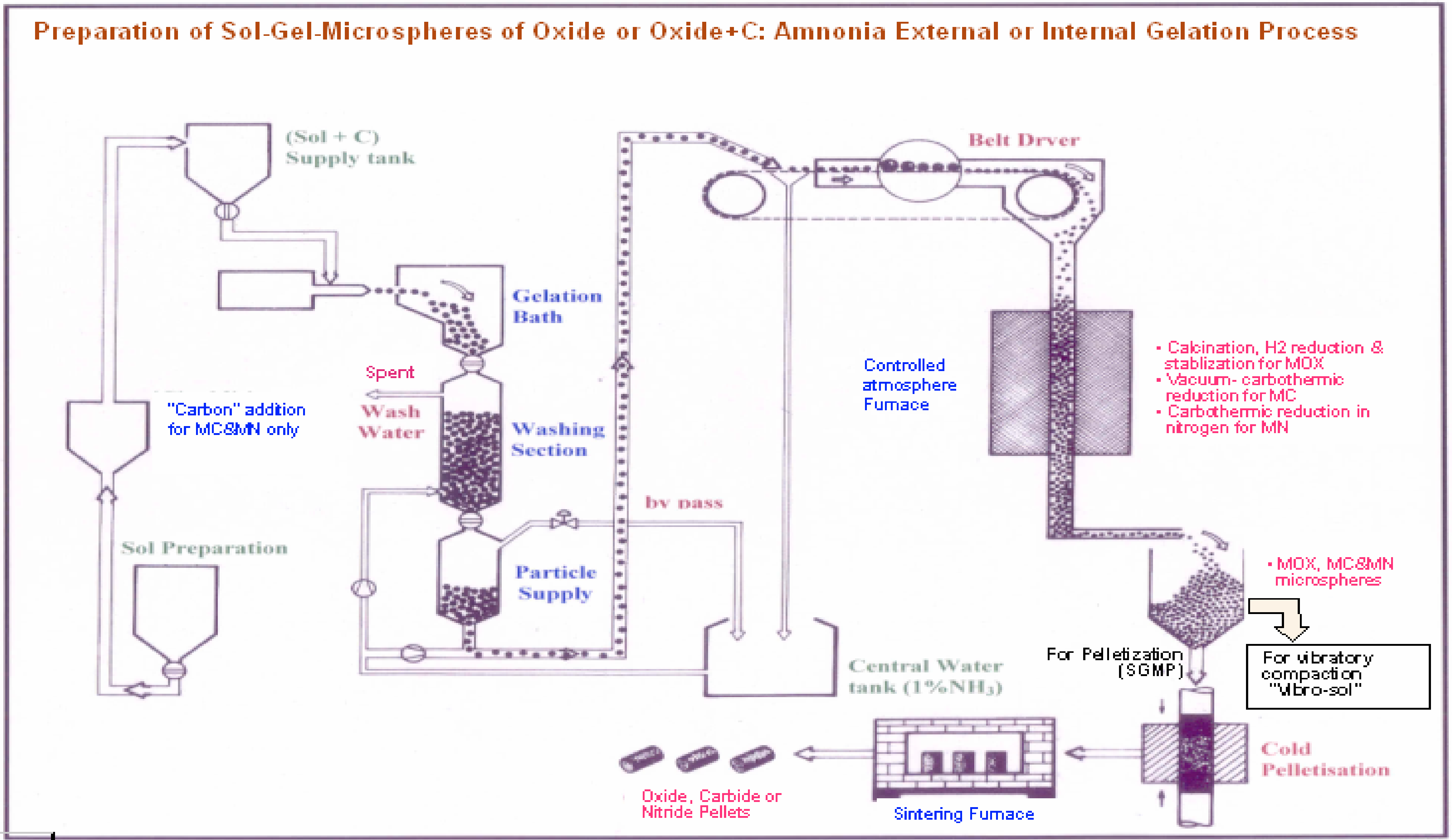
M = U, Pu, Th or their mixture for Oxide
M = U, Pu or their mixture for Monocarbide or Mononitride

Vibratory
compaction

Powder-pellet pin or
SGMP-pin

"Sol-Gel-Microsphere Pelletization (SGMP)" and "Vibro-Sol" Processes for Manufacturing Mixed Uranium Plutonium Oxide (MOX), Monocarbide (MC) and Mononitride (MN) Fuels for LMFR

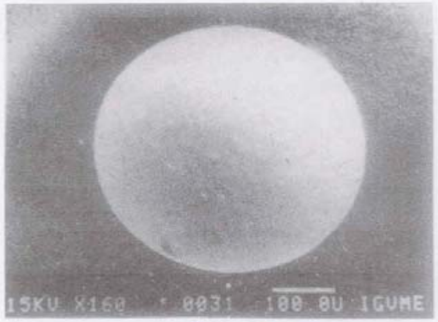
Preparation of Sol-Gel-Microspheres of Oxide or Oxide+C: Amnonia External or Internal Gelation Process



Tailored Microstructure of Oxide Fuel Pellets prepared by Sol-Gel Microsphere Pelletisation Process

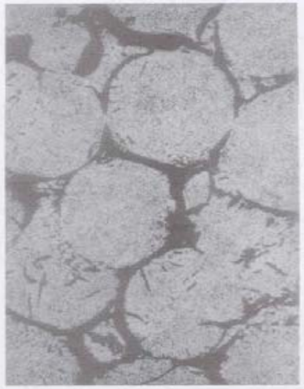
Porous Microspheres
(easily crushable)

Non-Porous Microspheres
(hard and not easily crushable)



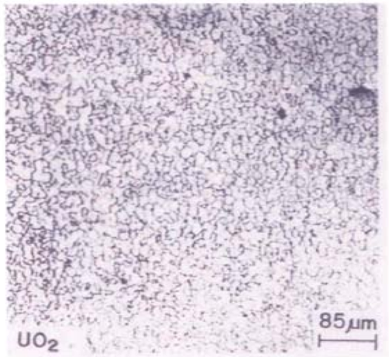
Direct pelletisation followed by Sintering

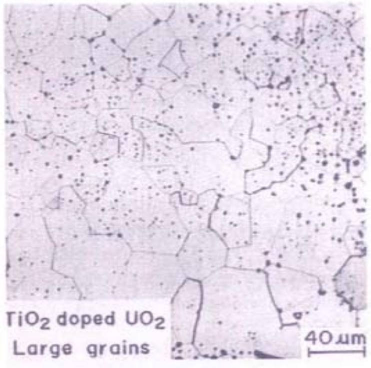
Sintered Fuel Pellets of High Density & Uniformly distributed 'closed' pores
– suitable for PHWRs & LWRs

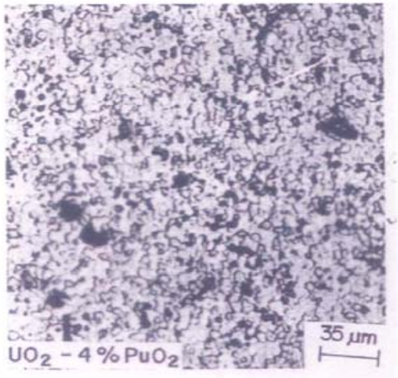


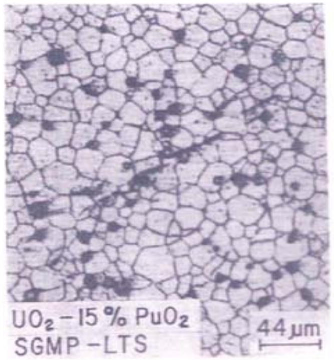
Sintered Fuel Pellets with black berry structure, low density and 'open' pores
– suitable for LMFBRs

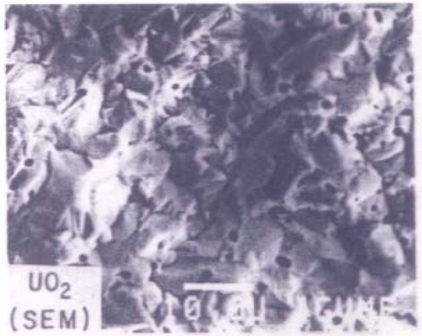
Microstructures of high density oxide and mixed oxide fuel pellets fabricated by SGMP route for PHWR and LWR

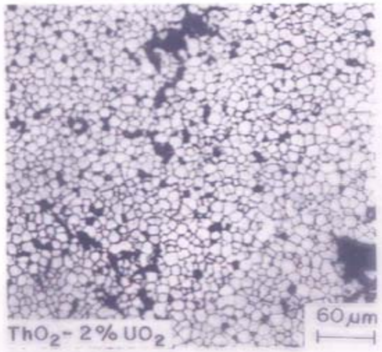

UO_2


TiO_2 doped UO_2
Large grains

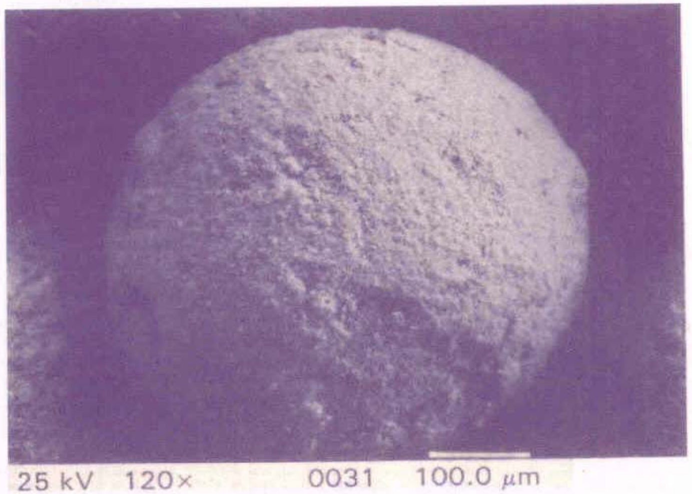

UO_2 - 4 % PuO_2


UO_2 - 15 % PuO_2
SGMP - LTS


UO_2
(SEM)

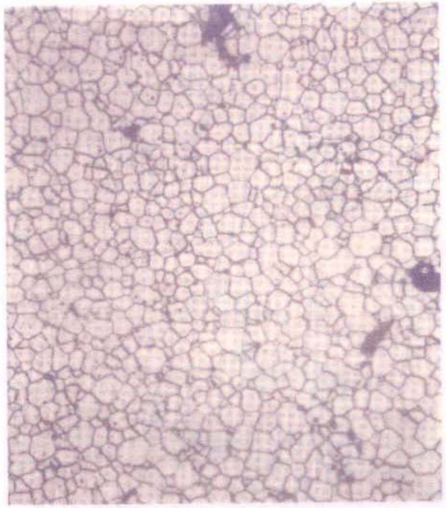

ThO_2 - 2 % UO_2

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

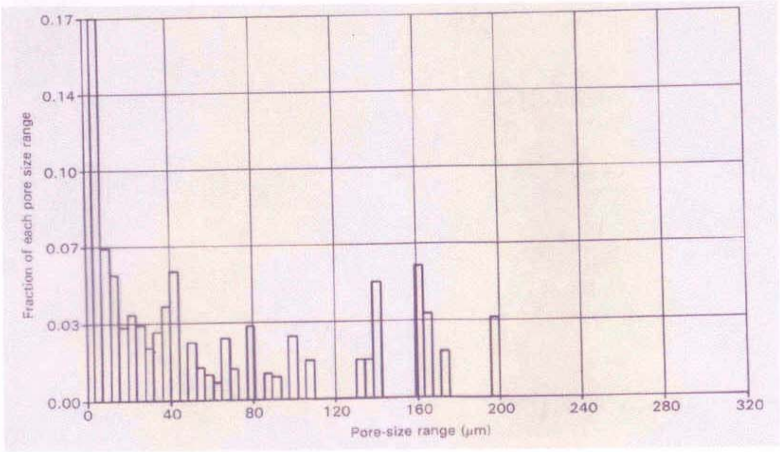


Image Analysis of entire cut section of pellet (for determining undissolved microsphere boundary)

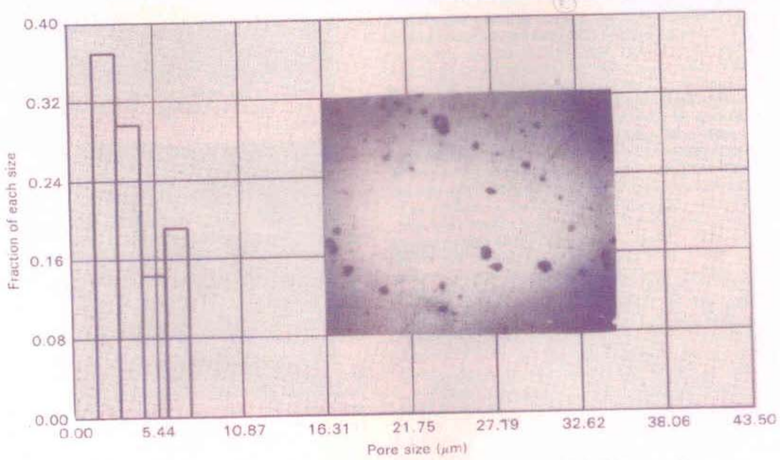


Image Analysis of selected areas (54µ x 54µ) (for average pore size & distribution)

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)

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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}U > 90 %) –alloys or compounds

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EXAMPLES OF INERT MATRIX

Inert Matrix type	Inert Matrix formula
Element	**C, Mg, Al, Si, Cr,V, Zr, Mo, W**
Inter-metallics	**AlSi, AlZr, ZrSi**
Alloy	**Stainless steel, zirconium alloys**
Carbide	**SiC, TiC, ZrC**
Nitrides	**AlN, TiN, ZrN, CeN,**
Binary oxide	MgO, Y_2O_3, ZrO_2, CeO_2
Ternary oxide	$MgAl_2O_4$, $Y_3Al_5O_{12}$, $ZrSiO_4$
Oxide solid solution	$Y_yZr_{1-y}O_{2-y/2}$, $Mg_{(1-x)}Al_{(2+x)}O_{(4-x)}$

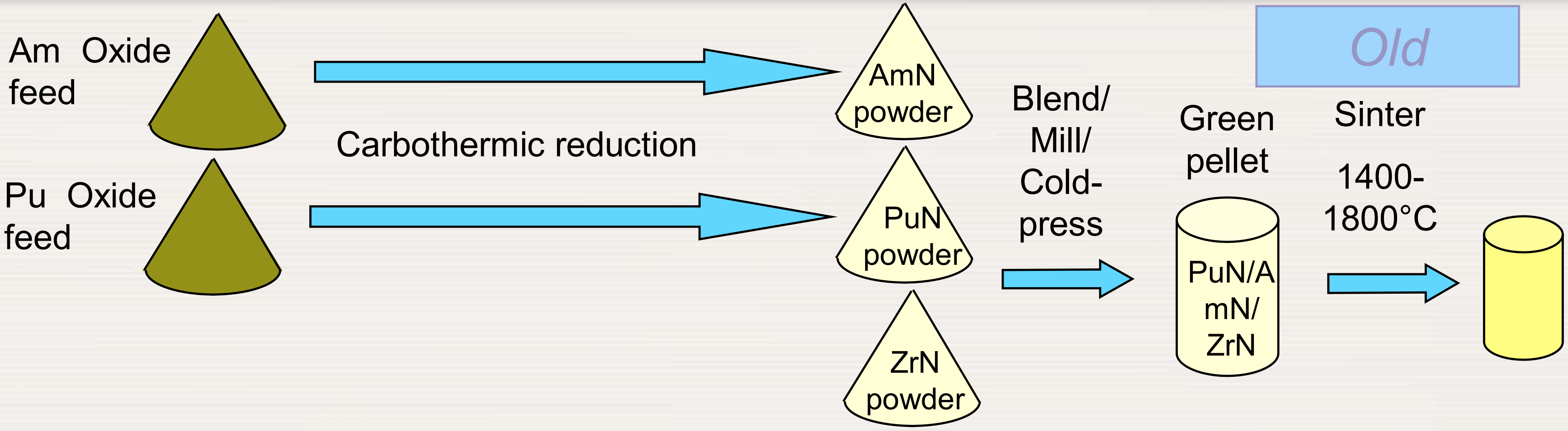
Example of heterogeneous materials as IMF's

Design	Composition
Solid solution	$An_zY_yPu_xZr_{1-y}O_{2-\psi}{}^*$
Cercer	$MgAl_2O_4$ - $Y_yPu_xZr_{1-y}O_{2-y/2}{}^*$
Cermet	Zr - $Y_yPu_xZr_{1-y-x}O_{2-y/2}{}^*$
Metmet	$PuAl_4{}^*$-Al

Examples of Inert Matrix design and additives

Additive type	**Additive formula**
Burnable poison	B, Gd, Dy, Ho, Er, Eu
Resonance additive	Fe, W, Th, U
Stabiliser	Y_2O_3, CaO in ZrO_2 Al_2O_3 in SiC

Development of Nitride Pellet Synthesis at LANL



Old

Am Oxide feed

Pu Oxide feed

Carbothermic reduction

AmN powder

PuN powder

ZrN powder

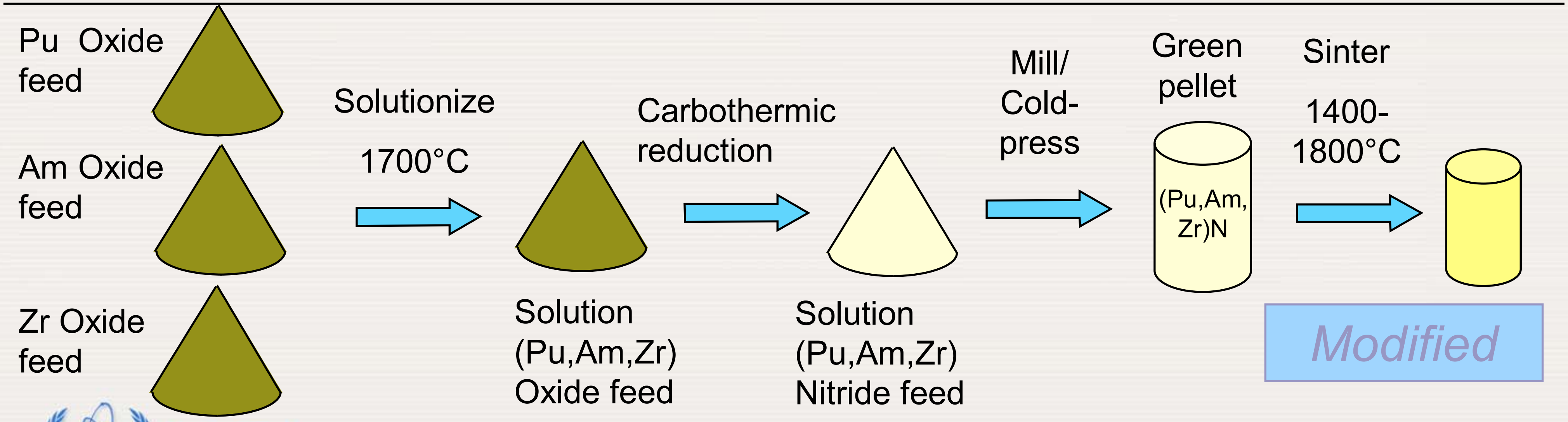
Blend/ Mill/ Cold-press

Green pellet

PuN/AmN/ZrN

Sinter

1400-1800°C



Pu Oxide feed

Am Oxide feed

Zr Oxide feed

Solutionize

1700°C

Solution (Pu,Am,Zr) Oxide feed

Carbothermic reduction

Solution (Pu,Am,Zr) Nitride feed

Mill/ Cold-press

Green pellet

(Pu,Am,Zr)N

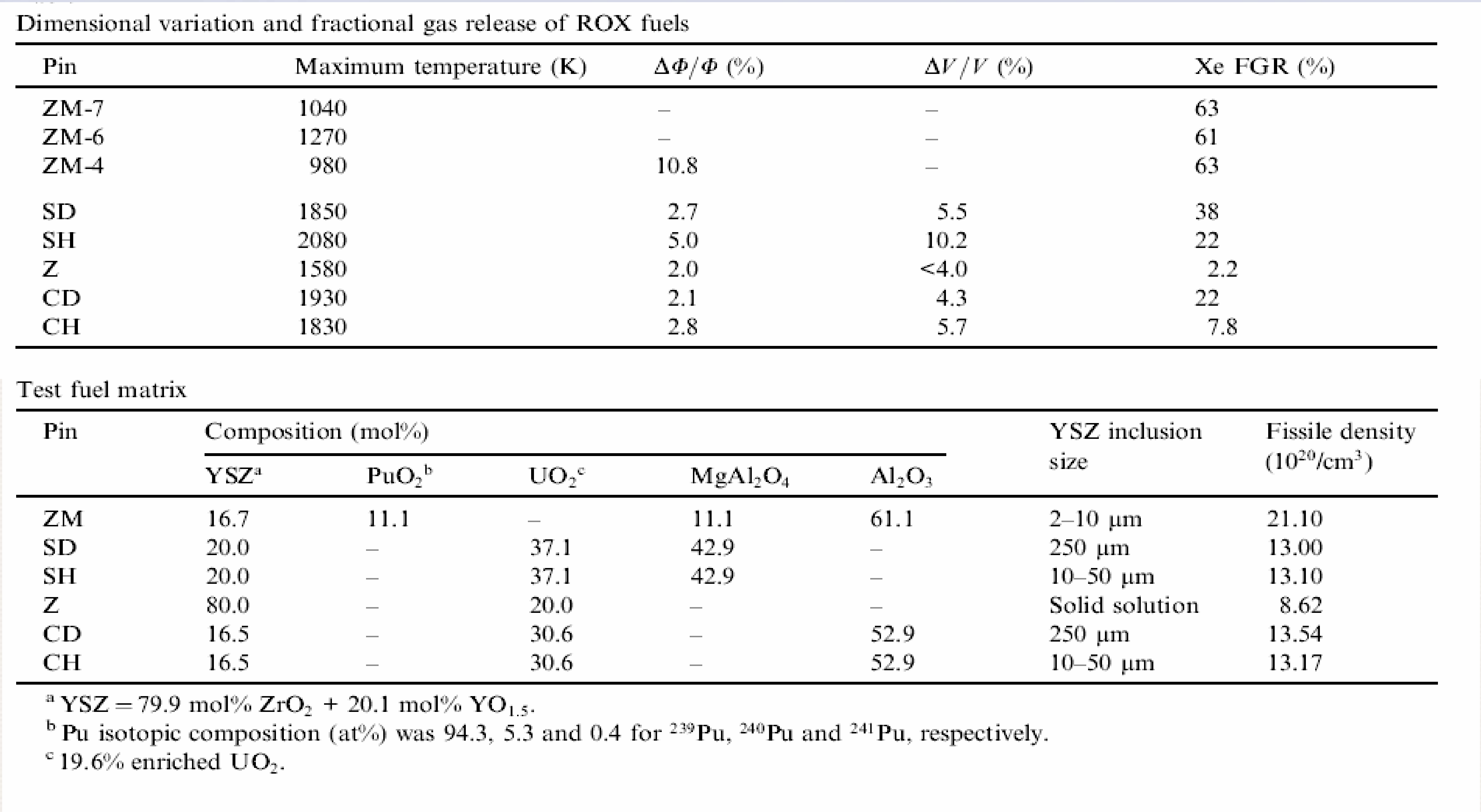
Sinter

1400-1800°C

Modified

S. Willson, LANL

Irradiation behavior of rock-like oxide fuels



Dimensional variation and fractional gas release of ROX fuels

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

Test fuel matrix

Pin	Composition (mol%)					YSZ inclusion size	Fissile density (10^{20}/cm^3)
	YSZ[a]	PuO_2[b]	UO_2[c]	$MgAl_2O_4$	Al_2O_3		
ZM	16.7	11.1	–	11.1	61.1	2–10 μm	21.10
SD	20.0	–	37.1	42.9	–	250 μm	13.00
SH	20.0	–	37.1	42.9	–	10–50 μm	13.10
Z	80.0	–	20.0	–	–	Solid solution	8.62
CD	16.5	–	30.6	–	52.9	250 μm	13.54
CH	16.5	–	30.6	–	52.9	10–50 μm	13.17

[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.

Ref; Yamashita et.al., Journal of Nuclear Materials 320 (2003) 126–132

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**

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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.

– **Pryoprocessing involving fluoride volatilization**
The process includes fluorination followed by distillation. The method has so far been demonstrated on a laboratory scale only.