Fuel and Fuel Cycle Aspects for GenIV Reactors

Alexander Bychkov - consultant, international expert
Ex-DDG-NE IAEA (2011-2015)

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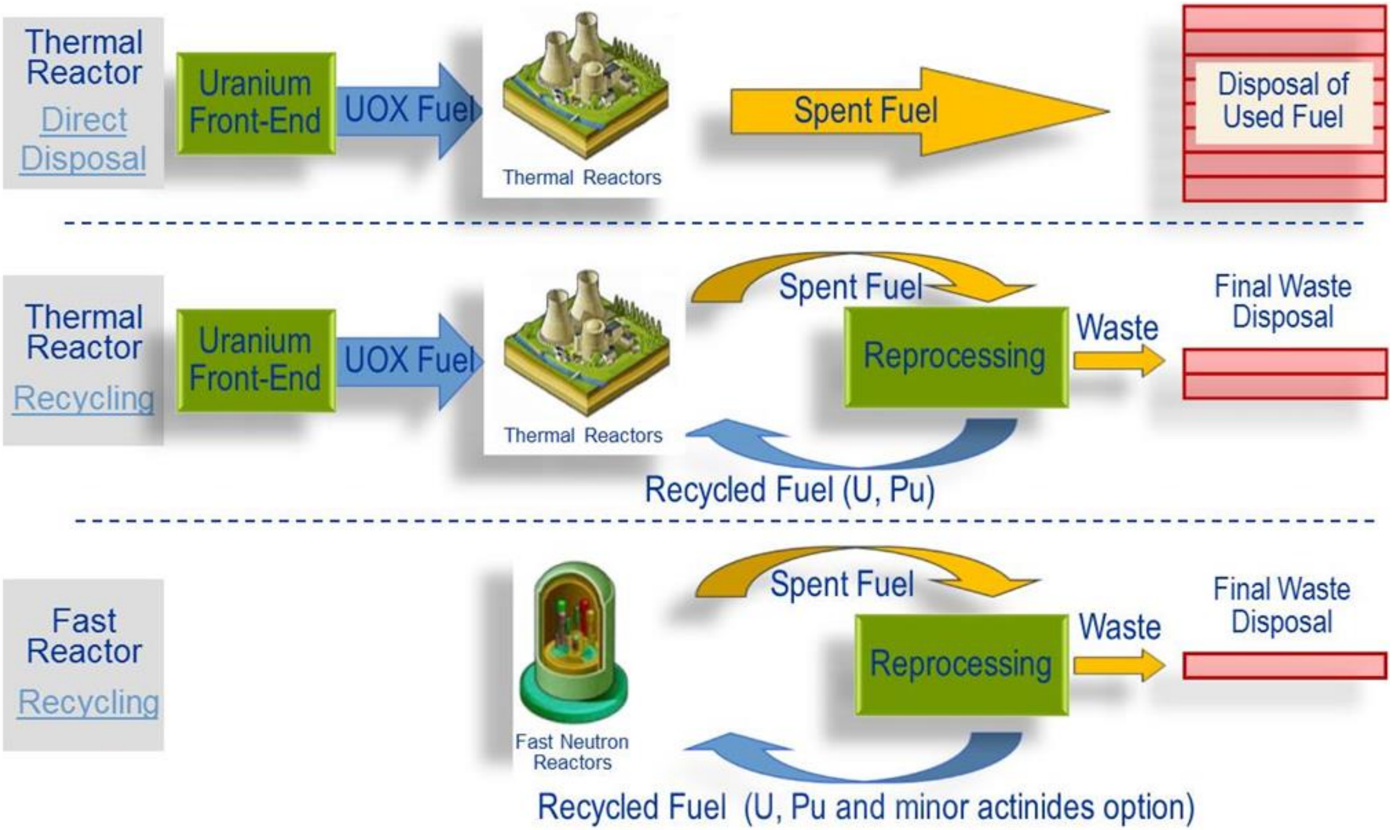
Introduction:

Current nuclear fuel cycle, fuel manufacture and future trends

Nuclear fuel cycle options and trends

- For Nuclear power **sustainability**, nuclear fuel cycle must remain economically viable and competitive through **Optimization of fissile materials**' use in reactor cores or valuable materials recycling
- This results in **different fuel cycle options**, some already implemented and others may be deployed in the future
- **Integrated approach** of the fuel cycle for advanced reactors

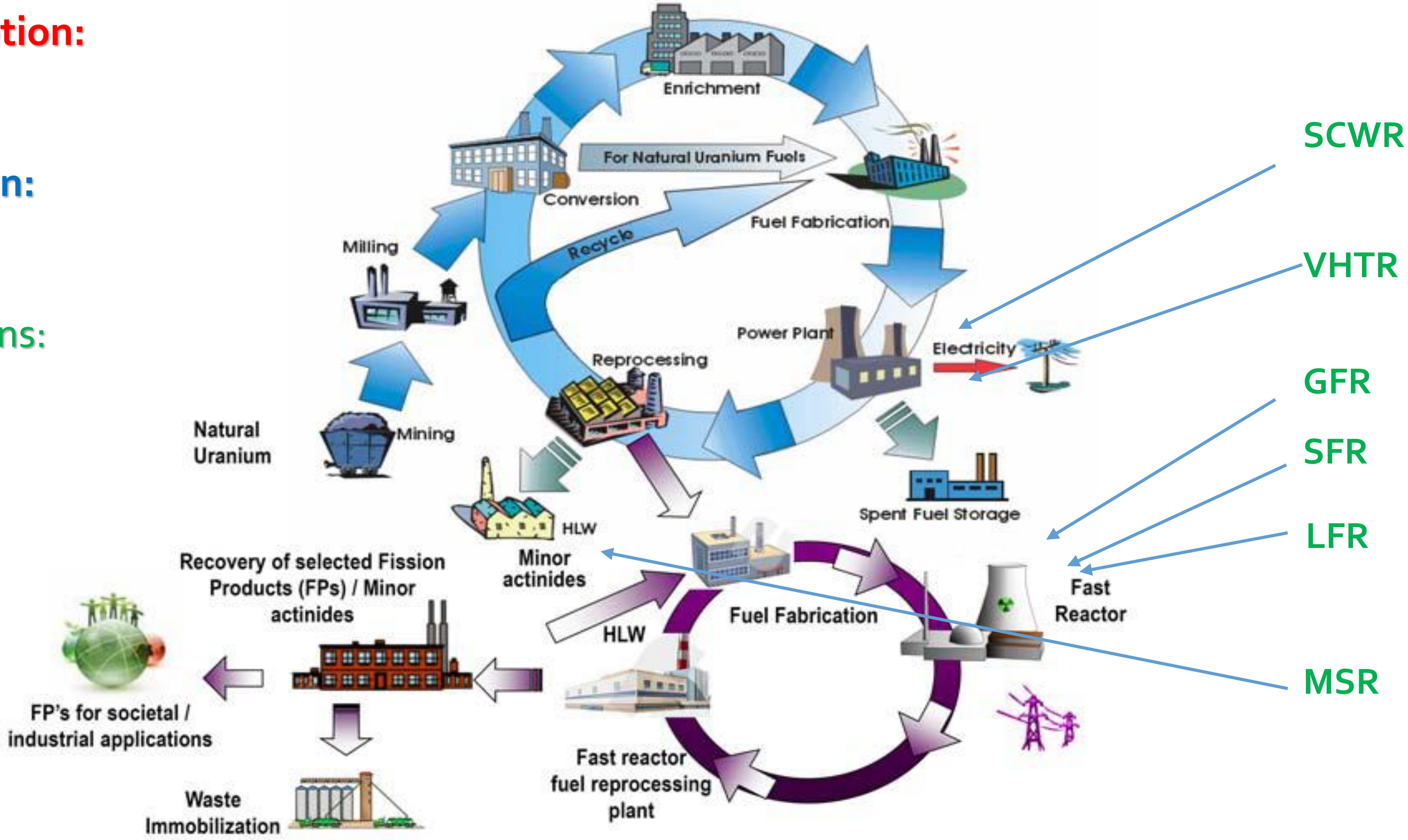


Nuclear Fuel Cycle Options for future

Main current option:
U-235

Advanced option:
U-238 – Pu

Innovative options:
U-Pu-MA,
Th – U-233



Nuclear Fuel Cycle more attractive for innovations than Reactor Systems

Innovations in Nuclear Fuel Cycle Front-end:

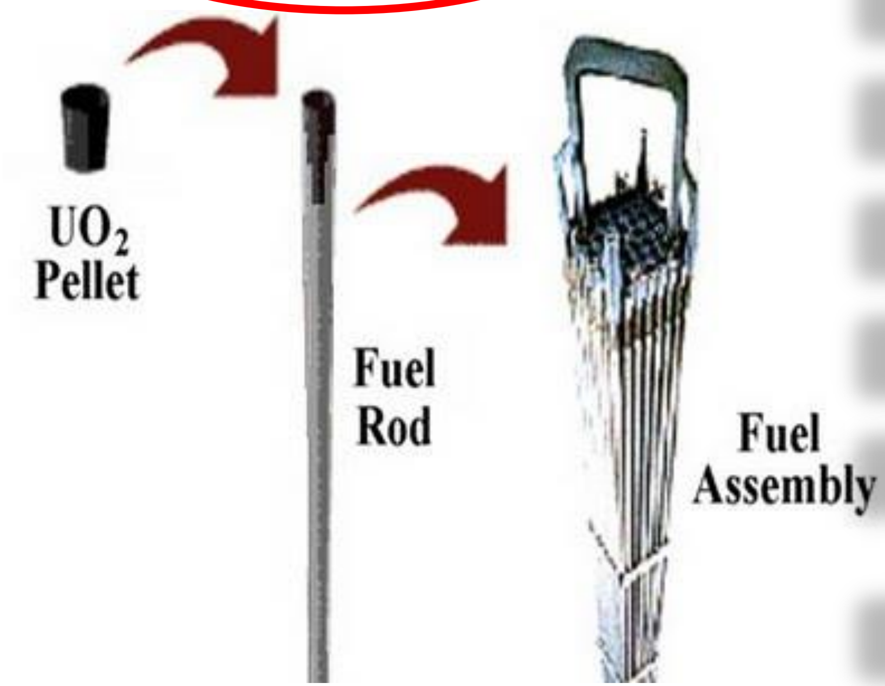
- **Uranium Exploration** – new remote technologies. Extension of available uranium resources
- **Uranium mining by ISL technology** – economically acceptable and drastically reduce environmental effects. No uranium mining tails etc.
- **Improvement of centrifuges and cascade management** – stable cost of LEU and possibility to use "depleted uranium" from early programs. Enrichment of UF_6 with 0.4% U-235 reduces demands of natural uranium and exclude all conversion stages.

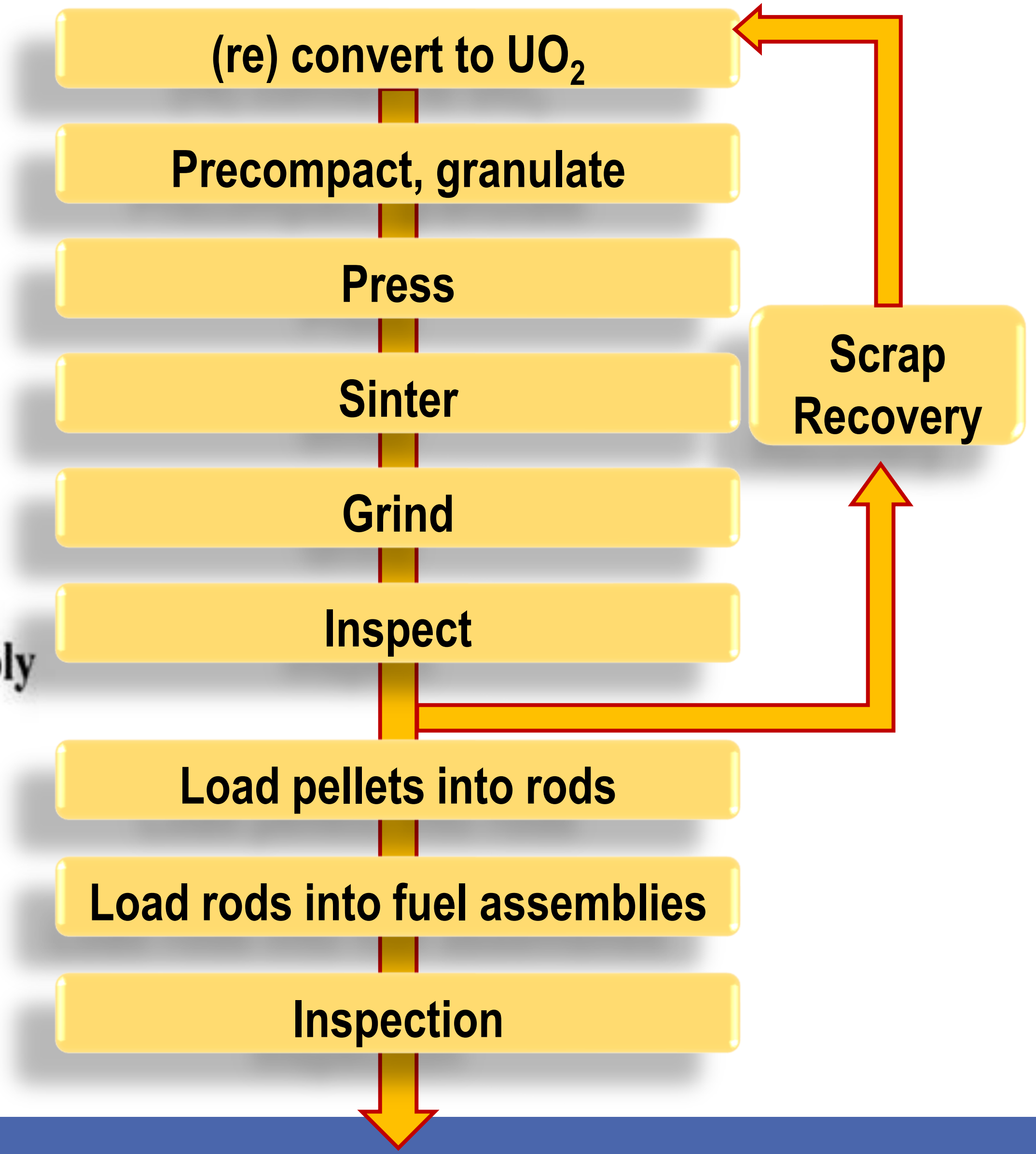
Some steps to Innovations in Nuclear Fuel Cycle Back-end:

- **Development and improvement of technologies for fuel manufacturing with reprocessed U and Pu** brings a complex effect: reduction of natural uranium demands, reduction of SNF storage expenses, reduction of HLW radiotoxicity, etc.
- **MOX (or mixed U-Pu nitride or metal) fuel technology** is a key technological way for closed fuel cycle of fast reactors.

Current technology of Nuclear Fuel Fabrication

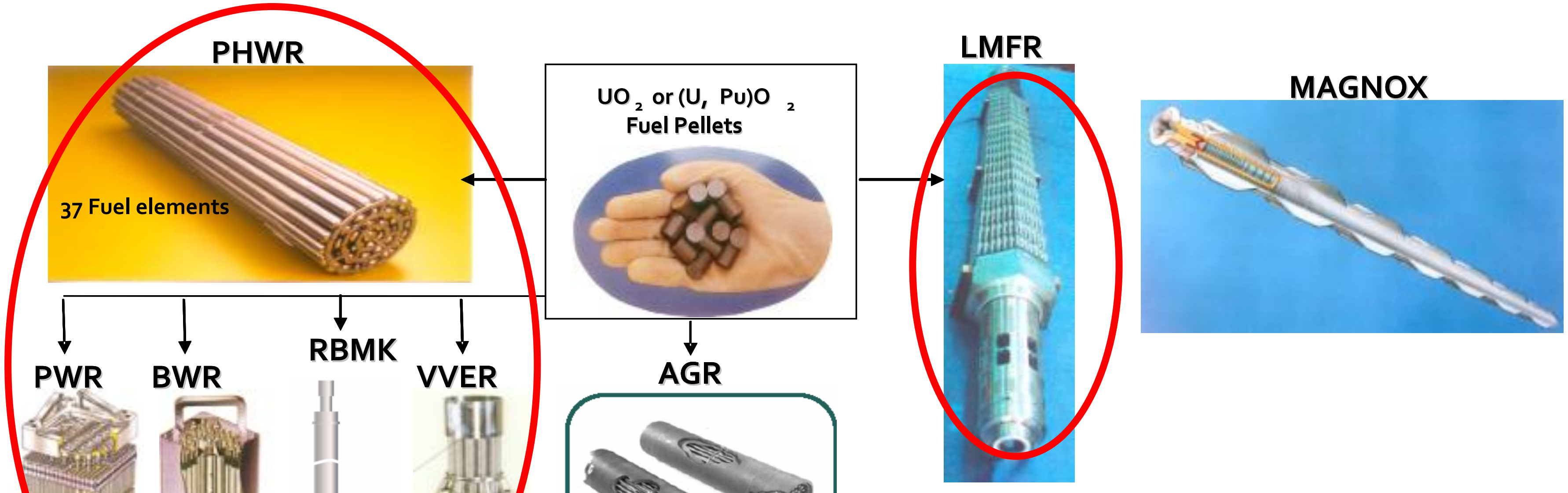
Quality control

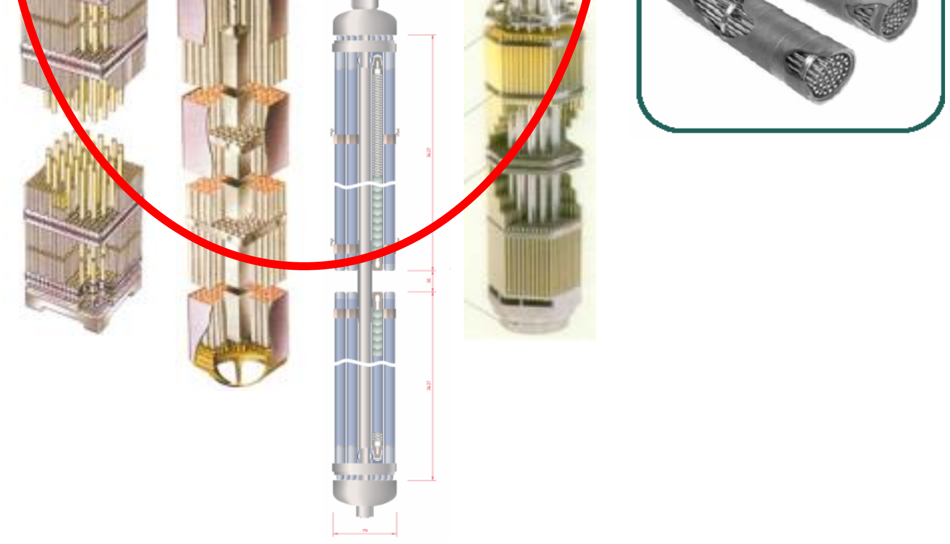




Fuel improvement

Current Nuclear Fuel Fabrication - Improvement





Main lines for fuel improvement:

“Zero Failure” programs

Extension of irradiation period

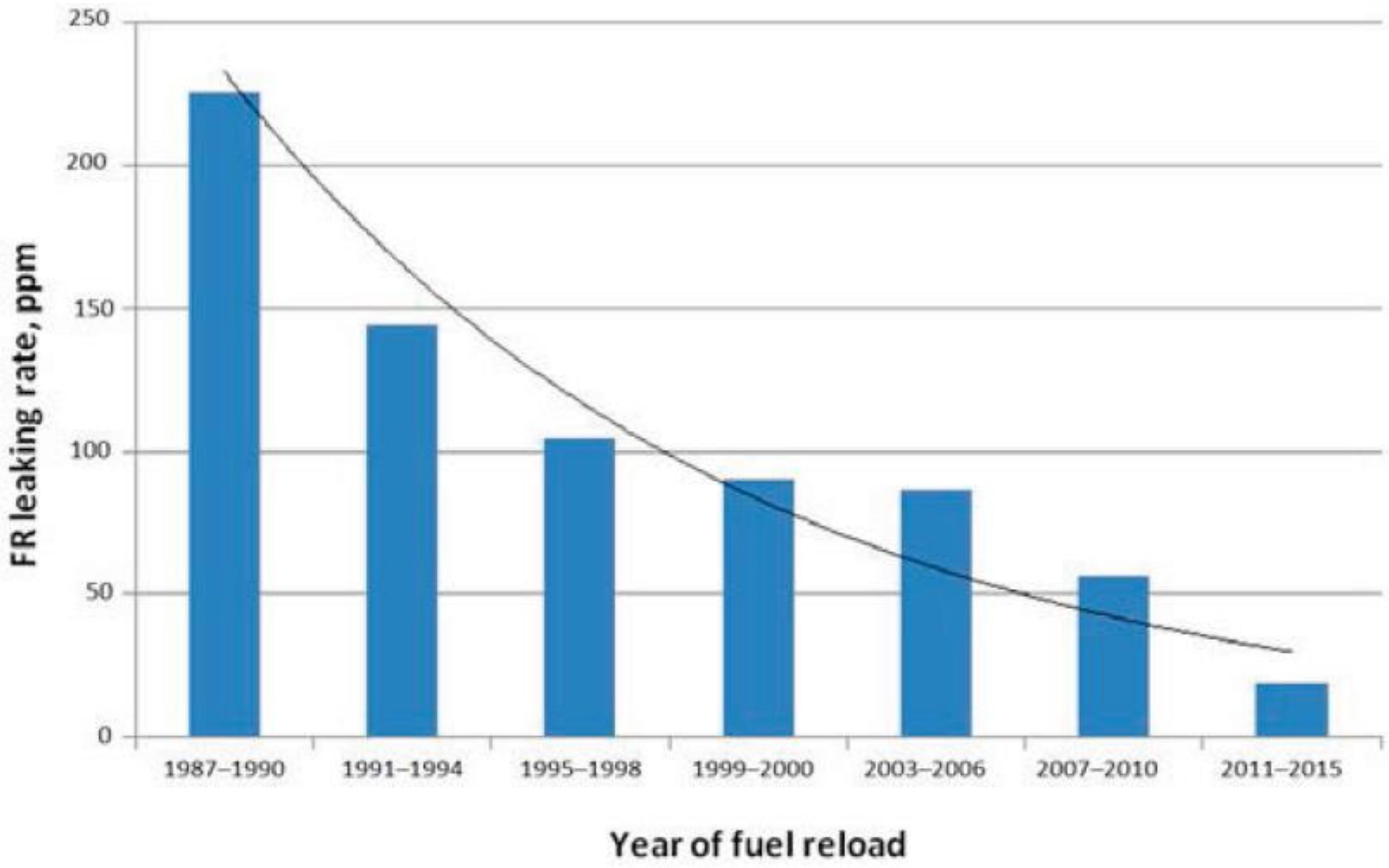
High burn-up

Burning absorber (Gd, Er)

Increased enrichment

Accident Tolerant Fuel

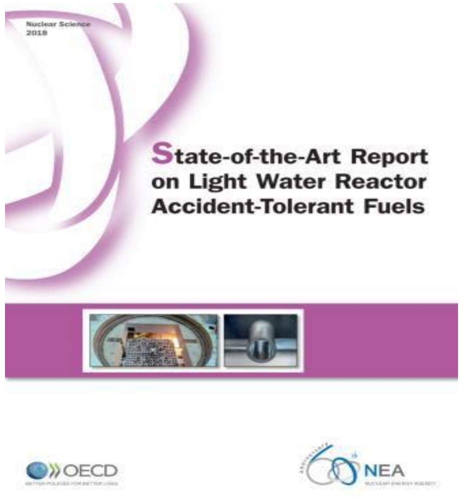
World average PWR fuel rod failure (IAEA data)



Accident Tolerant Fuel (ATF)

Accident at Fukushima-Daiichi in 2011

- Fukushima provided a focus for the industry to develop fuels with enhanced resilience to severe accident scenarios.
- Particular target to extend coping time during a Loss of Coolant Accident.
- Fuel and cladding concepts have been developed that range from *evolutionary* to *revolutionary* in their ambition.



OECD-NEA Report
published in October 2018

372 pages of detailed analysis of concepts

- Deployment potential in existing LWR fleets, new build LWRs and some SMR designs.
- *Revolutionary* concepts might also be applicable to Gen-IV reactors.
- Active irradiation programs are under way in USA and Russia.

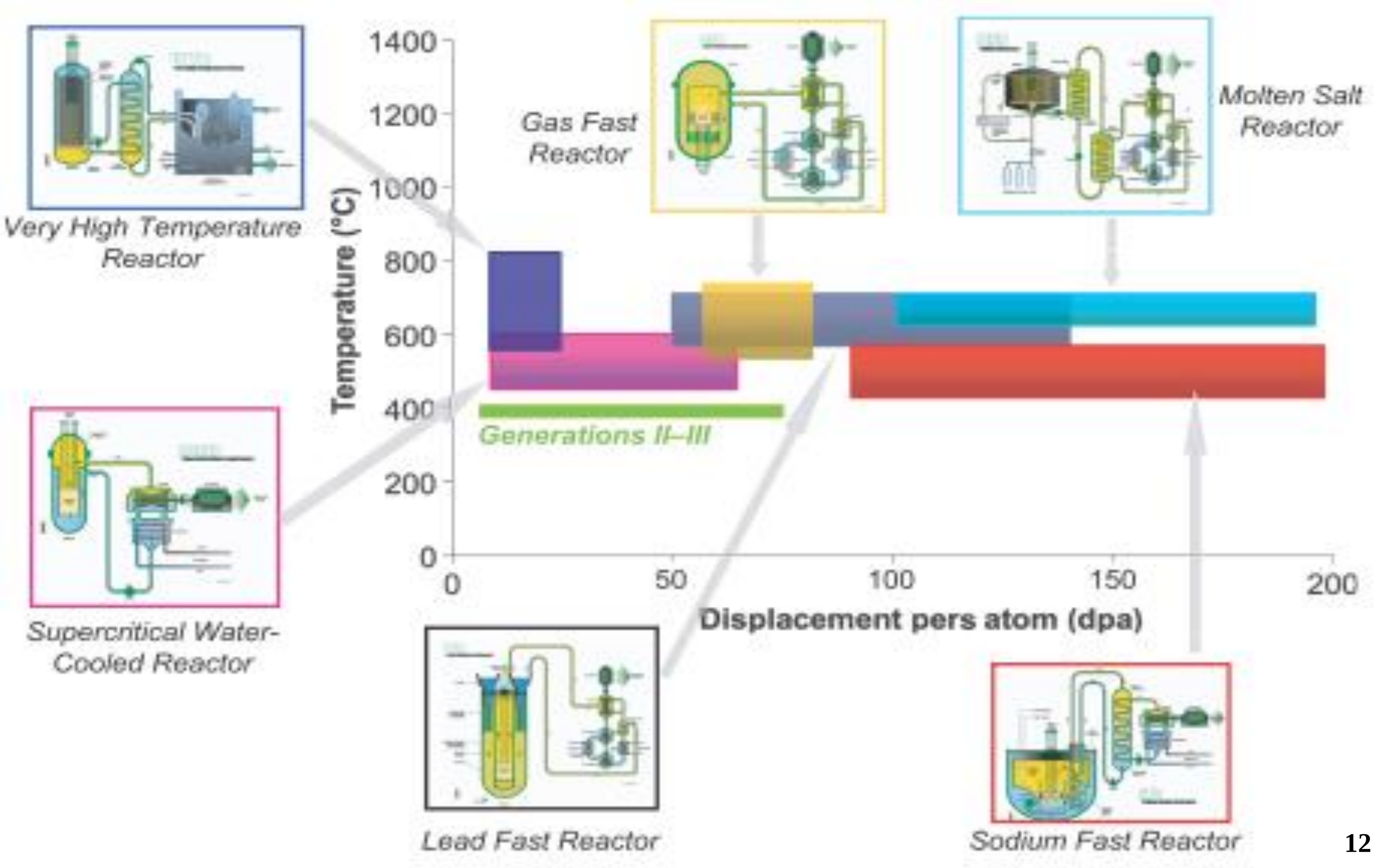
Generation IV reactors and nuclear fuel for them:

requirements, experience and tendencies

Comparison of Gen IV systems

System	Neutron Spectrum	Coolant	Outlet temp. (℃)	Fuel cycle	Power (MWe)
Sodium-cooled Fast Reactor (SFR)	Fast	Sodium	500-550	Closed	50-1500
Very-High-Temperature Reactor (VHTR)	Thermal	Helium	750-1000	Open	250-300
Lead-cooled Fast Reactor (LFR)	Fast	Lead	480-570	Closed	20-1200
Supercritical-Water-cooled Reactor (SCWR)	Thermal/ Fast	Water	510-625	Open/ Closed	300-1500
Gas-cooled Fast Reactor (GFR)	Fast	Helium	850	Closed	1200
Molten Salt Reactor (MSR)	Thermal/ Fast	Fluoride salts	700-800	Closed	1000

Irradiation Conditions for Advanced fuels of Gen IV Reactors



Candidate Fuels for GenIV Reactors

Ceramics :

Oxides (single phase or solid solution)
(UO_2 , $UPuO_2$)

Nitrides
(UN, UPuN)

Carbides
(UC, UC_2, UPuC)

Carbonitrides…

Oxicarbides…

Fluorides (salt)

Metals, alloys and intermetallic compounds :

UAl, PuAl, UZr, UPuZr, U_3Si_2, UMo

Cercer or Cermet Composites:

PuO_2-MgO, UO_2-Mo, UO_2-steel

Cercer = Ceramic - Ceramic
Cermet = Ceramic - Metal

IAEA Specific Safety Guide No SSG-52

IAEA Safety Standards
for protecting people and the environment

Design of the Reactor Core for Nuclear Power Plants

Specific Safety Guide
No. SSG-52

IAEA
International Atomic Energy Agency

SSG-52 provides recommendations on the design of the reactor core

Criteria for normal operation and accidental conditions:

01 The fuel elements must accommodate power cycles and meet the design objectives, such as adequate heat transfer, nuclear reactivity, retention of fission products, inherent safety under accident conditions, and retention of structural and mechanical integrity;

02 No cladding melting;

03 No mechanical (or chemically assisted) failures;

04 Manageable H embrittling effect: assure manipulation of irradiated material.

SSG-52 postulated aspects to be addressed in the design of fuel rods

Aspects to be addressed in the design of the fuel rod

Cladding:

01 Fuel rod vibration and wear (i.e., grid-to-rod fretting wear for light water reactors);

02 Evolution of the mechanical properties of the cladding with irradiation (displacement and pressure driven loadings);

03 Materials and chemical evaluation;

04 Stress corrosion cracking;

05 Cycling and fatigue;

06 Geometrical and chemical stability of the cladding under irradiation.

Aspects to be addressed in the design of the fuel rod

Fuel material (including burnable absorbers):

01 Dimensional stability of the fuel under irradiation;

02 Fuel densification (kinetics and amplitude);

03 Potential for chemical interaction with the cladding and the coolant;

04 Fission gas generation and distribution within the fuel pellets;

05 Fission gas release kinetics;

06 Gaseous swelling;

07 Thermomechanical properties under irradiation;

08 Microstructure changes as a function of irradiation.

SSG-52 postulated aspects to be addressed in:

Aspects to be addressed in the **Fuel rod performance:**

01 Pellet and cladding temperatures and temperature distributions;

02 Fuel-cladding gap closure kinetics and amplitude (to address issues relating to pellet-cladding interactions);

03 Irradiation effects on fuel rod behavior (e.g., fuel restructuring, cracking of fuel pellets, solid and gaseous fission product swelling, fission gas release and increases in internal pressure of fuel rods, degradation of thermal conductivity of fuel rods);

04 Fuel rod bowing;

05 Fuel rod growth.

Fuel assembly components (e.g., top and bottom nozzles, guide tubes, spacers, mixing grids, grid springs, connections and fuel assembly hold-down systems for pressurized water reactors) need to be designed to withstand the following conditions and loads:

01 Core restraint system loads;

02 Hydrodynamic loads;

03 Thermohydraulic limits (e.g., critical heat flux);

04 Accident loads (e.g., loss of coolant accident) and seismic loads;

05 Handling and shipping loads;

06 Fuel assembly bowing.

SSG-52 has recommendations on selecting the fuel pellet materials and cladding materials:

In selecting the **fuel pellet materials**, the following properties should be optimized:

01 Reactivity with thermal neutrons;

02 Impurities with low thermal neutron absorption properties;

03 Thermal performance (e.g., high thermal conductivity is desirable for operational states while high thermal diffusivity is desirable for accident conditions);

04 Dimensional stability;

05 Fission gas retention;

06 Resistance to pellet-cladding interaction.

Cladding materials should be selected with consideration of the following properties

01 Low absorption cross-section for thermal neutrons;

02 High resistance to irradiation conditions;

03 High thermal conductivity and high melting point;

04 High corrosion resistance and low hydrogen pick-up;

05 Low oxidation and low hydriding in high temperature conditions;

06 Adequate resistance to breakaway oxidation at high integrated-time temperature conditions;

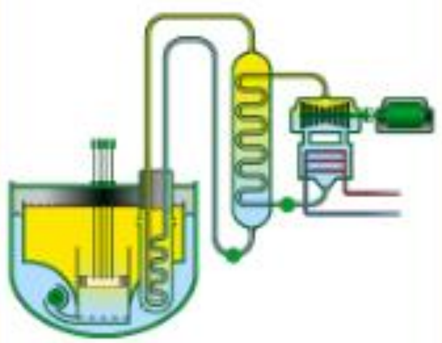
07 Adequate mechanical properties (e.g., high strength, high ductility, low creep rate in normal operation, high relaxation rate in transients);

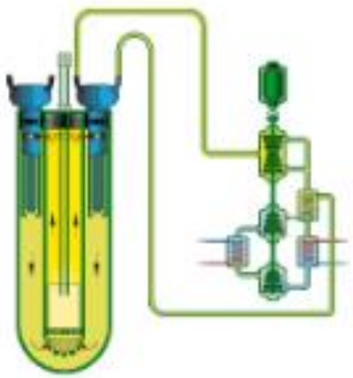
08 Low susceptibility to stress corrosion cracking;

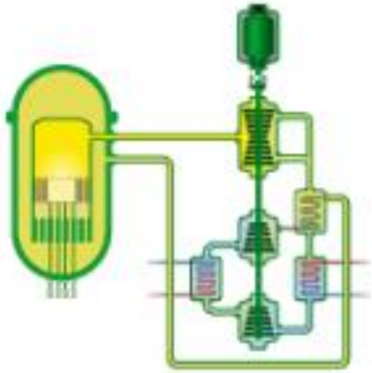
09 Adequate resistance to hydrogen assisted cracking and hydride related cracking in normal operation and for fuel storage.

Nuclear Fuel for Fast Reactors

Fuel assembly of SFR BN-800







	Sodium-cooled Fast Reactor (SFR)	Lead-cooled Fast Reactor (LFR)	Gas-cooled Fast Reactor (GFR)
Fuel material	$UPuO_2$, UPuZr, UPuN, UPuC	$UPuO_2$, UPuN	$UPuO_2$, UPuC
Fuel and fuel element form	Pellet (vibropack), metal rod	Pellet (vibropack)	Plate, Pin

Main characteristics of fast reactor fuels

Comparison of fast reactor fuels	Metal (U-20Pu-10Zr)	Oxide (UO_2-20PuO_2)	Nitride UN-20PuN	Carbide UC-20PuC
Heavy metal density, g/cm^3	14.1	9.3	13.1	12.4
Melting temperature, K	1,350	3,000	3,035 (dec.)	2,575
Thermal conductivity, W/cm-K	0.16	0.023	0.26	0.2
Operating centerline temperature at 40 kW/m, K, and (T/T_{melt})	1,060 (0.8)	2,360 (0.8)	1,000 (0.3)	1,030 (0.4)
Fuel-cladding solidus, °C	675	1,675	1,400	1,390
Thermal expansion, 1/K	17E-6	12E-6	10E-6	12E-6
Fuel/cladding chemical interaction	–	+	+	–/+
Fuel/cladding mechanical interaction	+	+	–	–
Fuel/coolant compatibility	+	–	+	+
Fuel swelling	–	–	+	+
Reprocessing amenability	Pyro-processing demonstrated on pilot plant scale	Demonstrated on industrial scale for aqueous and pilot scale for pyro-processes	Demonstrated on lab scale for aqueous and pyro-processes	Demonstrated on lab scale

Key features of fast reactors fuels

Oxide / UO_2-PuO_2
- Low thermal conductivity
- Low density of fissile atoms
- No reaction with sodium or lead
- Well-known in all the main countries

Metal / U-20Pu-10Zr
- Very high thermal conductivity
- High swelling
- Melts at relatively low 1160 C
- Not compatible with lead coolant
- Researched: USA, Japan, China, Russia

Nitride / UN-PuN
- High thermal conductivity
- High density of fissile atoms
- Subject to swelling
- C-14 contamination (pure N-15 is needed)
- Researched: Russia, USA and Japan

Carbide / UC-PuC
- High thermal conductivity
- High density of fissile atoms
- High swelling
- Poor compatibility with air and water
- Researched: India

SFR Fuels: history and current status

UOX-HEU or HALEU – industrial experience in past and present

MOX - Pilot manufacturing: UK, USA, France, Belgium, USSR – 1960-1990, Japan – 1990s
MOX – industrial production: France 1980-1990, Russia – 2020-x

Metal fuel – pilot manufacturing: USA 1960-x for U and U-Zr fuel,

Nitride furl – Russia: preparation of pilot manufacturing

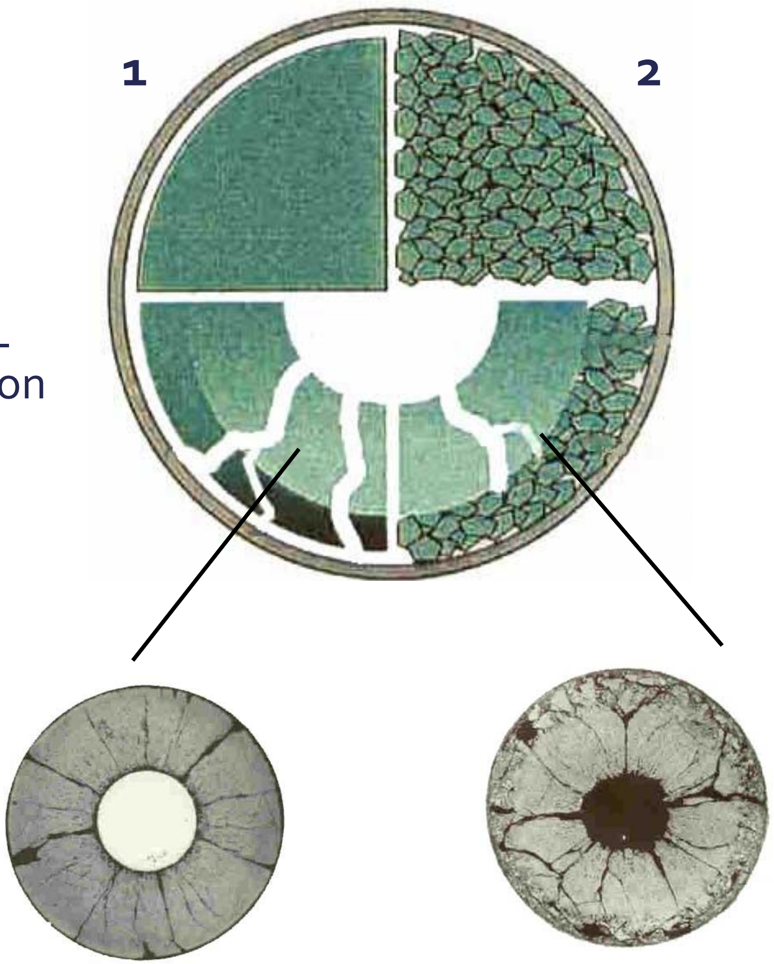
Carbide – large lab-scale: USSR - 1960-70, India from 1970-x

Vibropacking for Fast Reactor Oxide Fuel

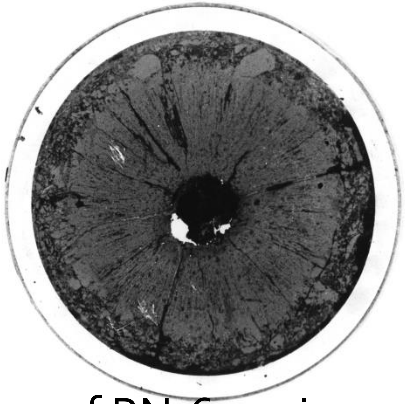
Macrostructure of pelletizing (1) and vibropacking (2) MOX fuel for fast reactors

Before irradiation

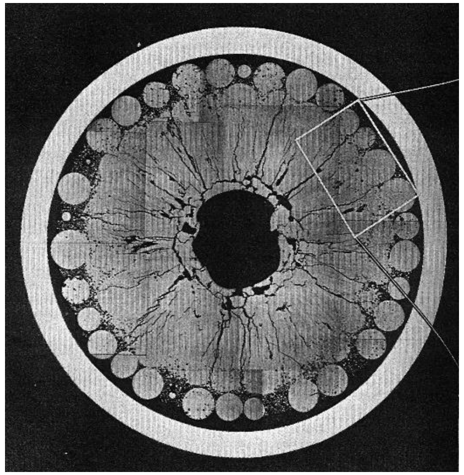
After Irradiation



Vibropacking technology: Integration with spent fuel recycling by pyro-process for MOX fuel. High potential for fuel with low decontamination and with MA

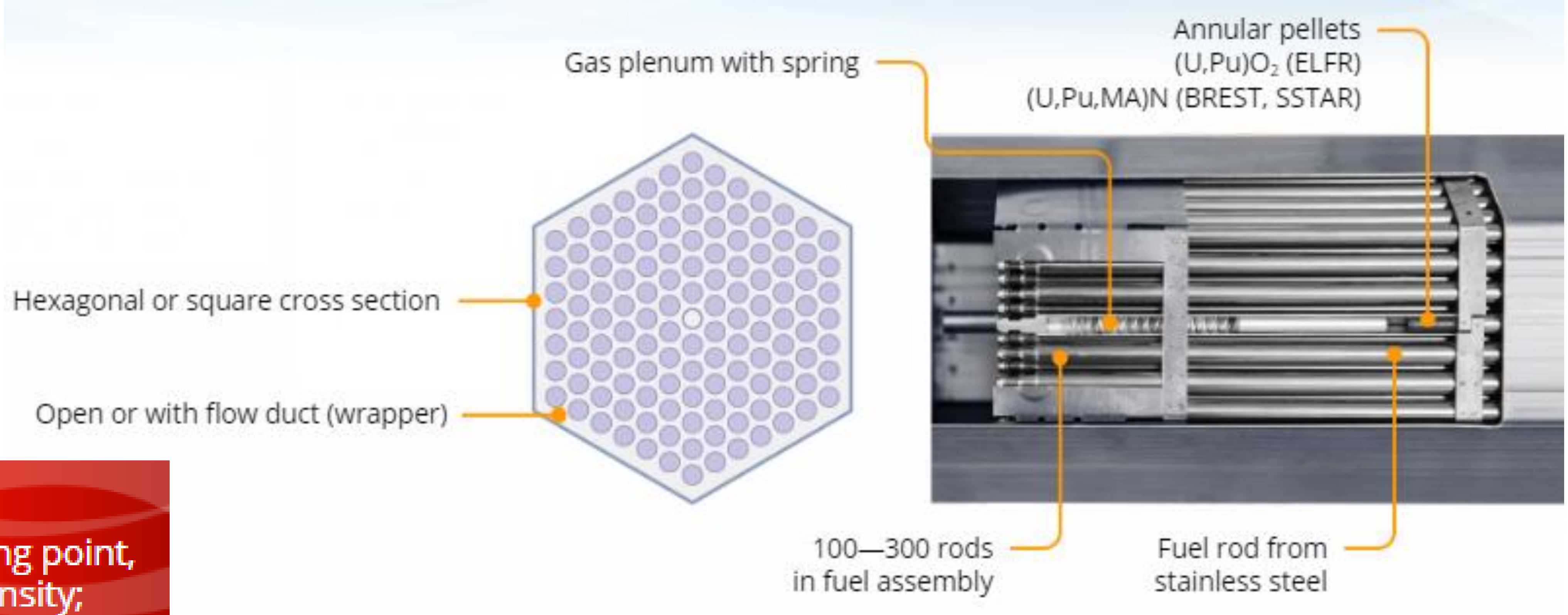


Cross-cutting of BN-600 pin



Cross-cutting of TREAT pin (ORNL)

Nuclear fuel for LFR



R&D focus

- Lead high melting point, opacity, high density;
- Materials corrosion;
- High burnup, MA-bearing fuels.

Russia has intensive manufacturing and irradiation program for (U,Pu)N fuel. Manufacturing facility is under construction.

Nuclear Fuel for GFR

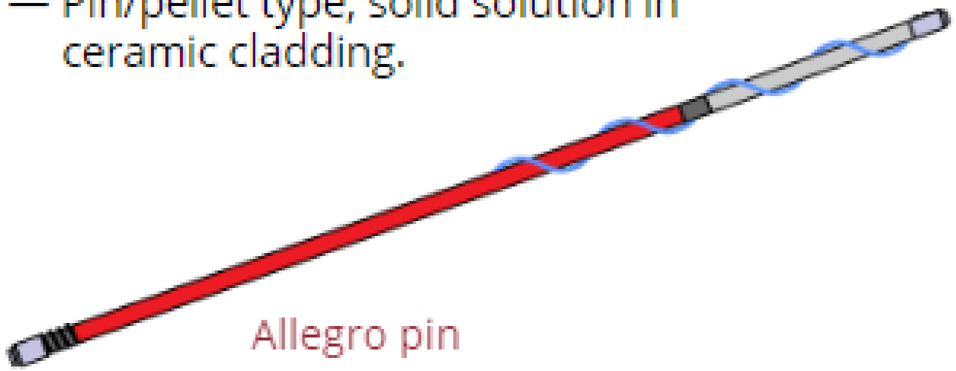
GFR pin

Candidate fuel for GFR

- Carbide preferred to nitride for its neutronic properties (^{15}N enrichment needed);
- Both have relatively high volatility;
- Oxide back-up but with lower core performance;
- Metallic fuel discarded due to low melting point.

Fuel for Allegro project (concept)

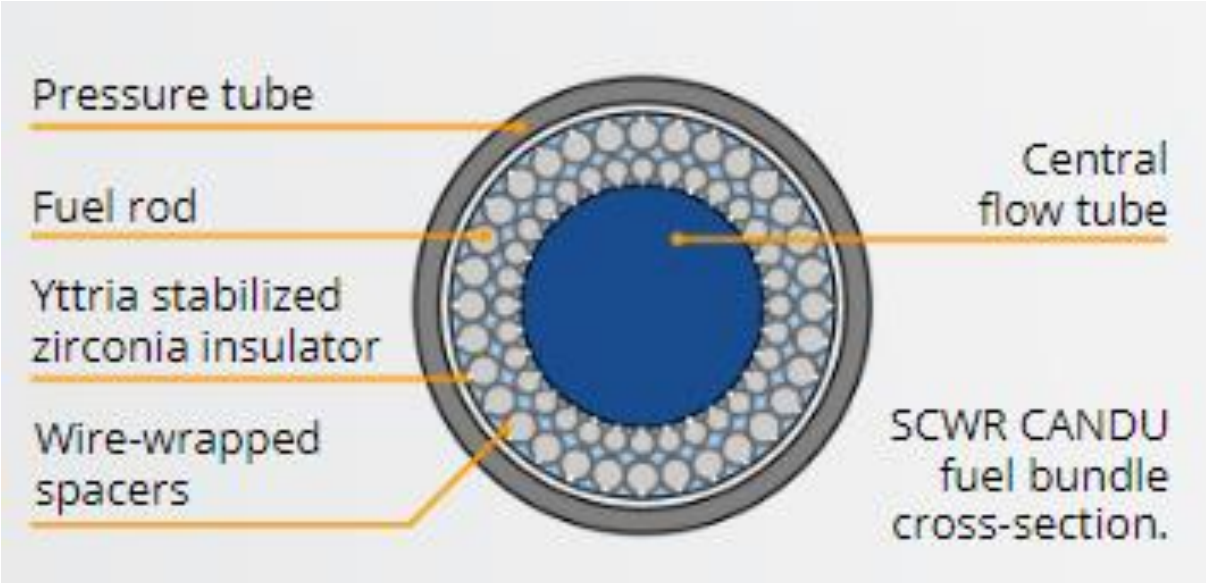
- UOX and MOX pellets in steel cladding (Allegro start-up core);
- Pin/pellet type, solid solution in ceramic cladding.



Allegro pin

Nuclear Fuel for SCWR

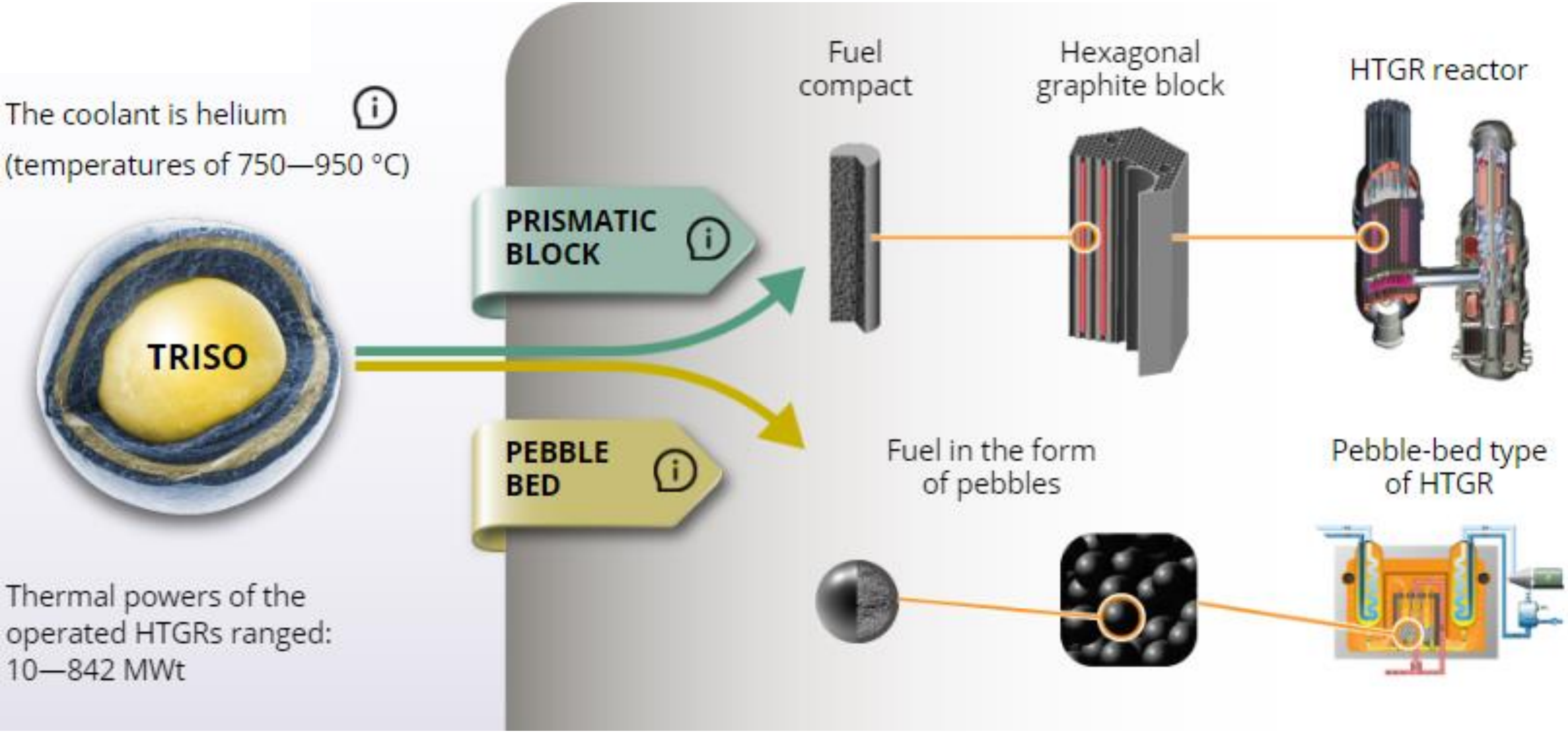
Example of concept design for SCWR fuel assembly



Fuel: UO_2 (5—7%) in a once-through NFC.
MOX fuel in a closed NFC.

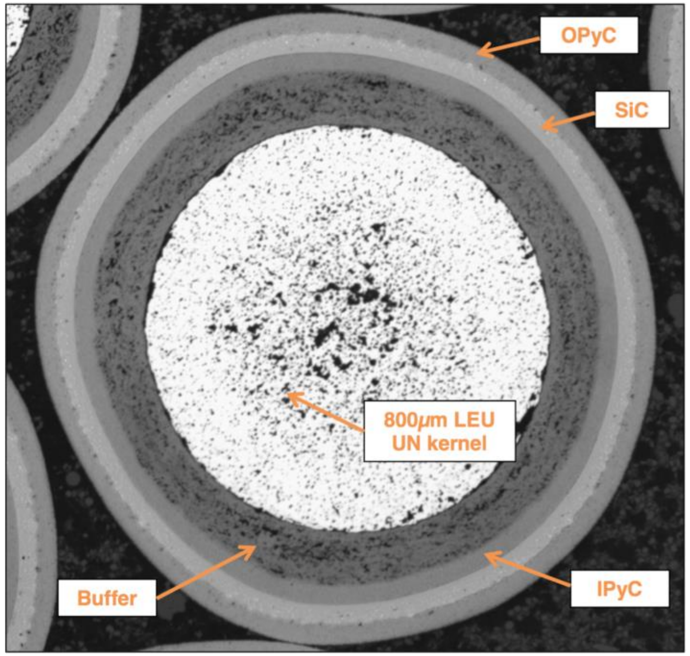
Prospective fuel:
Th and Pu (13%) mixed fuel

Nuclear Fuel for VHTR (HTGR)



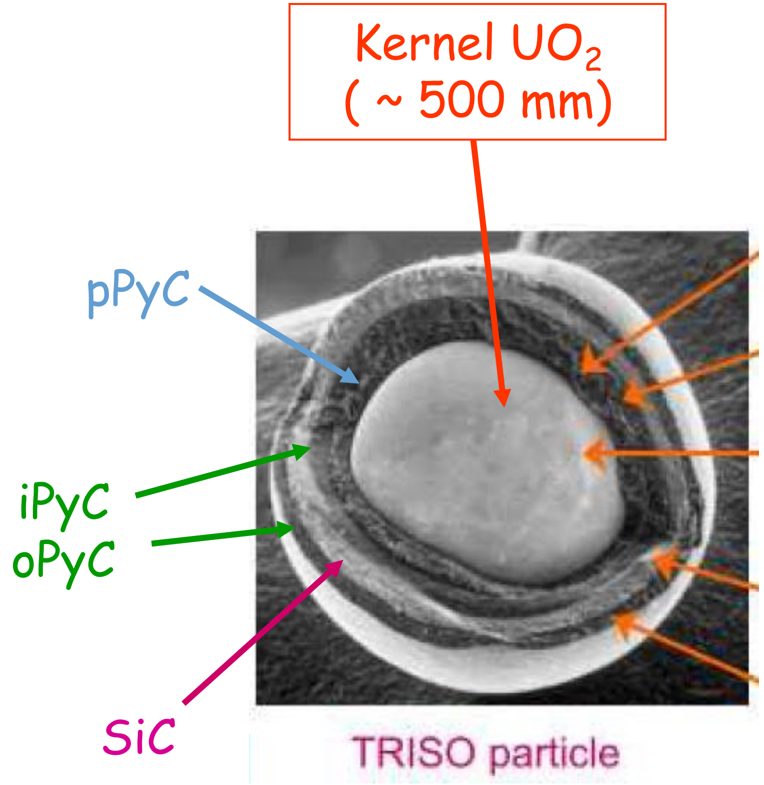
UO_2 and UN TRISO Fuel Particle
(for HTGR and as ATF for other reactors)

ATF - LEU UN TRISO particle



Optical cross sectional image of typical 800μm LEU UN TRISO particle

"Classical" UO2 TRISO particle



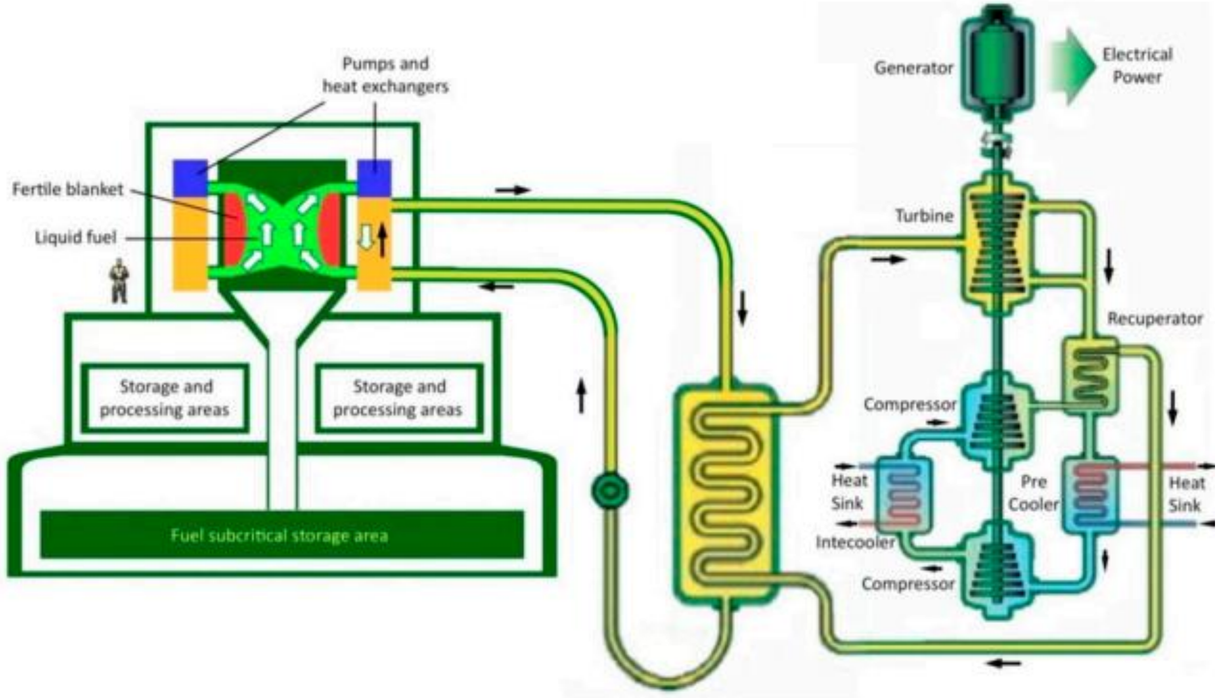
Molten Salt Reactors

DESIGN OPTIONS

— **Fuel dissolved in molten salt coolant** (traditional concept with on-line waste management);

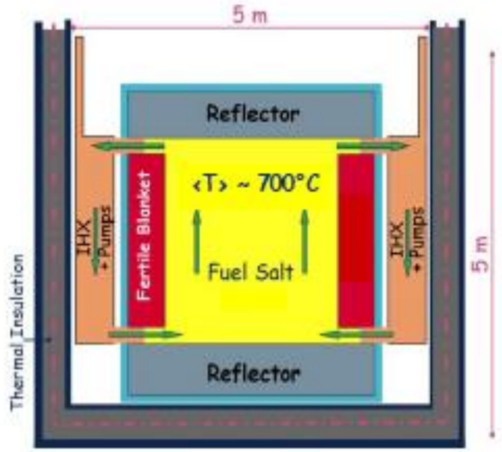
— **Solid fuel with molten salt coolant** (typical design for VHTR).

Nuclear reactor and Closed fuel cycle in "one package"



Molten Salt reactor:
Reactor and Fuel Cycle Facility as one unit

- MSRs: systems with liquid fuel, molten fluoride (chloride) salts, circulating in primary circuit
- Actinide-free molten salt in secondary circuit
- Online fuel reprocessing instead of fuel fabrication -(partial removal of fission products and correction of actinides content)
- Operation either as breeders (Th-U or U-Pu cycle), as nuclear waste incinerators (transmuters)
- Thermal (graphite moderator) or Fast neutron spectrum
- Typical fuel : Flourides of actinides dissolved in a carrier salt, such as $^{7}LiF\text{-}BeF_2$ or chlorides of actinides dissolved in a carrier salt, such as $LiCl\text{-}NaC\text{-}MgCl_2$.



Molten Salt Fuels and Coolants

REACTOR TYPE	NEUTRON SPECTRUM	MOLTEN SALT APPLICATION	REFERENCE SALT SYSTEMS
Molten Salt Breeder Reactor	Thermal	Fuel	$LiF-BeF_2-AnF_4$
	Fast	Secondary coolant	$NaF-NaBF_4$
		Fuel	$LiF-AnF_4$
			$NaCl-MgCl_2-UCl_3-PuCl_3$
			$LiF-NaF-BeF_2-AnF_3$
Advanced High Temperature Reactor	Thermal	Primary coolant	$LiF-BeF_2$
Very High Temperature Reactor	Thermal	Heat transfer coolant	LiF-NaF-KF
Liquid Salt Cooled Fast Reactor	Fast	Primary coolant	$LiCl-NaCl-MgCl_2$
		Intermediate coolant	$NaNO_3-KNO_3$

MSR Fuel: R&D needs

Research & Development

Potentially useful collaborative projects include:

— Measurement of salt thermochemical and thermophysical properties;

— Performance of integral and separate effect tests to validate safety performance;

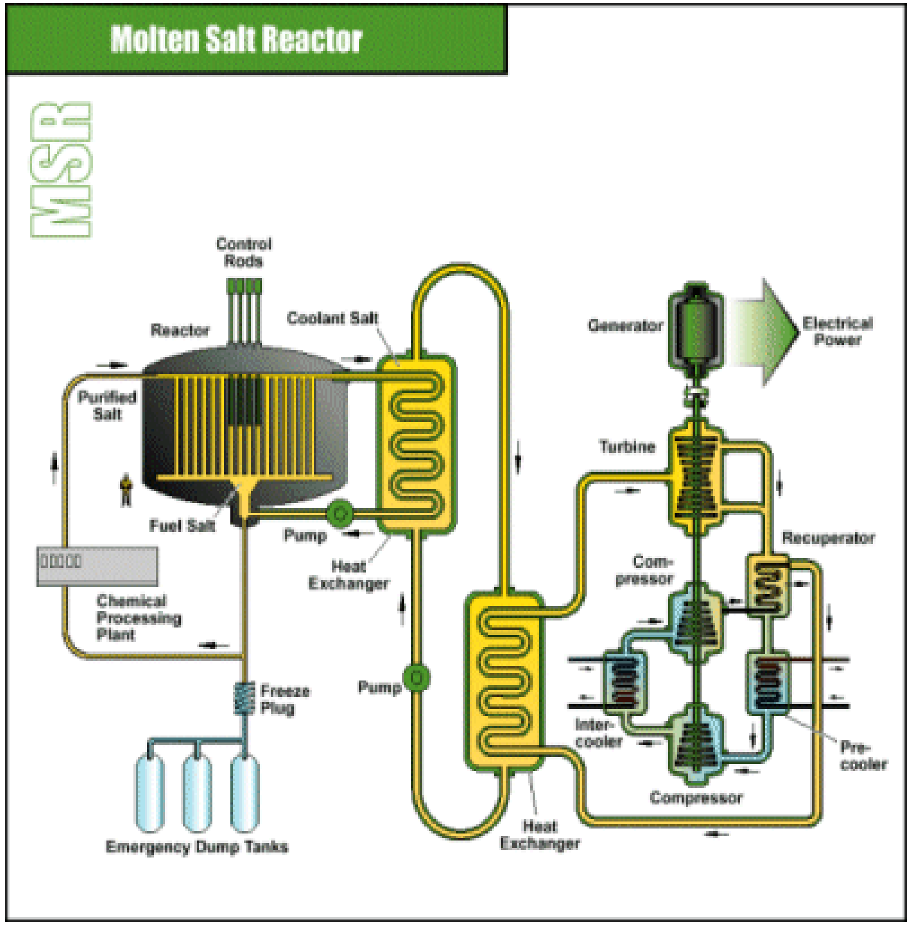
— Development of improved neutronic and thermal-hydraulic models and tools;

— Study of materials issues associated with use at molten salt reactors (e.g. erosion, corrosion, radiation damage, creep-fatigue);

— Demonstration of tritium management technologies;

— Salt redox control technologies to master corrosion of the primary fuel circuit and other components;

— Demonstration of surveillance and maintenance technologies for high radiation areas, such as molten salt reactor containments.



IAEA activity on fuel of advance reactors

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Module 3: Nuclear fuel for advance reactors

This module consists of the 4 lectures and provides the overview on Generation IV reactors and SMRs goals and requirements; basic knowledge on nuclear fuel for Advanced Reactors, including general description, design parameters and specifications, fuel manufacturing, fuel behavior under irradiation, fuel performance assessment and fuel performance modelling.

Learning objectives

The objective of these e-learning module is to provide a high-level guidance in taking a systematic approach to advance reactors nuclear fuel design, fabrication, and nuclear fuel behaviour during irradiation.

Duration: 45 min per each lecture (around 180 min for module)

Language availability: English

Target Audience for the e-Learnings: students, young professional, regulators, operators, high level government officials

Prepared by NE/NFCMS
Anzhelika Khaperskaia

Options for fuel recycling in Gen IV systems

System	Fuel cycle	Fuel type	Reprocessing method	Fuel refabrication
Sodium-cooled Fast Reactor (SFR)	Closed (U-Pu-MA)	Ceramic/ Metallic	Aqueous (for oxides mainly), pyroprocess	Pellets, vibro-compating, injection melting
Very-High-Temperature Reactor (VHTR)	Open (U, U-Pu, Th-U)	Ceramic / TRISO	(complicated pyroprocess)	(TRISO)
Lead-cooled Fast Reactor (LFR)	Closed (U-Pu-MA)	Ceramic	Aqueous / pyroprocess	Pellets
Supercritical-Water-cooled Reactor (SCWR)	Open/ Closed	Ceramic (oxides)	Aqueous (pyroprocess)	Pellets (vibro)
Gas-cooled Fast Reactor (GFR)	Closed (U-Pu-MA)	Ceramic	Aqueous (for oxides mainly). Pyroprocess	Pellets or others
Molten Salt Reactor (MSR)	Closed (U-Pu-MA, U-Th-MA)	Fluoride / chloride salts	Pyroprocess on-site	

Reprocessing and recycling of GenIV fuel

Current industrial reprocessing based on PUREX-process.

- **Advanced methods and technologies:**
  - Advanced processes based on aqueous extraction or precipitation
  - Advanced methods for MA recycling
  - Other low temperature methods
- **High temperature recycling process**
  - Pyro-process in molten salts (chlorides, mainly)
  - Fluoride volatility process (removal of UF6 from spent fuel)
  - Oxidation-reduction methods (ex. DUPIC)
- **Combination of pyro-process and aqueous methods**

Spent fuel reprocessing – PUREX is the industrially applied aqueous process

Head End Mechanical Process

Chemical Separation

Product Purification

PuO_2

Rep UO_3

Waste metallic structural components (e.g. hulls) for treatment & conditioning

Vitrification of Fission Products & Minor Actinides

In addition:
Acid recovery
Solvent Treatment
Off-gas treatment
Liquid effluent management

Advanced Spent Fuel Reprocessing Methods

Advanced Aqueous Partitioning Methods

GANEX
Uextraction followed by group extraction of all actinides

UREX +
- Separation of U & Tc by UREX
- Recovery Cs & Sr by CCD-PEG
- Recovery of Pu & Np by NPEX
- Recovery of Am, Cm and Ln by TRUEX
- Separation of Am and Cm from Ln

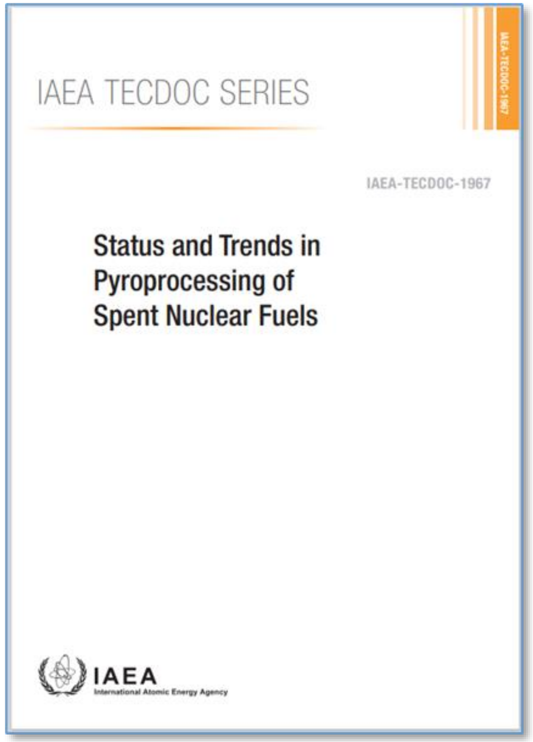
DIAMEX, TODGA
Separation of MA and Ln from HLLW

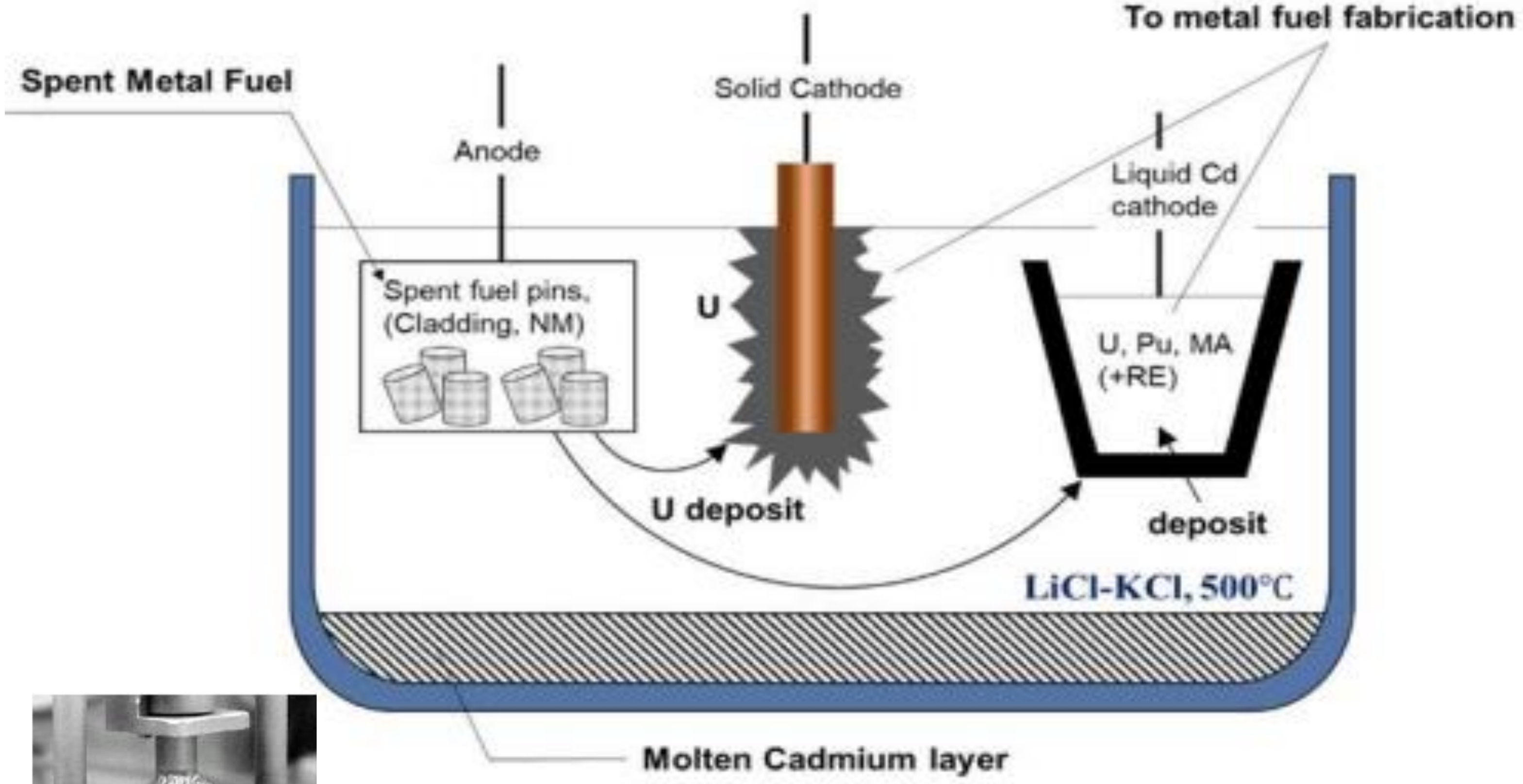
TRUEX
TRU elements extraction from HLLW

SANEX, ARTIST, TALSPEAK
Separation of Am, Cm from Ln

SESAME
Separation of Am from Cm

Advanced Pyro-process Options



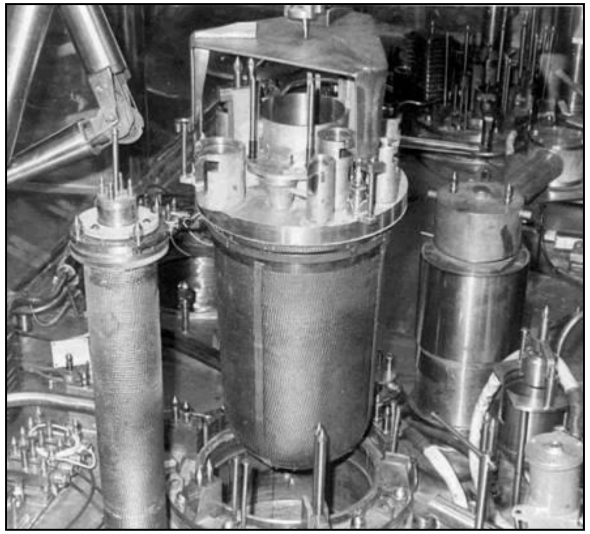


The process for metallic U-Pu-Zr and nitride (U,Pu)N fuel

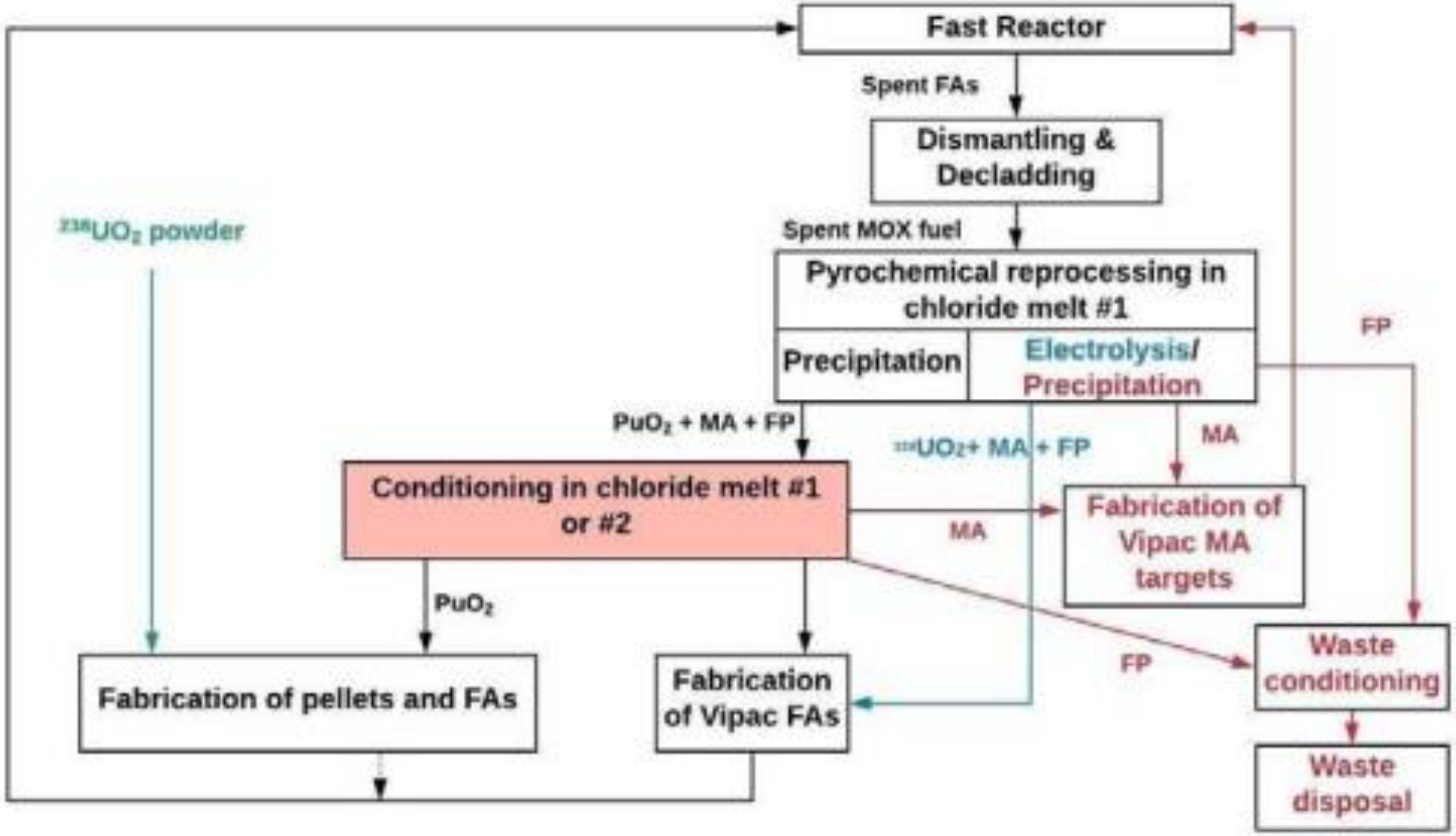
U deposit

U, MA and Pu in a liquid Cd cathode

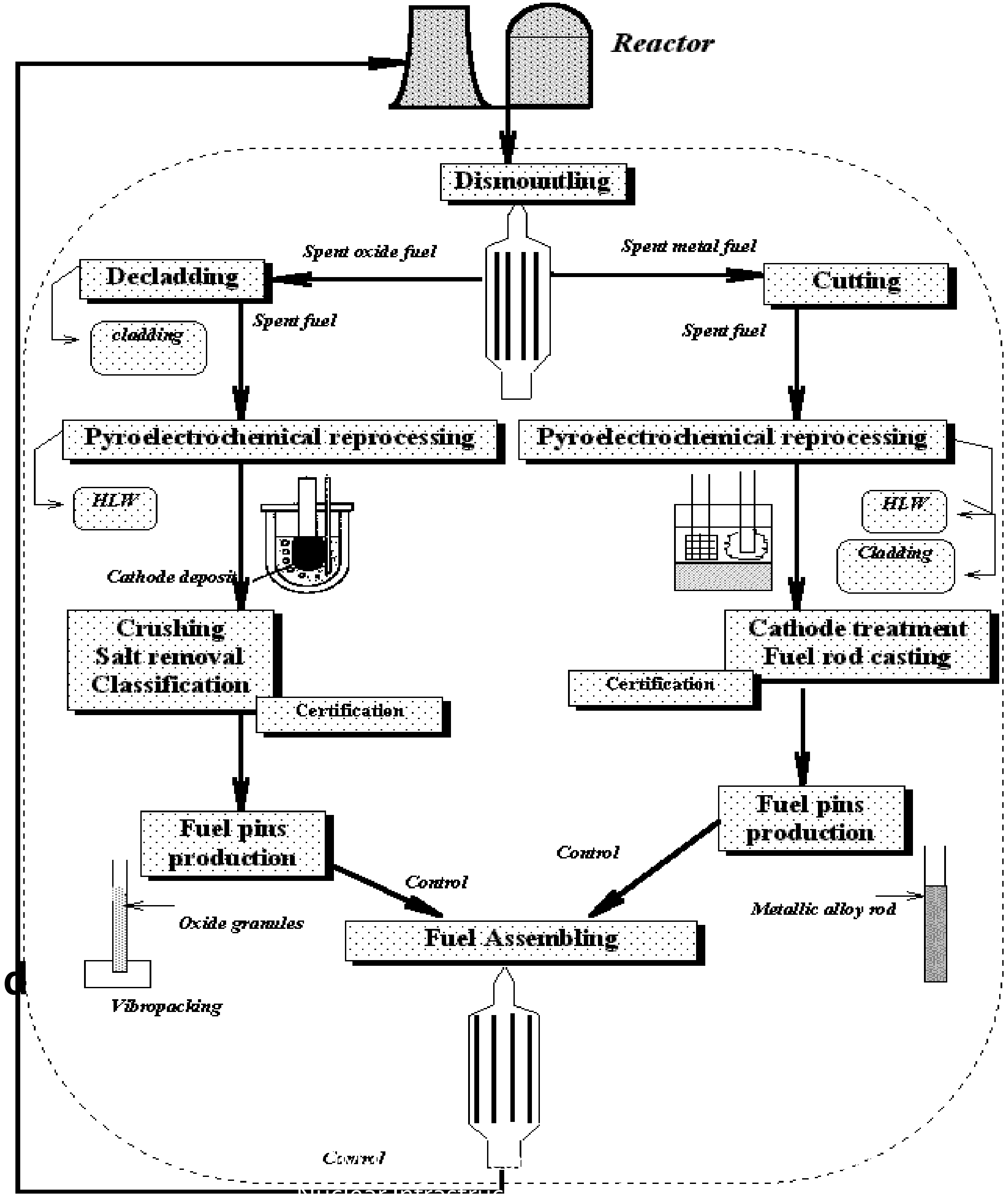
Production and recycling of FR MOX fuel by pyro-process and vibropacking



Russia has experience on pilot manufacturing and recycling of MOX fuel.
Remote controlled equipment lines tested.
RIAR, Dimitrovgrad, from 1970 -2010-s



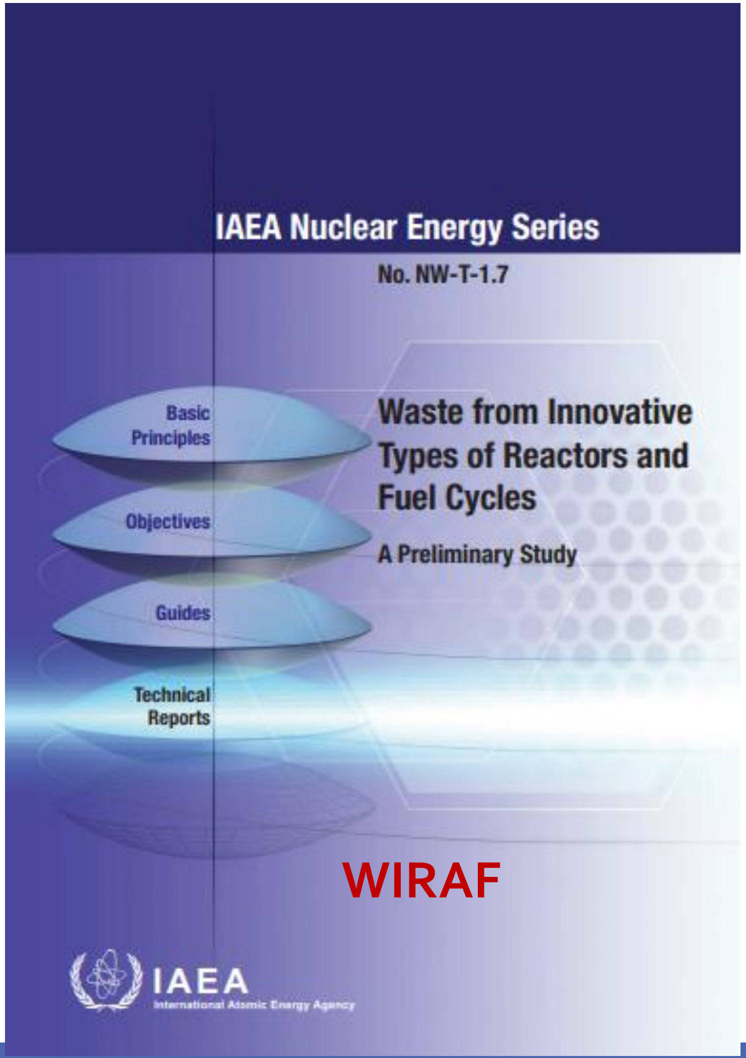
Old author's diagrams (early 1990-s) for comparison of MOX recycling by pyro/vibro (RIAR, Russia) and U-Pu-Zr fuel recycling by pyro/injection casting (ANL, USA)



MOX fuel recycling for Fast reactor (Dimitrovgrad Dry Process)

UPuZr fuel pyro-recycling for the Integral Fast Reactor Concept developed by ANL/INL, USA

Consideration of reprocessing options for spent fuel of advanced reactors by INPRO



INPRO is IAEA International Project on **Innovative** Nuclear Reactors and Fuel Cycles

A number of INPRO collaborative studies considered different options of Spent Nuclear Fuel management and recycling as key element of future sustainable nuclear energy programs.

Examples: Innovative fuel fabrication technologies for fast breeder reactors

WIRAF

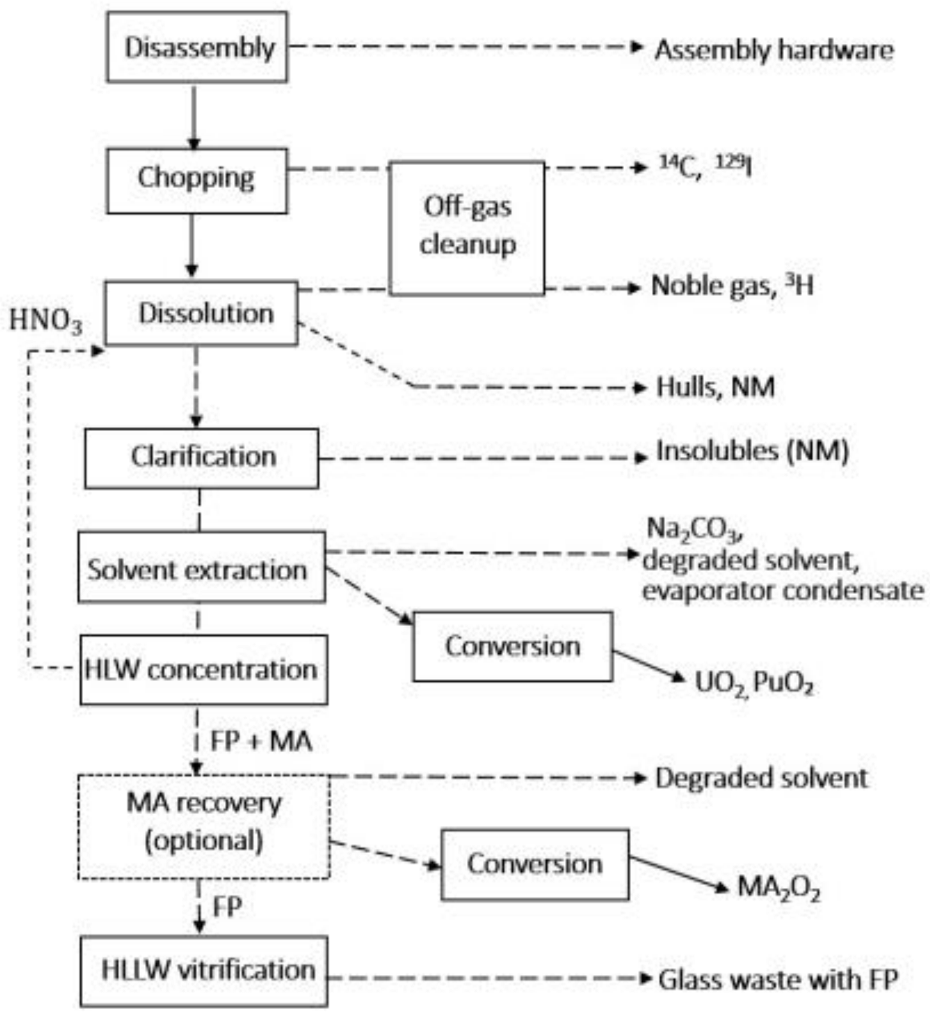
Reprocessing	Simplified PUREX process	Alternative aqueous process	Oxide-electrowinning process	Metal-electrorefining process
	Dissolution	Direct extraction		Oxide reduction
U Recovery	Crystallization	(Supercritical fluid extraction TBP-CO2-HNO3)	U Electrowinning	U electrorefining (solid cathode)
U, Pu, MA recovery	Single cycle extraction		MOX electro-codeposition	U, Pu, MA electrorefining (liq. Cd cathode)
MA recovery	SETFICS process /TRUEX process	Ion exchange/amine extraction	MA Electrowinning	Cd extraction (pyro contactor)

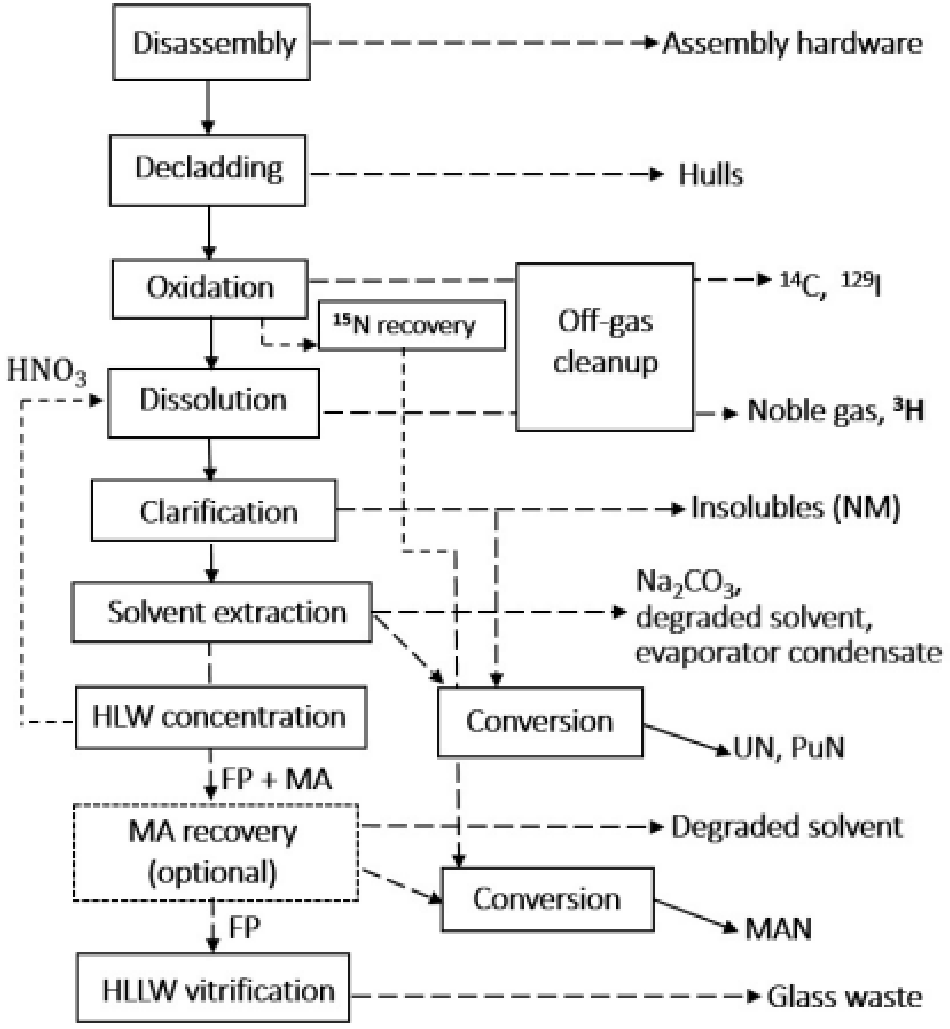
Fuel refabrication	Pellet short process	Sphere packing process	Vibro-packing process	Casting process
	Simplified pelletizing	Gelation (MOX-MA)	Granulation (MOX, MN-MA)	Casting (U-Pu-Zr-MA)
	Stacking	Vibration packing	Vibration packing	Stacking
	MOX fuel		MOX fuel	Metal fuel

: MOX aqueous recycle   : MOX non-aqueous recycle   : Metal non-aqueous recycle

Process flows and primary waste streams for aqueous reprocessing of FR oxide and nitride SNF

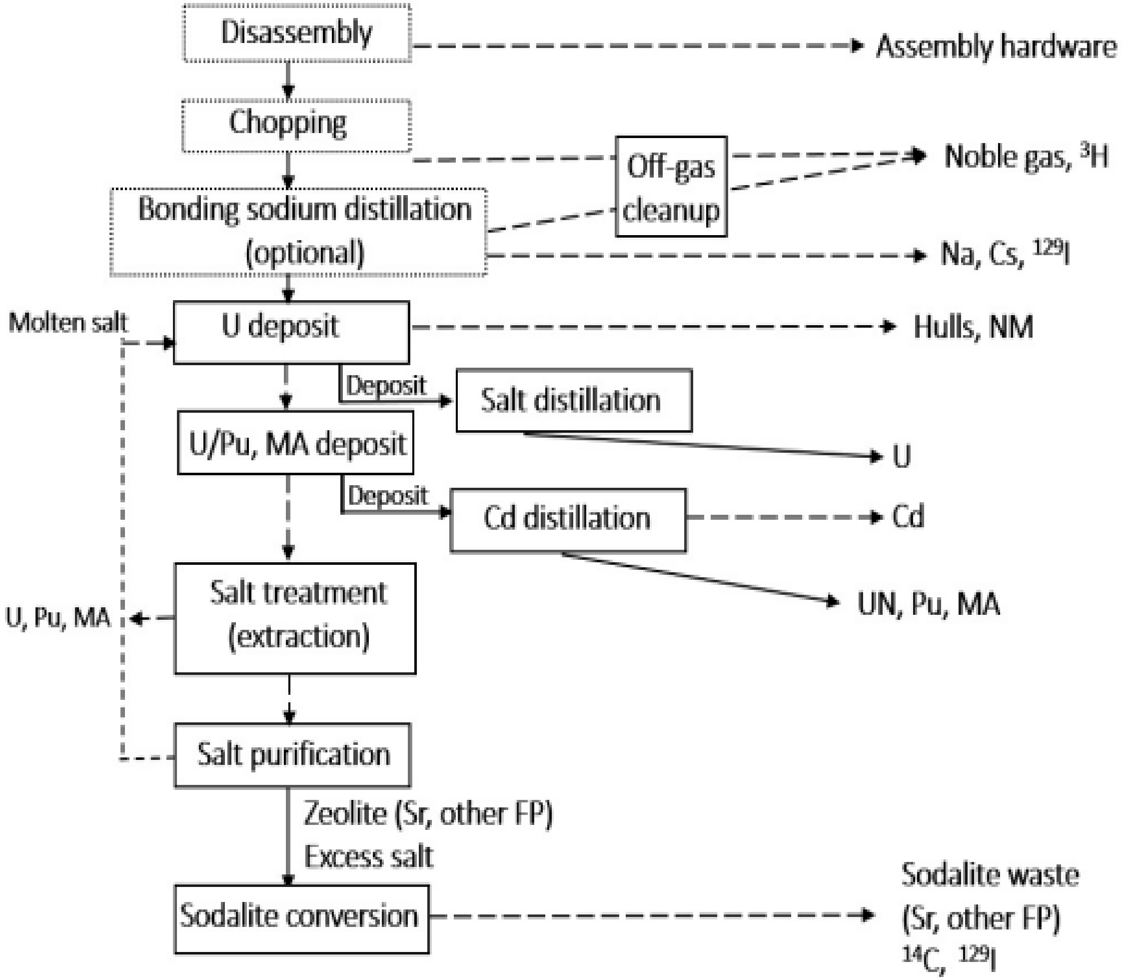
WIRAF

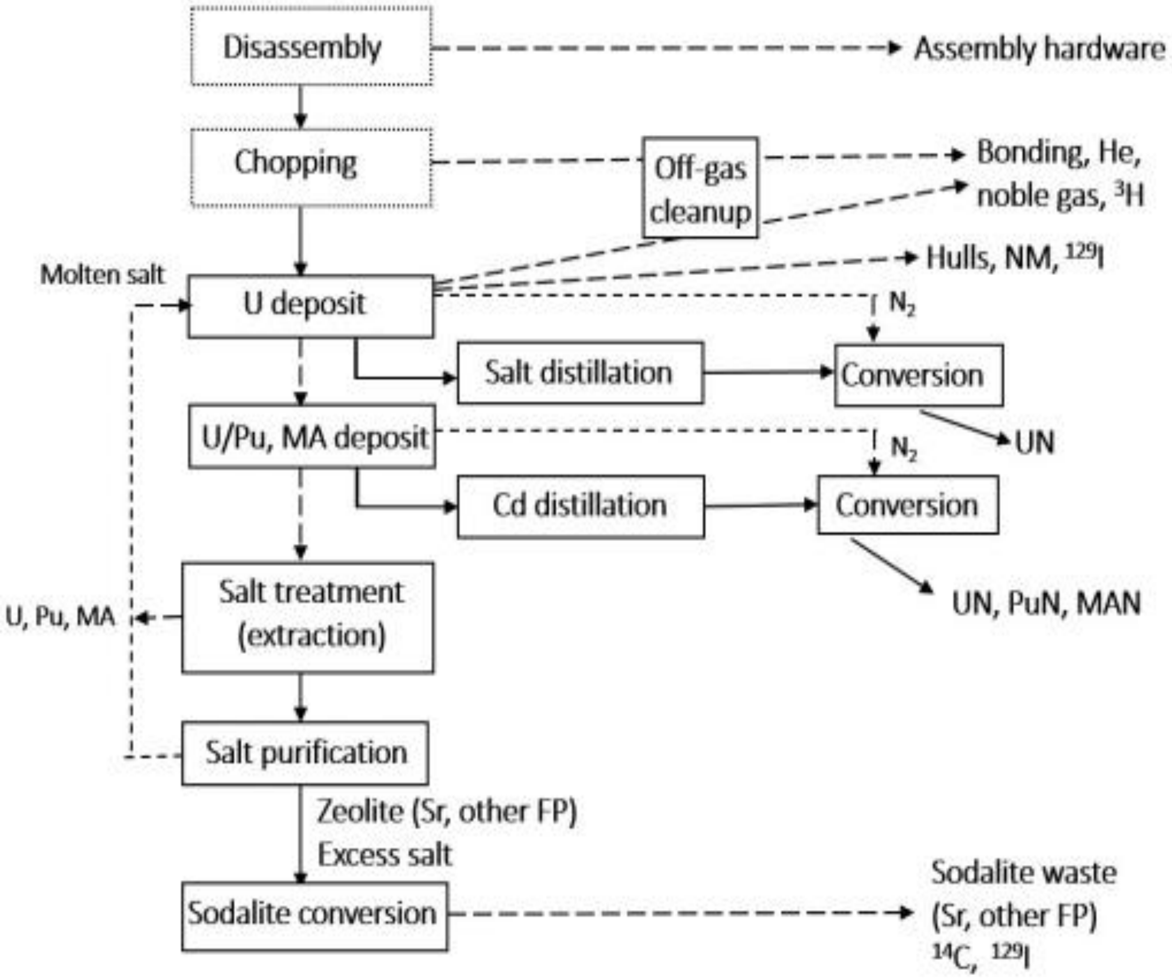




Process flows and primary waste streams for pyro-reprocessing of FR metal and nitride SNF

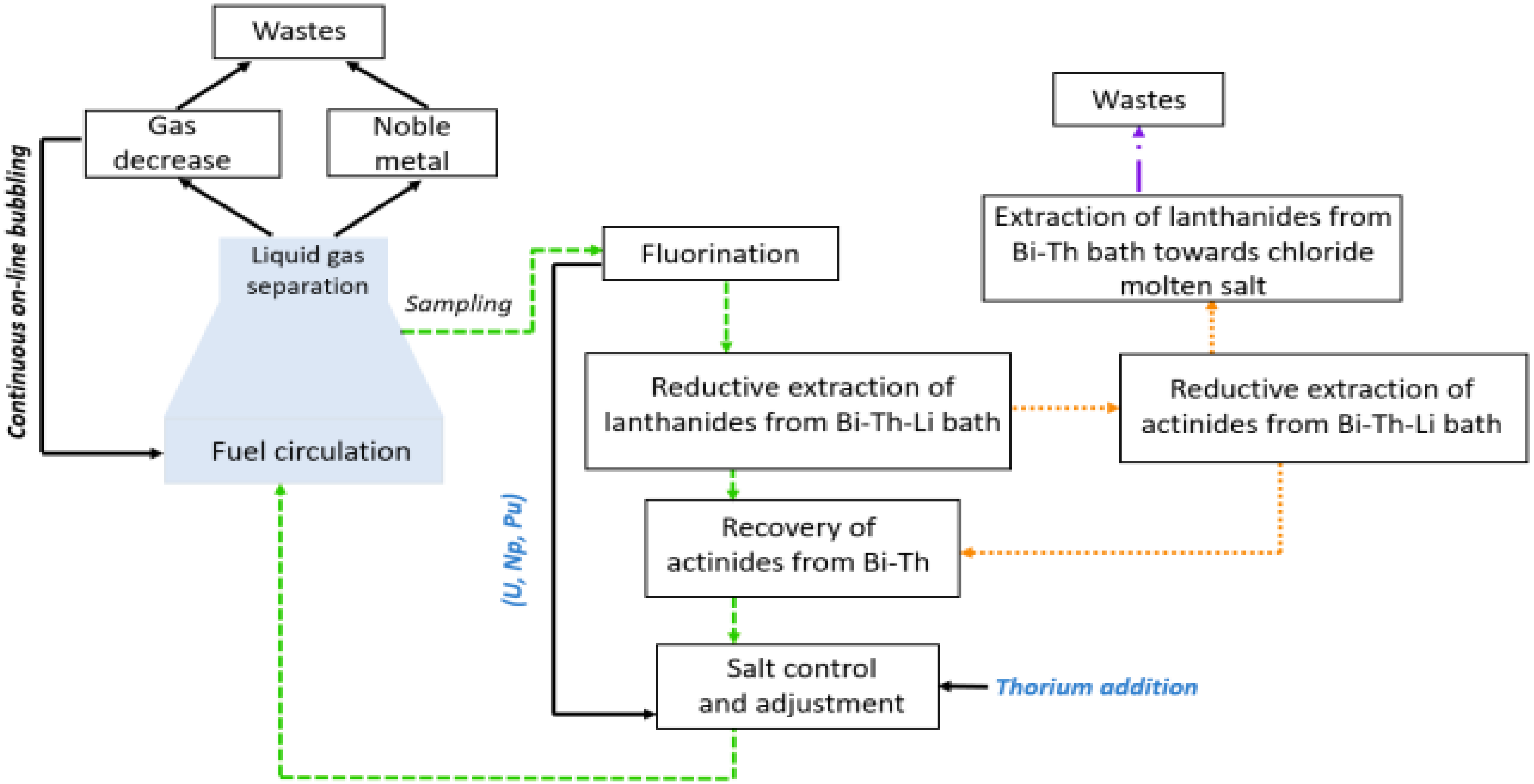
WIRAF





Process flow and primary waste stream for recycling of MSR fuel

WIRAF



Some Author’s Considerations (1)

MOX for FRs

- PUREX application - criticality issues, for avoiding them, FR MOX may be reprocessed as additions to PWR reprocessing flows (Pu mass balance issues)
- Pyro-process is applied for recycling by batch approach

Metal fuel for FRs

- Not convenient for aqueous-reprocessing without oxidation (and some other difficulties as Na-water reaction etc
- Pyro-process is more advanced method.

Nitride fuel for FRs

- Aqueous processes need oxidation; issues related C-14 migration and its capturing; capturing of N-15 (if enriched).

  Pyro-process is convenient as initial step: N-15 can be captured and C-14 should collected in molten salt.

Carbide fuel for FRs

- Similar options as for Nitrides – pyro-process as first stage.

Some Author’s Considerations (2)

TRISO or GFR fuel in graphite matrix

- non (easy) recyclable. As manufacturing technology is sensitive to radioactive aerosol, it can be applied mainly for pure LEU, HALEO or HEU. In case of reprocessing, reprocessed uranium could be used for recycling in other reactors.
- aqueous reprocessing is not applicable without preparation steps. Pyro-process is possible for removal graphite and SiC before electrowinning of UO_2 in molten salt.

MOX or UOX fuel of SCWR

- Could be reprocessed and recycled by conventual methods.
- Th-based fuel will require new pyro or aqueous processes

Molten salt reactor fuels.

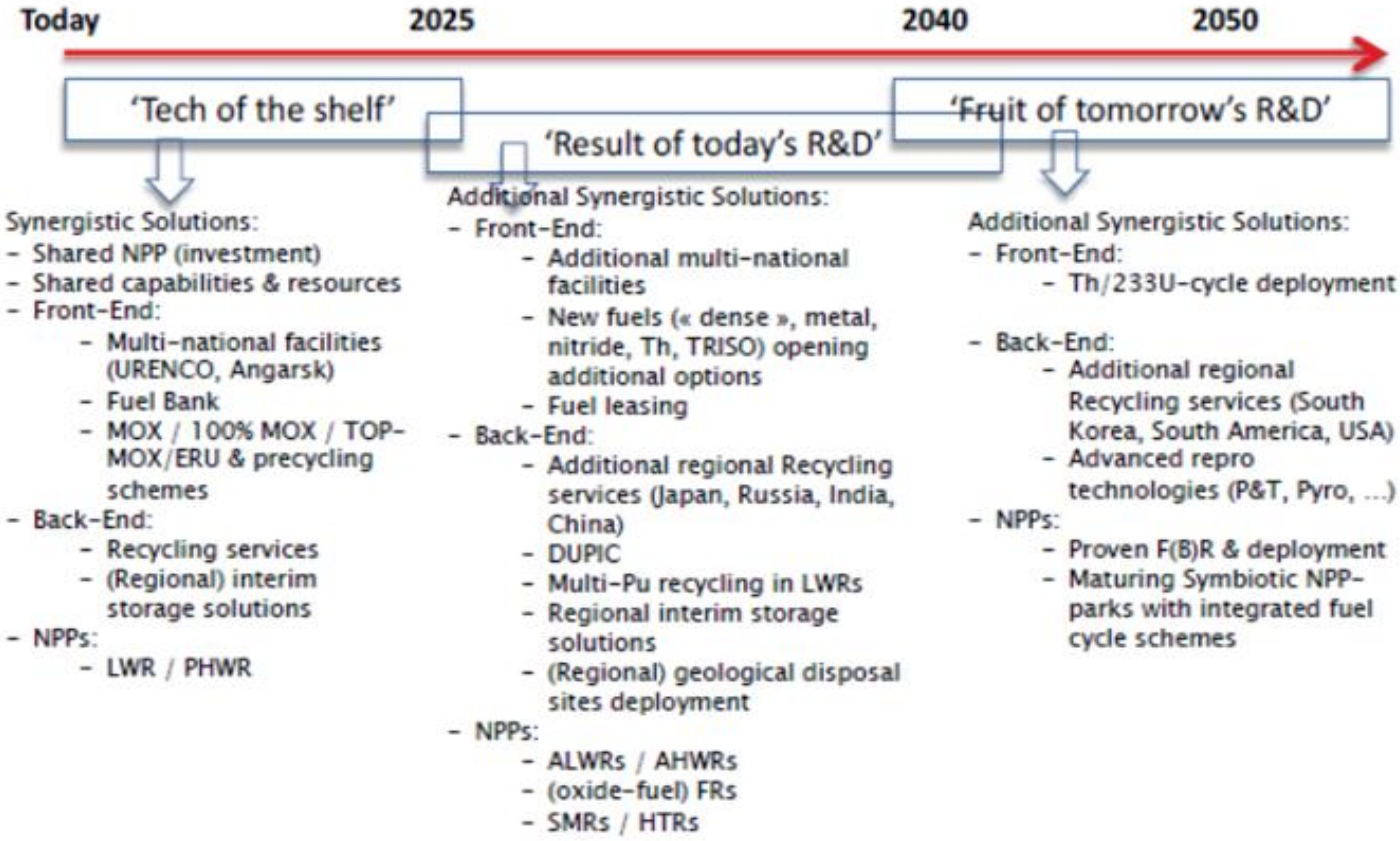
- Noble gases and Noble Metals fission product can be removed online by special traps.
- For complete reprocessing of salt: some pyro-process could be applied as molten salt electrolysis, precipitation or fluorination methods.
- No complete understanding and models for behavior of fusion products atoms in ionic liquids after actinide fission

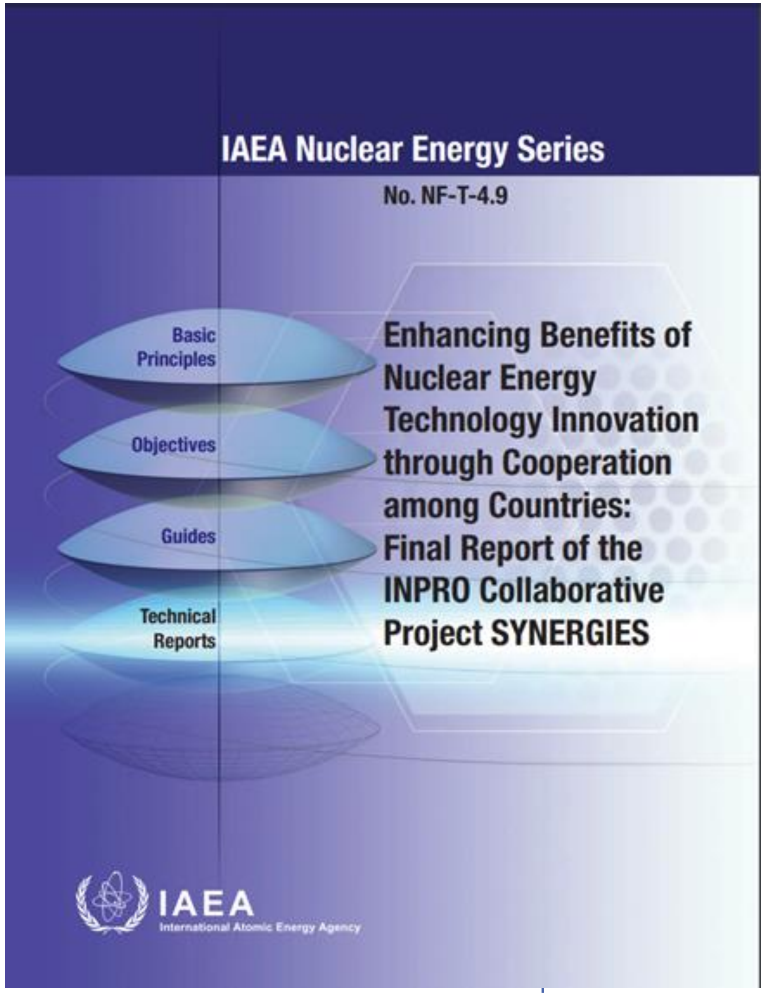
Instead of conclusion

Advanced reactors, advanced fuel and recycling in future international architecture of Nuclear Power

INPRO: "SYNERGIES" project

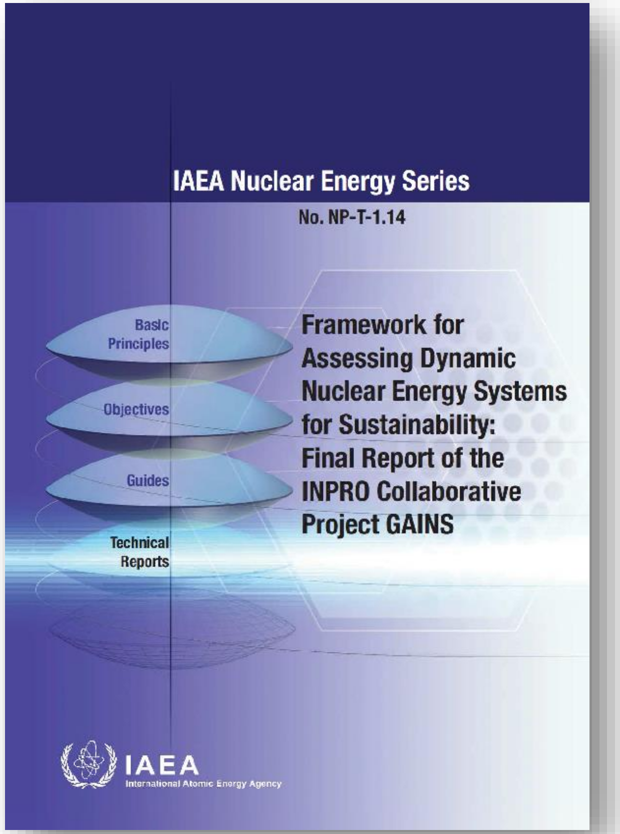
What Can We Do Before 2050?

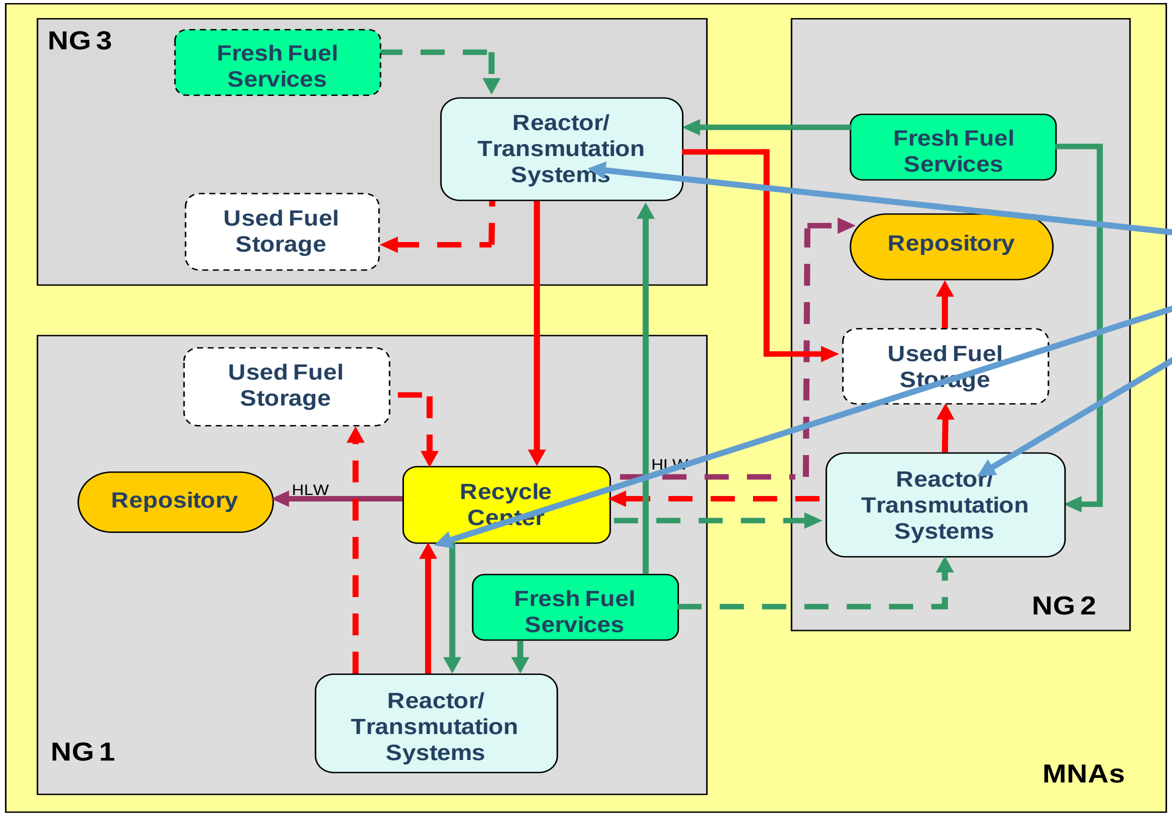




INPRO Collaboration Project GAINS 2008-2012

Global Architecture of Innovative Nuclear Energy Systems Based on Thermal and Fast Reactors including a Closed Fuel Cycle





Gen IV Reactors and fuel cycle

Thank you!

The presentation contains illustrations from publicly available publications of IAEA, OECD/NEA, Generation IV International Forum, and from E-learning course on Advanced nuclear fuel training.

Special thanks to Anzhelika Khaperskaia (IAEA/NFCMS)!