European Commission

International Energy Agency
Energy Technology Network
Advancing Materials Research for Power Generation
A strategic discussion led by the Fusion Power Co-ordinating Committee (FPCC)
28 January 2015, Paris, France

Nuclear science and materials for advanced fuel cycles and nuclear waste transmutation

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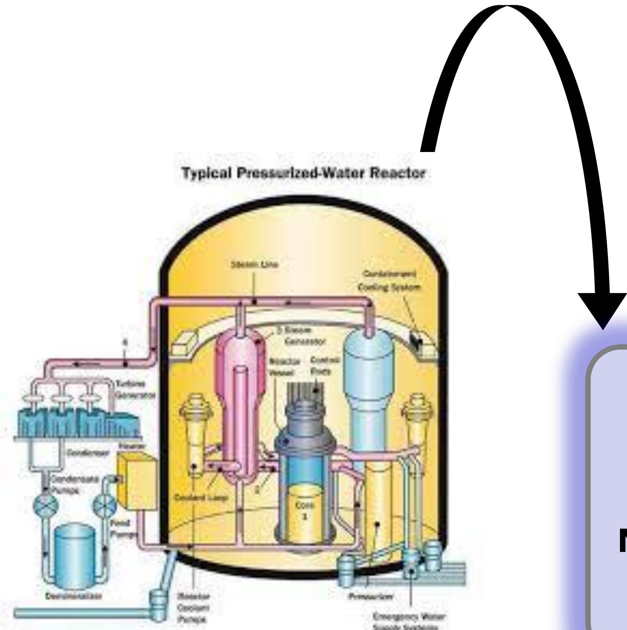
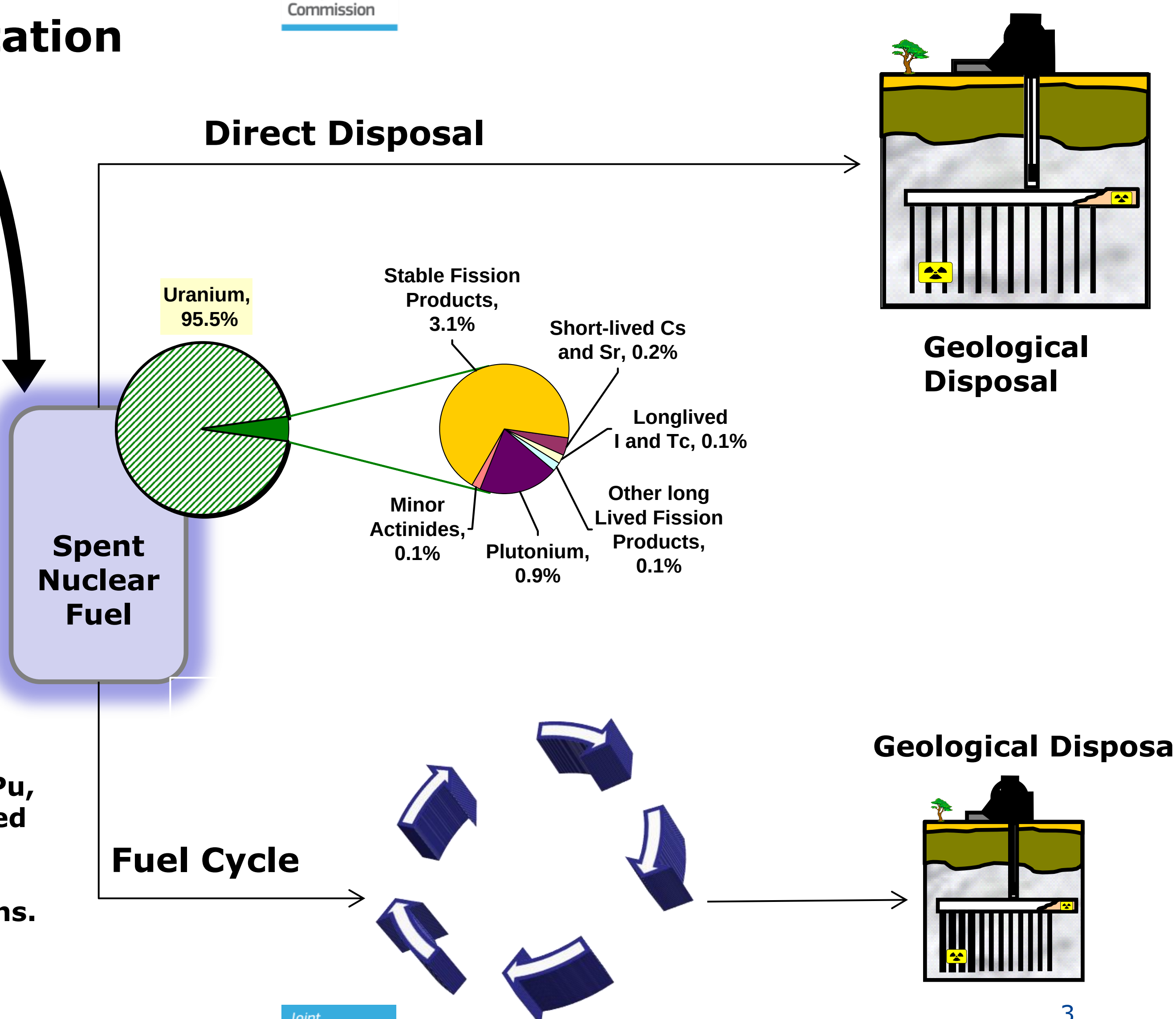
Outline

- Advanced fuel cycles and waste transmutation
- Materials and operational conditions of Transmutation systems
- Challenges of selected Transmutation systems
  - Fusion Fission Hybrid (FFH)
  - Accelerator Driven System (ADS)
- Two examples on materials issues:
  - The window material of the MEGAPIE neutron spallation target
  - Nuclear transmutation fuel pin development
- Outlook on future cross-cutting topics

Advanced fuel cycles

and waste transmutation

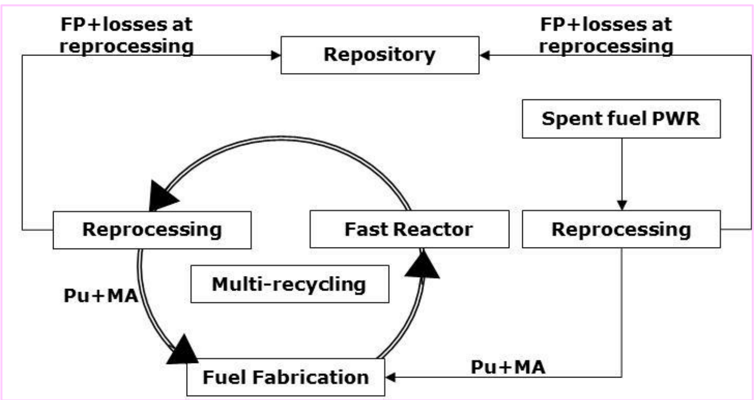




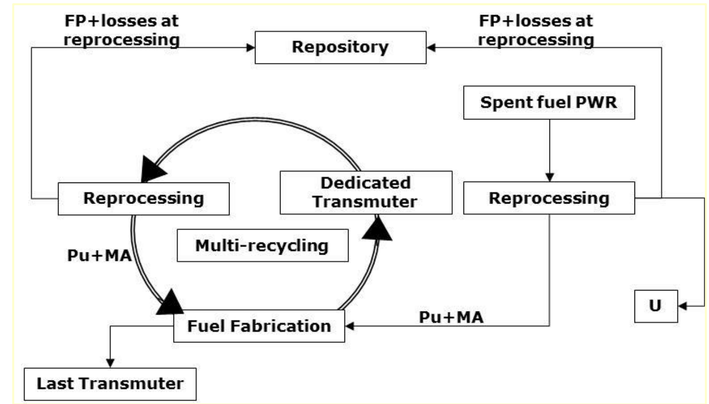
Most of the hazard stems from Pu, MA and some LLFP when released into the environment, and their disposal requires isolation in stable deep geological formations.

Advanced fuel cycles

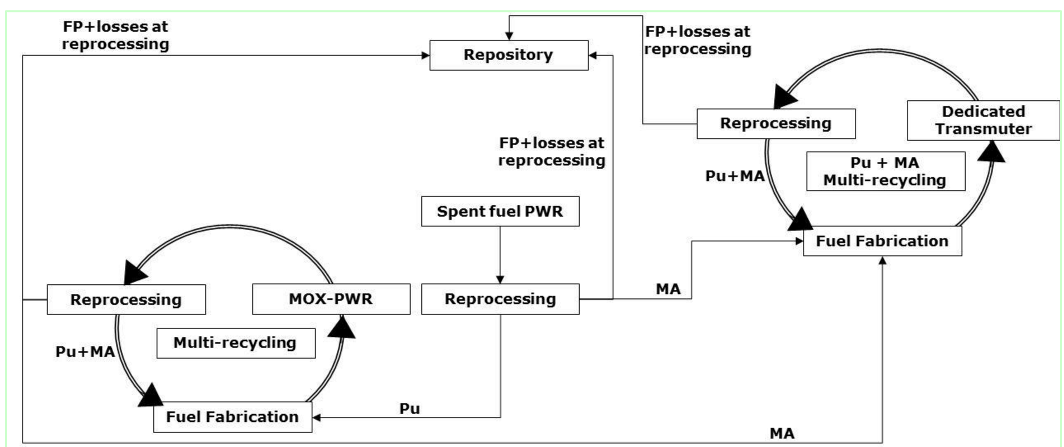
Sustainable development of nuclear energy and waste minimisation



Reduction (elimination) of TRU inventory as unloaded from LWRs



Reduction of MA inventory (pure waste management objective)

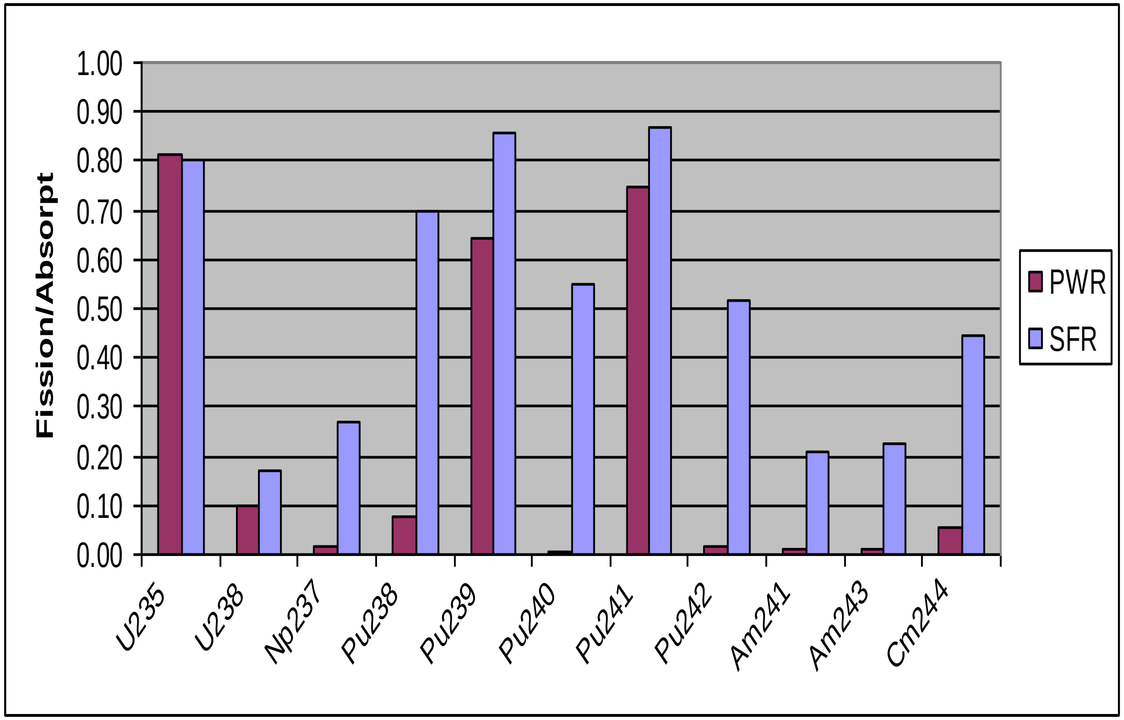


Which Transmuter?

Transmutation: Use of nuclear reactions to transform long lived nuclides into stable or short-lived nuclides

Fast neutron spectrum reactors have a favourable neutron economy with respect to thermal neutron spectrum reactors

Fission-to-Absorption Ratio for PWR and SFR



- **Fissile isotopes (e.g. U-235, Pu-239, Pu-241) are likely to fission in both thermal/fast spectrum. However, the fission fraction is higher in fast spectrum**
- **Significant fission (up to 50%) of fertile isotopes (e.g. U-238, Pu-240) in a fast spectrum**

Transmutation systems*

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System	SFR	LFR & ADS	GFR
Coolant	Na, few bars	Pb or Pb-Bi eutectic, few bars	He, 70 bars
Core components	Fuel: U, TRU Fuel Cladding: ODS (15Cr-15Ni Ti stabilised austenitic steel) Subassembly Wrapper: 9Cr MS	Fuel: IMF (MgO or Mo), TRU Cladding: F/M Steels, ODS, 15Cr-15Ni Ti stabilised austenitic steel Wrapper: 9Cr MS ADS target: 9Cr F/MS 350-550° C 100dpa+He+H	Fuel: U, TRU Cladding: SiC-SiCf composite or (backup) ODS Low-power dn: steel structure under consideration
Core Inlet/Outlet Temp.	390-750° C	Pb, 400-480° C; LBE, 300 – 450 ° C	500-1200° C
Max. Dose	up to 200dpa	100dpa	60-90dpa
Other components	primary/secondary/steam generator: 9-12Cr F/MS; austenitic steels, Ni-alloys 390-600° C Vessel: steel	Heat Exchanger: T91 or 316L Vessel: AISI316L	Heat Exchanger: Ni-based alloy Vessel: 9-12Cr MS 350-500° C <<1dpa

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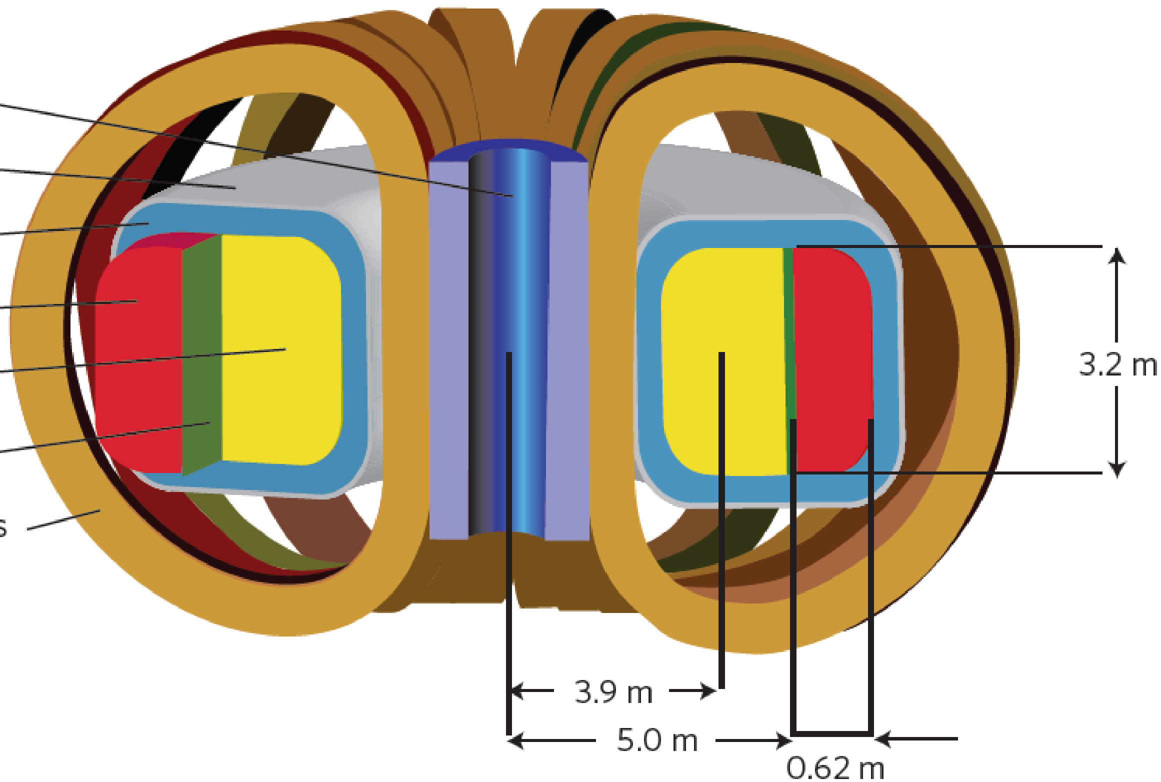
*Fusion-Fission Hybrid not included

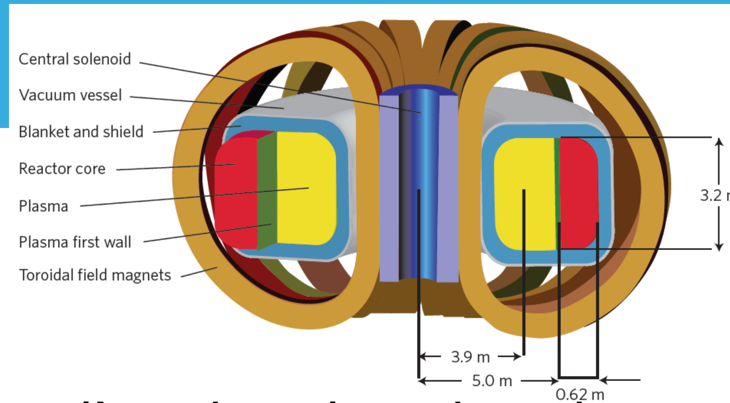
A FFH based on ITER and ANL FR



"The most common hybrid design consists of a "small" fusion reactor (100-500 MW_{th}) core surrounded by a subcritical blanket of fissile material (~ 3000 MW_{th}). The generation of neutrons by the fusion of hydrogen isotopes in the core drives fission reactions in the fast neutron blanket. These neutrons can be used to transmute waste"

Stacey, W. M. *et. al. Nucl. Technol.* 162, 53–79 (2008)
J. P. Freidberg, Nature Physics, Vol 5, 2009.

FFH based on ITER and ANL FR



Challenges:

- Design of the fission blanket (sub-critical) to be placed in the vicinity of a fusion device
- Materials selection
- Fissile fuel might be metal in a steel cladding or might be a molten salt fissile fuel
- Coolant Thermal-hydraulics and technology
- Safety
- Fuel handling and maintenance
- …..

	100 dpa	200 dpa	300 dpa
Fuel residence time (yr)	~ 4	~ 8	~ 11
TRU to the repository per year (kg)	68	32	20

Stacey, W. M. *et. al. OECD-NEA, 2012*

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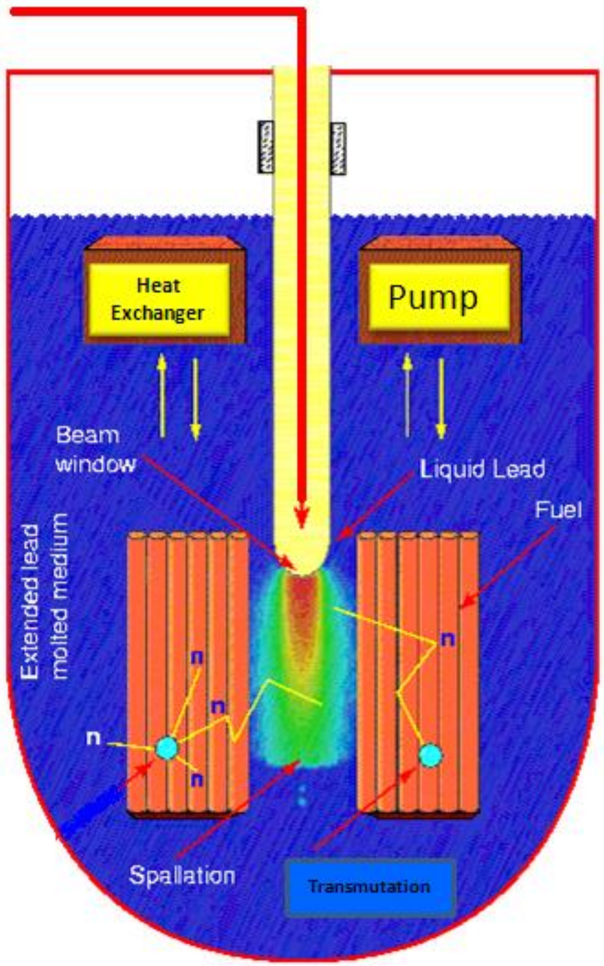
ADS: Challenges

Accelerator Development

Transmutation fuel development

Coupling Experiments

Safety oriented fuel and core design



Accelerator Driven System (ADS)

Thermal-hydraulics and Instrument.

Components: pump, heat exch.

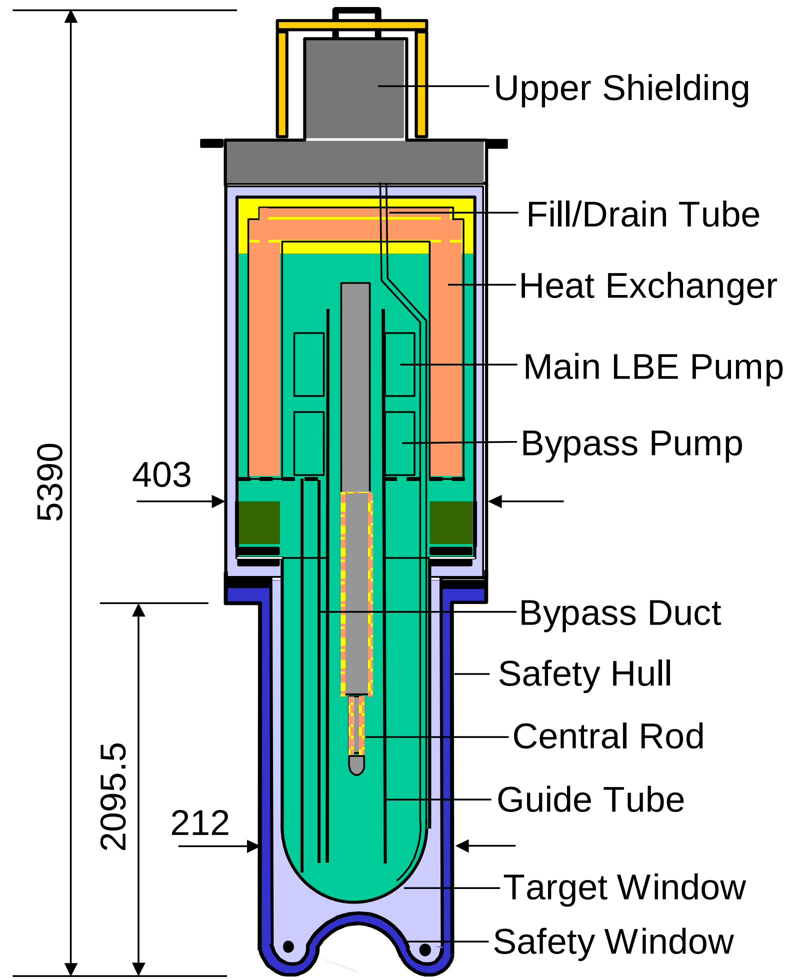
Neutron Spallation Target

Materials and coolant chemistry

European Facility for Industrial Transmutation (EFIT) Ref.: EUROTRANS

The MEGAPIE Target

Main Specifications	
Length	5.35 m
Diameter lower part	10.6 cm
Diameter upper part	20 cm
LBE volume	About 87 l
Structural materials	
Lower liquid metal container	**T91 steel**
Upper container	316L steel
Lower target enclosure	AlMg3
Operation parameters	
LBE temperature range	230–380 C
Max LBE flow velocity	1.2 m/s
Window temperature range	330–380 C
Beam characteristics	
Proton beam energy	**575 MeV**
Maximum proton current	**~ 1.4 mA**



A critical component has been the beam entrance (Target) window

L. Zanini, et al. J. Nucl. Mater. 415 (2011) 367–377

C. Fazio et al. Nucl. Eng. Des. 238 (2008) 1471–1495

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MEGAPIE Target: Power deposition

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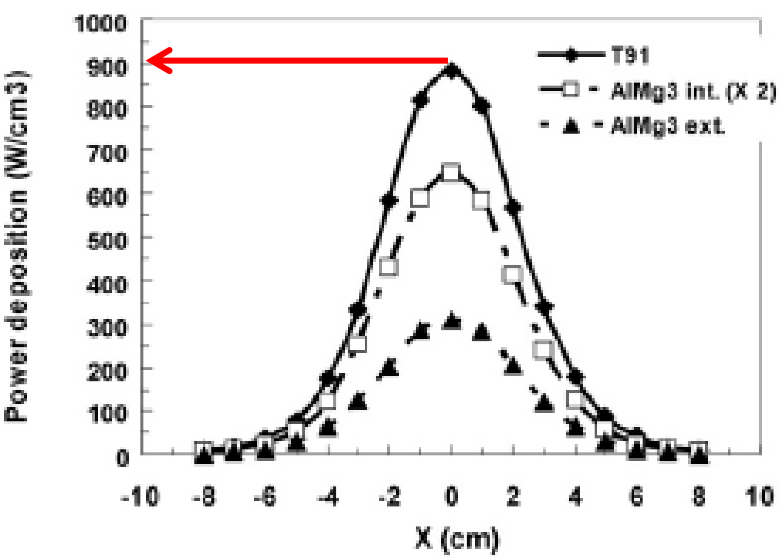
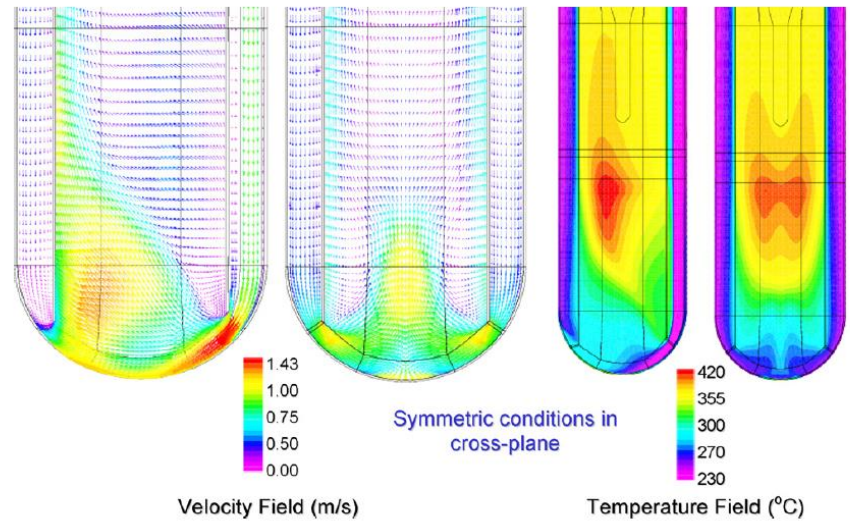
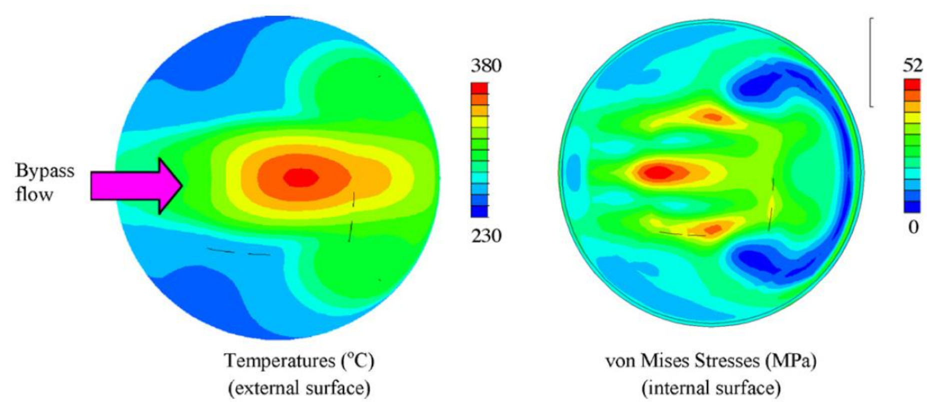
Material	Power kW
Total	852
Pb-Bi	710 (71%)
T91	**~ 9 (1%)**
Other parts	133 (28 %)



Unacceptably high power density in the T91 window (2 mm thickness): forced cooling needed





C. Fazio et al. Nucl. Eng. Des. 238 (2008) 1471–1495

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Irradiation: dpa and He

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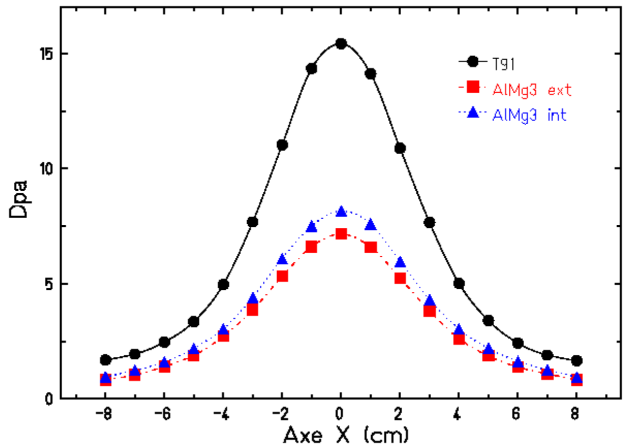
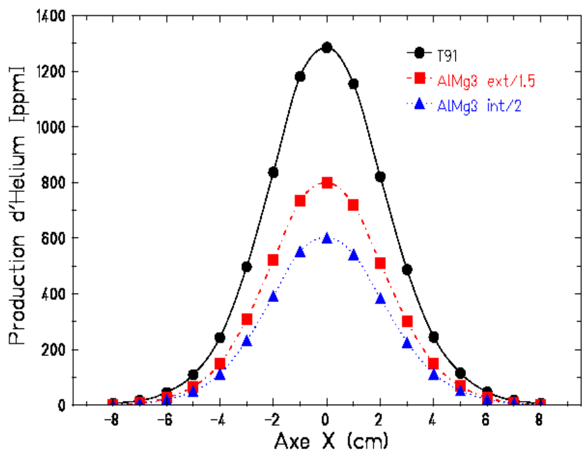
dpa after 6 Ah



He after 6 Ah



6 A h = 1.25 mA during 200 days (~ 7 months)

Y. Foucher in FZKA 6876, 2003

Summary Beam Window operational conditions

- **Material** — **T91 Steel**
- **Thickness** — **~ 2 mm**
- **Temperature (max)** — **380 ° C**
- **LBE Flow Velocity (max)** — **1.2 m/s**
- **Van Mises Stress (max)** — **50 MPa**
- **Max dpa** — **~ 15**
- **Max He ppm** — **~ 1300**

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Window lifetime assessment

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Experimental validation done to assess the performance of the window materials in the frame of the MEGAPIE initiative and the EC supported project MEGAPIE-TEST:

- **Thermal-hydraulics experiments → window coolability (e.g. KALLA, KIT and PSI)**
- **Corrosion → wall thickness reduction: loss of bearing load capability (e.g. LECOR, ENEA and KALLA, KIT)**
- **Environmental assisted mechanical degradation: e.g. liquid metal embrittlement (ENEA, PSI, SCK-CEN, CNRS)**
- **Irradiation → p, n irradiation damage, transmutation products (He and H): impact on mechanical properties (CEA and PSI)**
- **Combined effect LBE/irradiation field: synergetic effects on mechanical properties degradation (PSI, SCK-CEN)**

Several literature data available

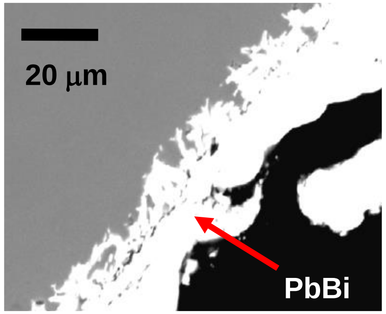
Corrosion

- **Oxygen sources in the MEGAPIE Target:**
  - **Saturated in LBE at $T_{filling}$ (250° C) i.e. $2.6 \cdot 10^{-6}$ wt. %**
  - **Dissolved in cover gas**
  - **Adsorbed on the steel structures**

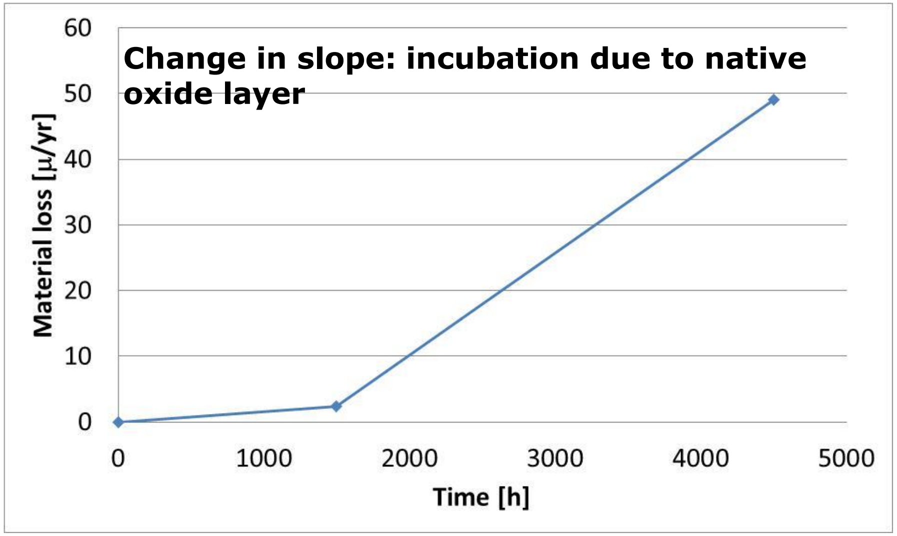
- **Continuous Oxygen depletion during operation:**
  - **Reaction with Spallation products (e.g. Hydrogen)**
  - **Reaction with hottest structures**
  - **Stripping with cover gas**

- **Corrosion mechanism might change over time from oxidation to dissolution**

- **Focus on dissolution: worse condition**



T91 in flowing LBE at 400 ° C



In 1 year ~ 3 % reduction of beam window wall thickness if temperature would be 400° C

A. Aiello et al., J. of Nucl. Mater. 335 (2004) 169–173

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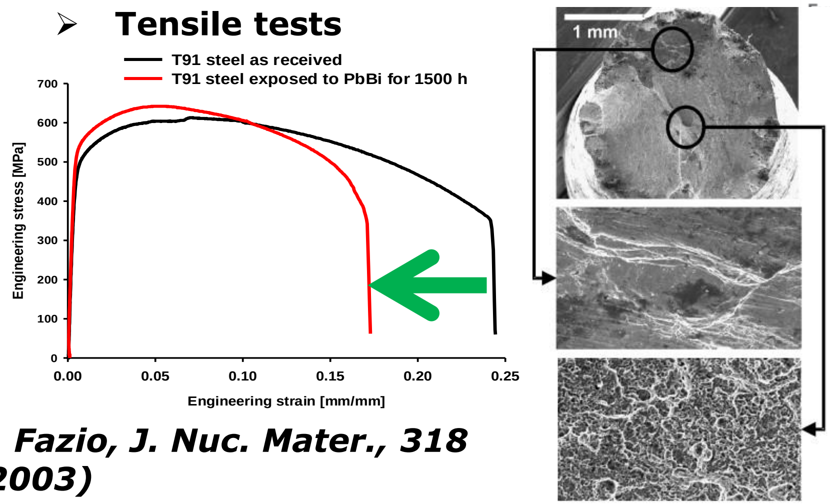
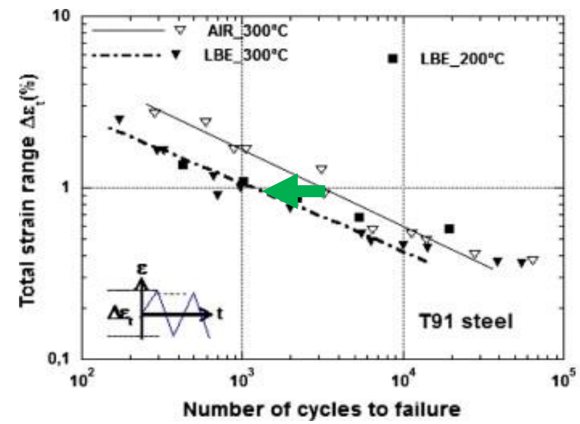
Mechanical properties

➤ **Tensile tests**



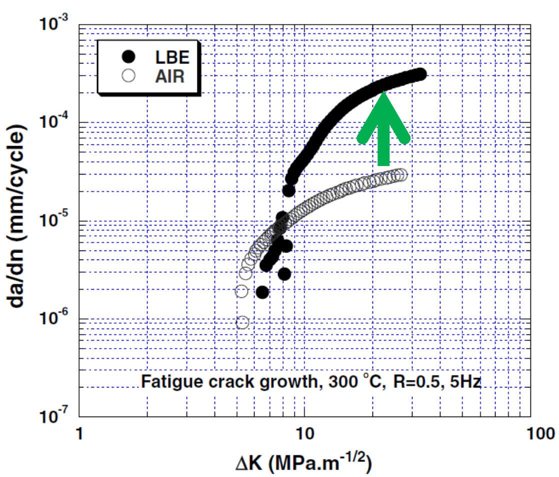
C. Fazio, J. Nuc. Mater., 318 (2003)

➤ **LCF tests**



D. Gorse, J. Nuc. Mater, 415 (2011)

➤ **Bending tests**



Y. Dai et al., J. Nucl. Mater. 356 (2006) 308

Summary of LBE impact on mechanical properties

- **Tensile: reduction of total elongation**
- **LCF: Reduction of number of cycle to failure for high strain (stress) ranges**
- **Bending: faster LCF crack grow rate in LBE with respect to air**

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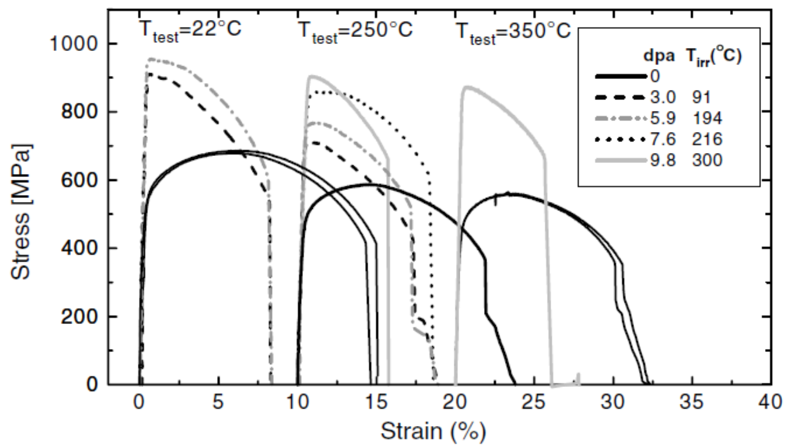
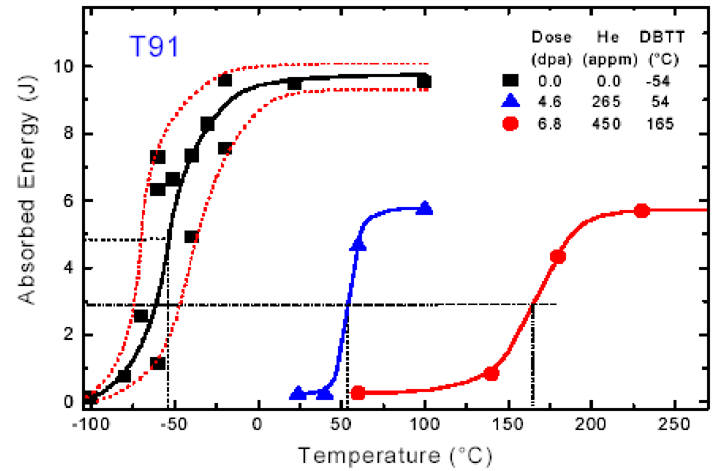
Mechanical properties

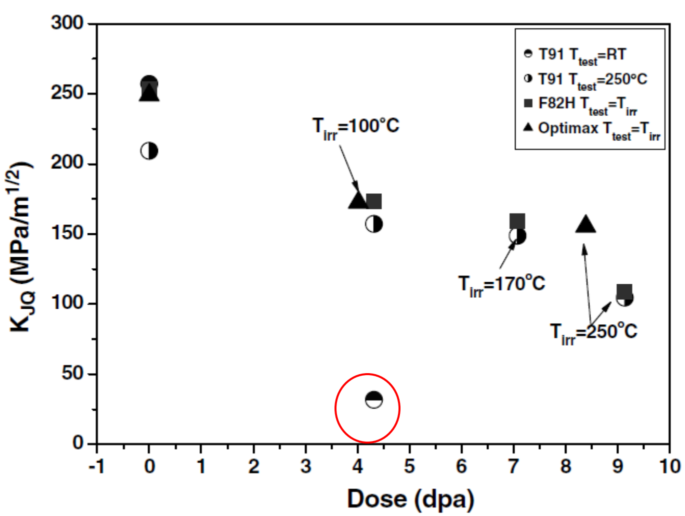
➢ **Tensile test**



➢ **Impact test**



➢ **Bending test**



Summary Irradiation impact on mechanical properties

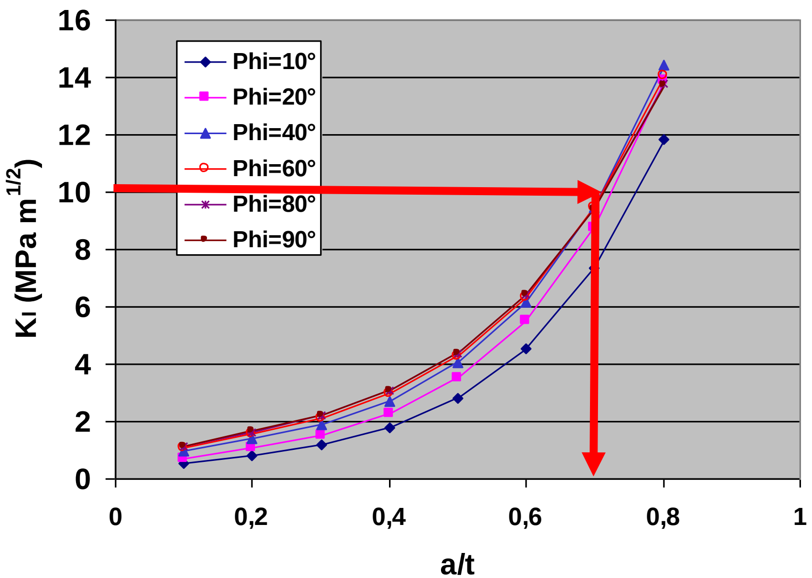
- **Tensile: Hardening and reduction of uniform and total elongation**
- **Impact: Ductile to brittle temperature shift**
- **Bending: Reduction of fracture toughness (value is above 40 MPa·m$^{1/2}$ at ~ 9dpa)**

Y. Dai et al., J. Nucl. Mater. 356 (2006) 308

Window performance assessment

1. Method: Evaluation of stress intensity factor (K_I) to assess failure

- $K_I = f(1/\sigma_{tip}, \sigma, a)$
  - σ = mechanical load in normal operation condition (von Mises Stress)
  - σ_{tip} = mechanical load at crack tip
  - a = crack depth (through wall thickness)
- $K_I \geq K_{IC}$ (Fracture toughness) crack propagation / failure occurs



e.g. for a stress intensity factor of ~ 10 MPa·m$^{1/2}$ and a crack depth ~ 1 mm (a/t ~ 0.7) crack propagation/failure occurs

J. Henry at al., Springer Science, 2008

- If K_I decreases the crack depth at which failure occurs decreases.
- LBE: Reduction in fatigue life and increase in fatigue crack growth disappear in low stress ranges
- Irradiation: Up to 9 dpa (in spallation environment) K_{IC} > 40 MPa·m$^{-1/2}$

Conclusion: apparently no risk of failure

Y. Dai et al., J. Nucl. Mater. 356 (2006) 308

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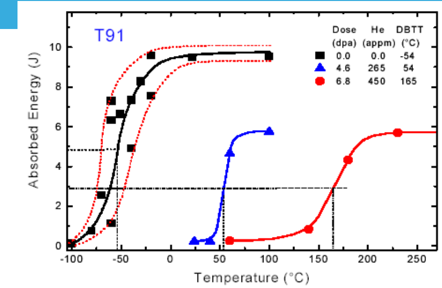
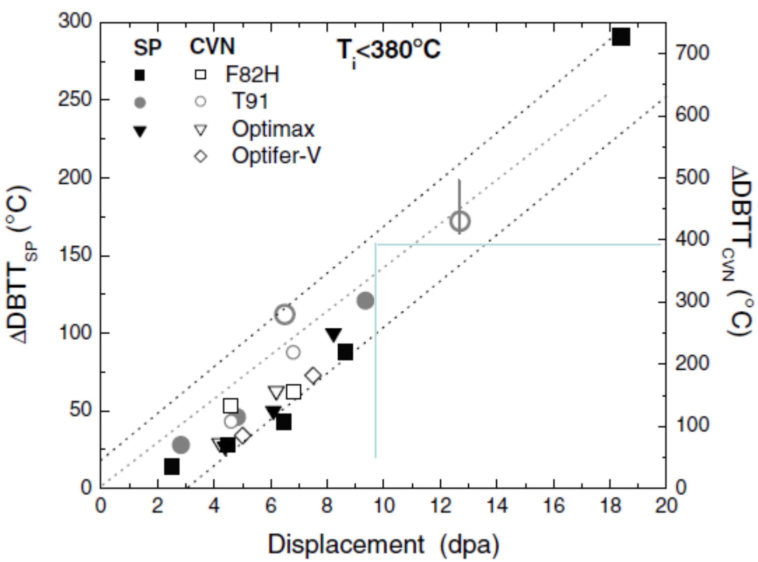
Window performance assessment

2. Method: Operation of window in the ductile regime

- T91 exhibits a ductile to brittle transition temperature (DBTT)
- The DBTT increases with increasing n/p irradiation
- The upper shelf energy decreases





- **Criteria: DBTT should not reach lowest operational temperature at beam off condition = 230 ° C**
- **DBTT of not irradiated T91 ~ -50 C then ΔDBTT is < 280° C for dpa < 10**
- **Safety margin of 30% → dpa ~ 6 which corresponds to ~ 3 Ah (~ 90 days at 1.4 mA continuous irradiation)**

Y. Dai et al., J. Nucl. Mater. 356 (2006) 308

Conclusion: a limit on lifetime is given by DBTT considerations

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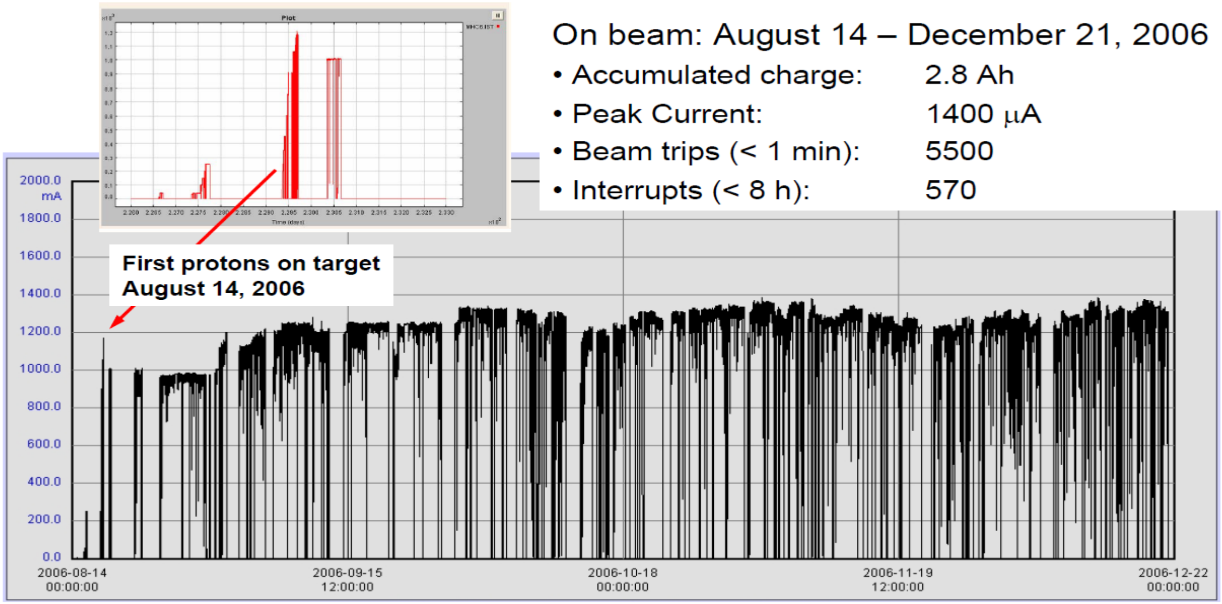
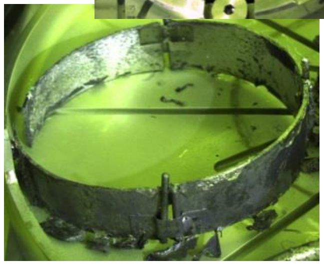
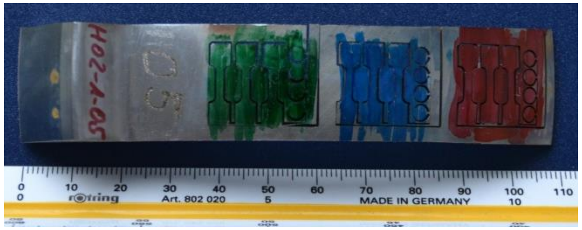
What happened to the MEGAPIE Target?



Design, fabrication, assembling → **Irradiation in SINQ** → **Target cutting** → **Sample extraction** → **Post Irradiation Examination**





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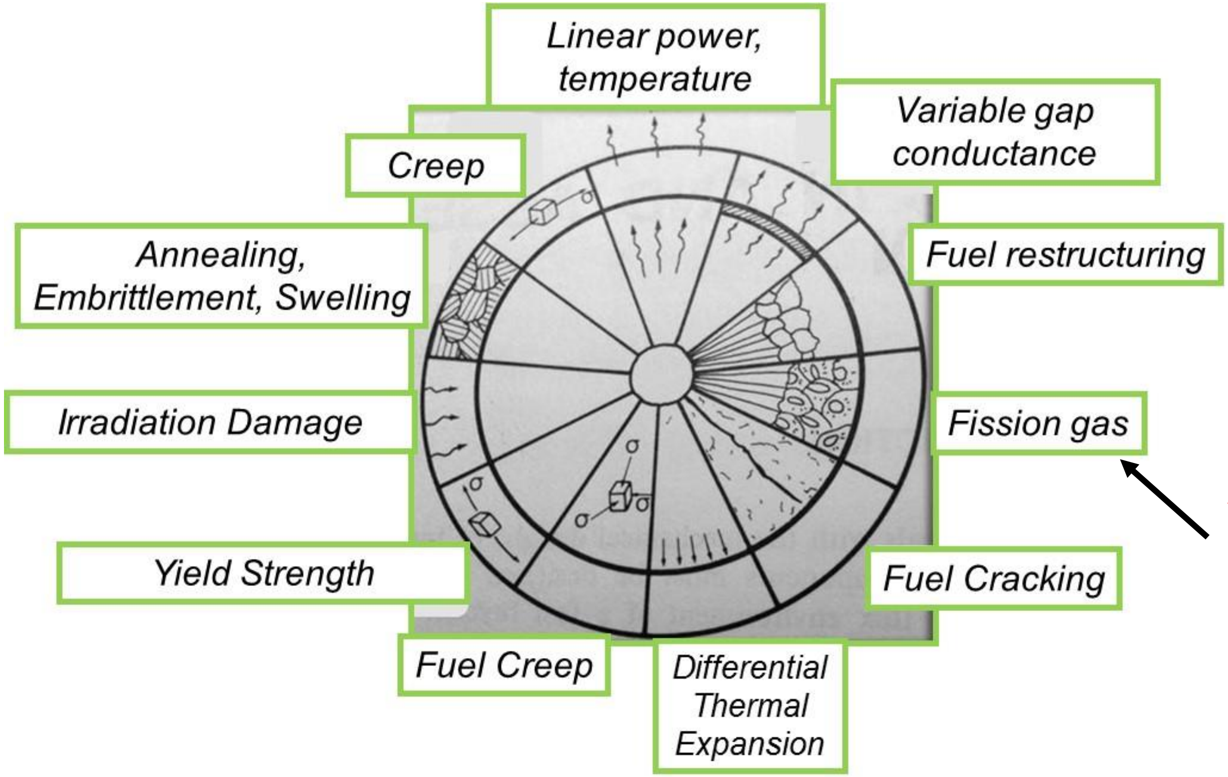
Transmutation fuel pin requirements

- The fuel must accommodate Pu and MA within a wide range of composition
- The fuel must be able to achieve high burn-up (up to 250 GW d/ ton HM)
- Fabrication and quality control process must be amenable to fully remotized operations
- TRU losses during fabrication must be minimized
- Fuel form must be compatible with coolant
- Fuel form must be compatible with clad material
- Fuel form must be compatible with reprocessing scheme

Phenomena affecting fuel pin performance



He production (from Cm242 alpha decay) in a Am 241 bearing fuel, is very high with respect to standard MOX FR fuel and impacts the pin performance

Source A. E. Walter, Fast Breeder Reactors, Pergamon Press

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Table 5.1: Inert matrix oxide fuel irradiation programmes and their status

Fuel Form	Composition	Reactor	Programme	Status
CER	$(Zr,Y,Am)O_2$	Phénix	CAMIX	Irradiated
	$(Zr,Y,Am)O_2$	HFR Petten	HELIOS 2	Irradiated
	$(Zr,Y,Pu,Am)O_2$	HFR Petten	HELIOS 3	Irradiated
CERCER	$MgAl_2O_4 – AmAlO_3$	HFR Petten	EFTTRA T4	PIE complete
	$MgO – AmO_2$	Phénix	ECRIX- B	Irradiated
	$MgO – AmO_2$	Phénix	ECRIX- H	Irradiated
	MgO - $(Zr,Y,Am)O_2$	Phénix	COCHIX	Irradiated
	MgO – $(Pu,Am)O_2$	Phénix	FUTURIX 7	Irradiated
	MgO – $(Pu,Am)O_2$	Phénix	FUTURIX 8	Irradiated
	$MgO – Zr_2Am_2O_7$	HFR Petten	HELIOS 1	Irradiated
CERMET	Mo – $(Pu,Am)O_2$	Phénix	FUTURIX 5	Irradiated
	Mo – $(Zr,Y,Pu,Am)O_2$	Phénix	FUTURIX 6	Irradiated
	Mo – $(Pu,Am)O_2$	HFR Petten	HELIOS	Irradiated
	Mo – $(Zr,Y,Pu,Am)O_2$	HFR Petten	HELIOS	Irradiated

State of the art report on innovative fuels for advanced nuclear fuel cycle systems, OECD-NEA, 2014

Cladding materials options

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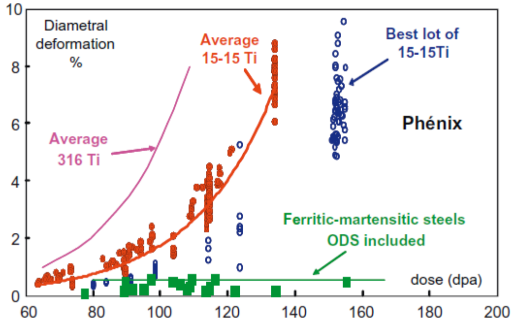
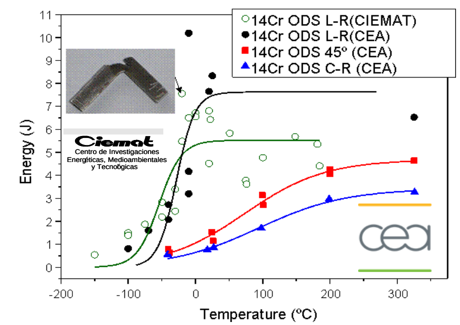
Austenitic steels



Dimensional stability: relatively high swelling

P. Dubuisson et al. / Journal of Nuclear Materials 428 (2012) 6–12

ODS F and F/M steels



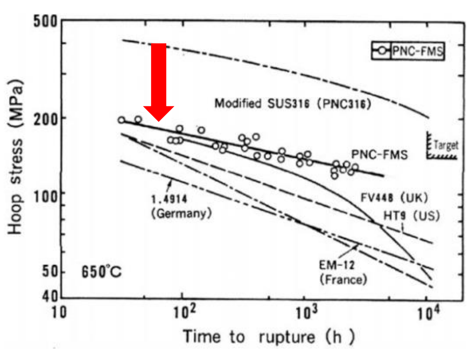
Impact Properties: non isotropic and non Conventional creep behaviour

GETMAT Project, courtesy M. Serrano, CIEMAT 2012

Challenges on clad material:

- **All material show potential for improvement:**
  - **composition**
  - **at fabrication**
  - **performance assessment**
  - **…**

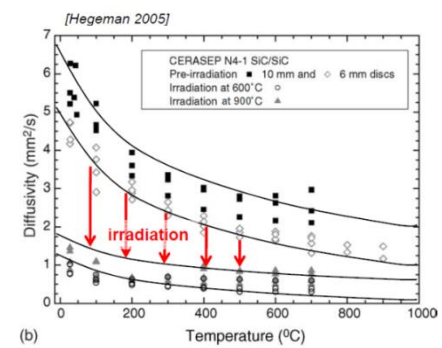
F/M steels



Mechanical: relatively low hoop stress

J.S. Cheon et al. / Journal of Nuclear Materials 392 (2009) 324–330

SiC_fSiC



Thermal conductivity: drops with irradiation

Source: M. Le Flem, CEA 2012

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Cross cutting topics

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- A multi-disciplinary methodology for the practical applications
- Modelling (first principles)
- Tools for experimental validation (irradiation facilities, advanced post irradiation investigation techniques)
- Specific materials development for multiple applications, e.g. fabrication process of ODS or advanced austenitic and F/M steels
- Codification / standardisation of materials e.g. RCC-MRx
- Cross fertilization in the education and training programs on nuclear materials science and associated disciplines as e.g. reactor physics, thermal-hydraulics, chemistry, safety, etc. should play a growing role. New initiatives to be worked out

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