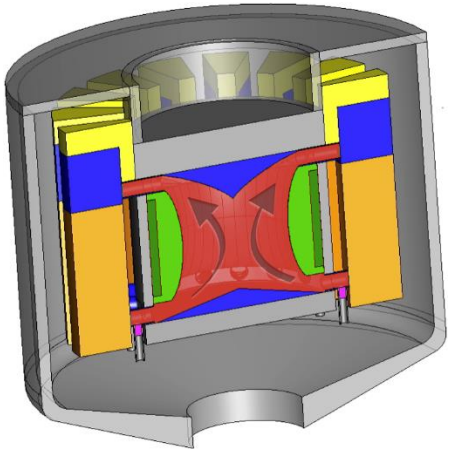
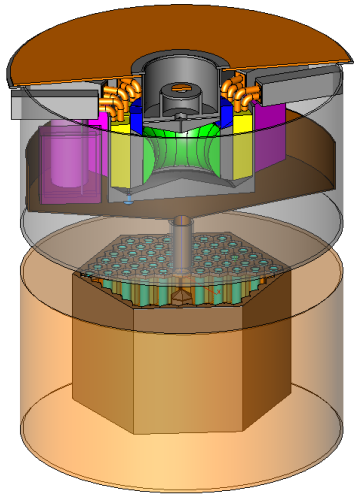
Concept of Molten Salt Fast Reactor





Prof. Elsa MERLE – merle@lpsc.in2p3.fr

On behalf of the MSFR CNRS French Team – Michel Allibert, Sylvie Delpech, Delphine Gérardin, Lydie, Giot, Daniel Heuer, Axel Laureau, Johann Martinet, Elsa Merle

LPSC Grenoble (CNRS-IN2P3 / Grenoble INP – PHELMA / Grenoble Alpes University), Subatech Nantes and IPN Orsay (CNRS-IN2P3)

With the support of the IN2P3/CNRS institute and the PACEN and NEEDS French Interdisciplinary Programs, Grenoble Institute of Technology, and of the EVOL and SAMOFAR Euratom Projects

Liquid-fueled reactors: why "molten salt reactors"?

Advantages of a Liquid Fuel

- ✓ Homogeneity of the fuel (no loading plan)
- ✓ Heat is produced directly in the heat transfer fluid
- ✓ Possibility to reconfigure passively the geometry of the fuel:
  - One configuration optimizes the electricity production managing the criticality
  - An other configuration allows a long term storage with a passive cooling system
- ✓ Possibility to reprocess the fuel without stopping the reactor:
  - Better management of the fission products that damage the neutronic and physicochemical characteristics
  - No reactivity reserve (fertile/fissile matter adjusted during reactor operation)

Which constraints for a liquid fuel?

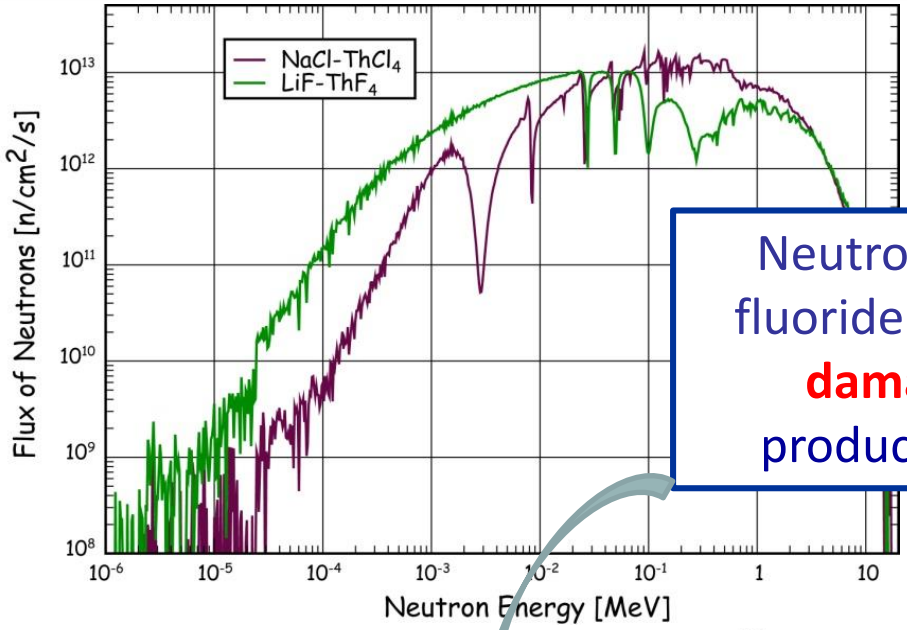
- Melting temperature not too high
- High boiling temperature
- Low vapor pressure
- Transparent to neutrons
- Good thermal and hydraulic properties (fuel = coolant)
- Stability under irradiation
- Good solubility of fissile and fertile matters
- No production of radio-isotopes hardly manageable
- Solutions to reprocess/control the fuel salt

Fluoride and chloride salts fulfill all constraints

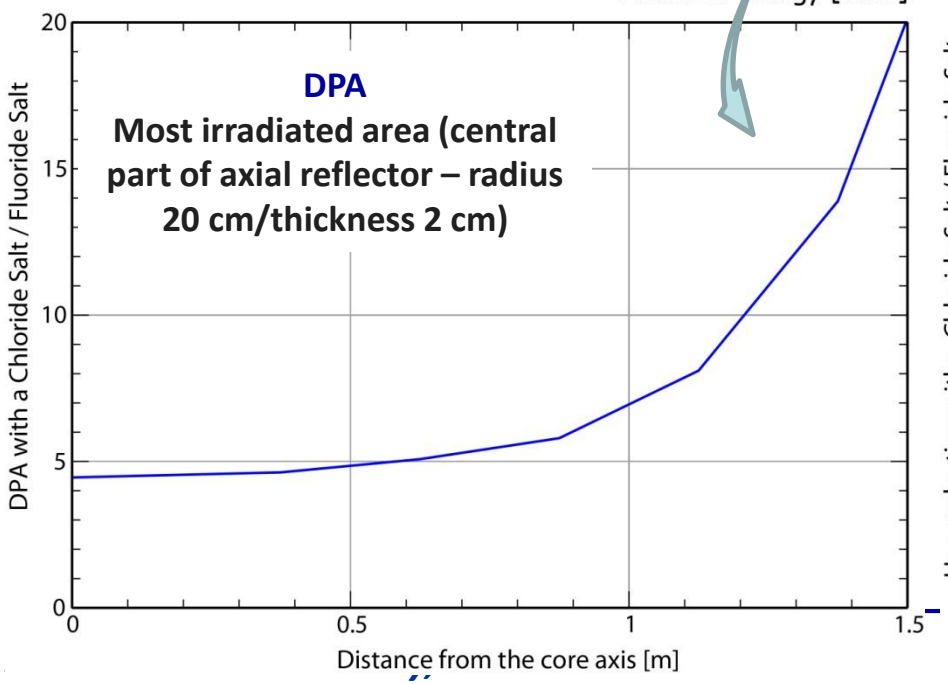
⇩

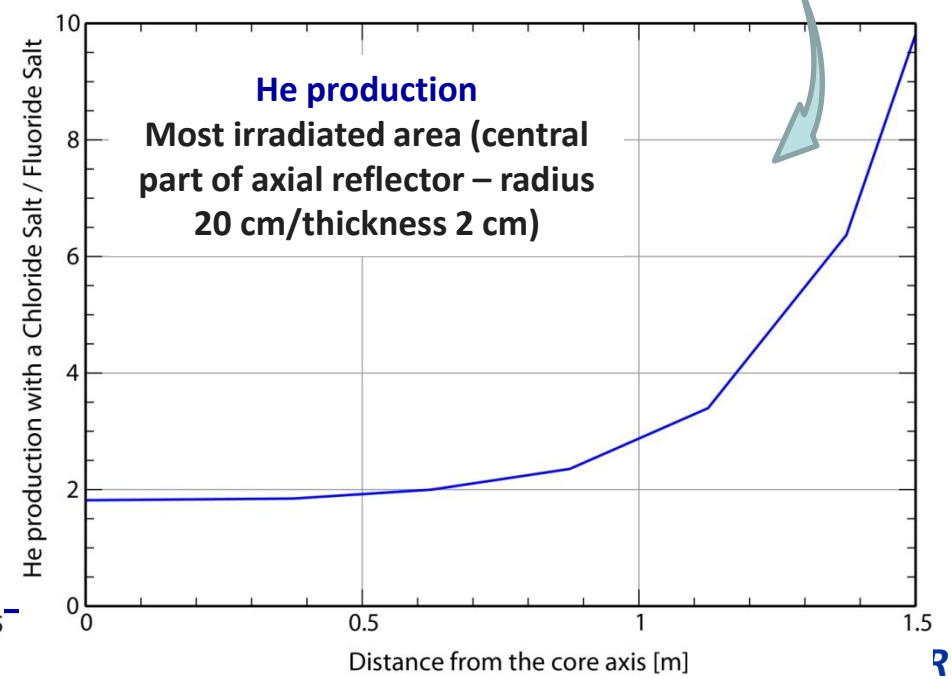
Molten Salt Reactors

Liquid-fueled reactors: fluoride or chloride? *Irradiation damages*



Neutron spectrum less fast with fluoride salt = **reduced irradiation damages** (both DPA and He production) **with a fluoride salt**





Liquid-fueled reactors: fluoride or chloride?

Elements produced

Element produced	Problem	Fluoride Salt	Chloride Salt
^{36}Cl produced via $^{35}Cl(n,\gamma)^{36}Cl$ and $^{37}Cl(n,2n)^{36}Cl$	Radioactivity in waste - $T_{1/2}$ = 301000y		10 moles / y (373 g/year)
^{3}H produced via $^{6}Li(n,\alpha)$ t and $^{6}Li(n,t)$ α	Radioactivity - $T_{1/2}$ = 12 years	55 moles / y (166 g/y)	
Sulphur produced via $^{37}Cl(n,\alpha)^{34}P(\beta$-[12.34s])$^{34}S$ and $^{35}Cl(n,\alpha)^{32}P(\beta$-[14.262 days])$^{32}S$	Corrosion (located in the grain boundaries)		10 moles / year
Oxygen produced via $^{19}F(n,\alpha)^{16}O$	Corrosion (surface of metals)	88.6 moles/year	
Tellurium produced via fissions and extracted by the on-line bubbling	Corrosion (cf. Sulphur)	200 moles/year	200 moles/year

Combination of all these considerations

⇩

'reference MSFR' based on a molten LiF fuel salt

Liquid-fueled reactors: why "molten salt reactors"?

Which constraints for a liquid fuel?

- Melting temperature not too high
- High boiling temperature, Low vapor pressure
- Transparent to neutrons
- Good thermal and hydraulic properties (fuel = coolant)
- Stability under irradiation
- Good solub
- No product
- Solutions t

Lithium fluorides fulfill all constraints

Molten Salt Reactors

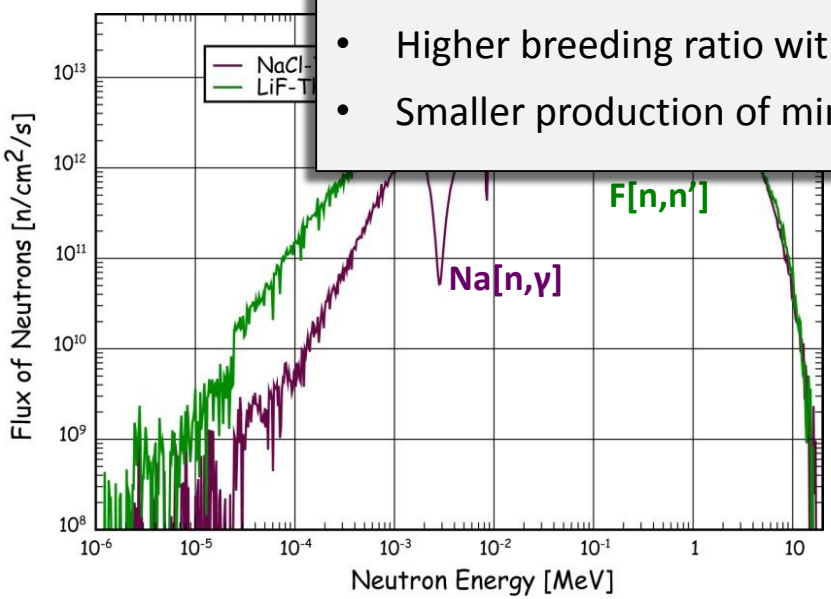
Choice of the fluoride salt:

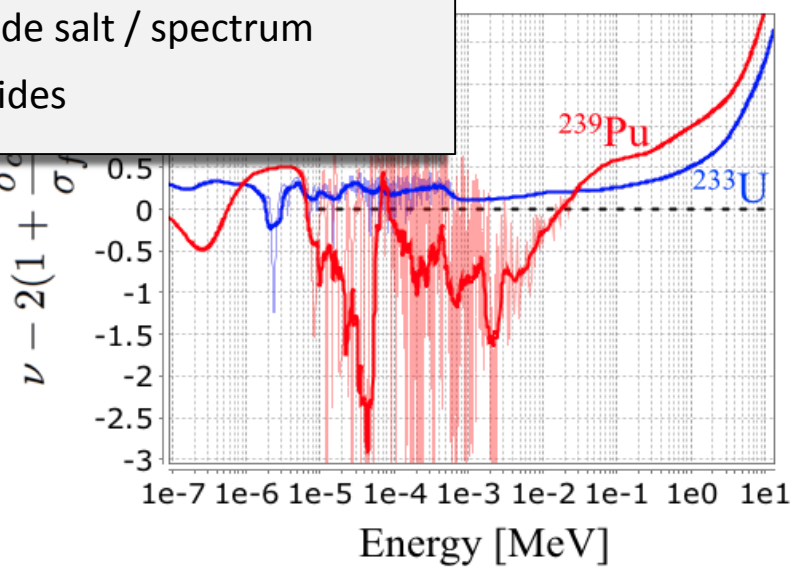
- Waste considerations (production of ^{36}Cl with chloride salt)
- Reduced irradiation damages (spectrum less fast)

Choice of the Th fuel cycle:

- Higher breeding ratio with a fluoride salt / spectrum
- Smaller production of minor actinides

cross-sections of versus neutron in the fuel cycle





Historical studies of MSR: Oak Ridge Nat. Lab. - USA

Fluoride salt and graphite matrix – Thermal neutron spectrum

- 1954 : **Aircraft Reactor Experiment (ARE)** – 2.5 MWth – Operated 1000h
- 1964 – 1969: **Molten Salt Reactor Experiment (MSRE) -** 7.4 MWth – 650°C
  ^{enr}U enriched 30% then ^{233}U and ^{239}Pu
- 1970 - 1976: **Molten Salt Breeder Reactor (MSBR)** - 2500 MWth - Project

Re-evaluation 15 years ago → **highlight some problems (safety, graphite, processing)**

MSR - Renewal of the concept – CNRS studies

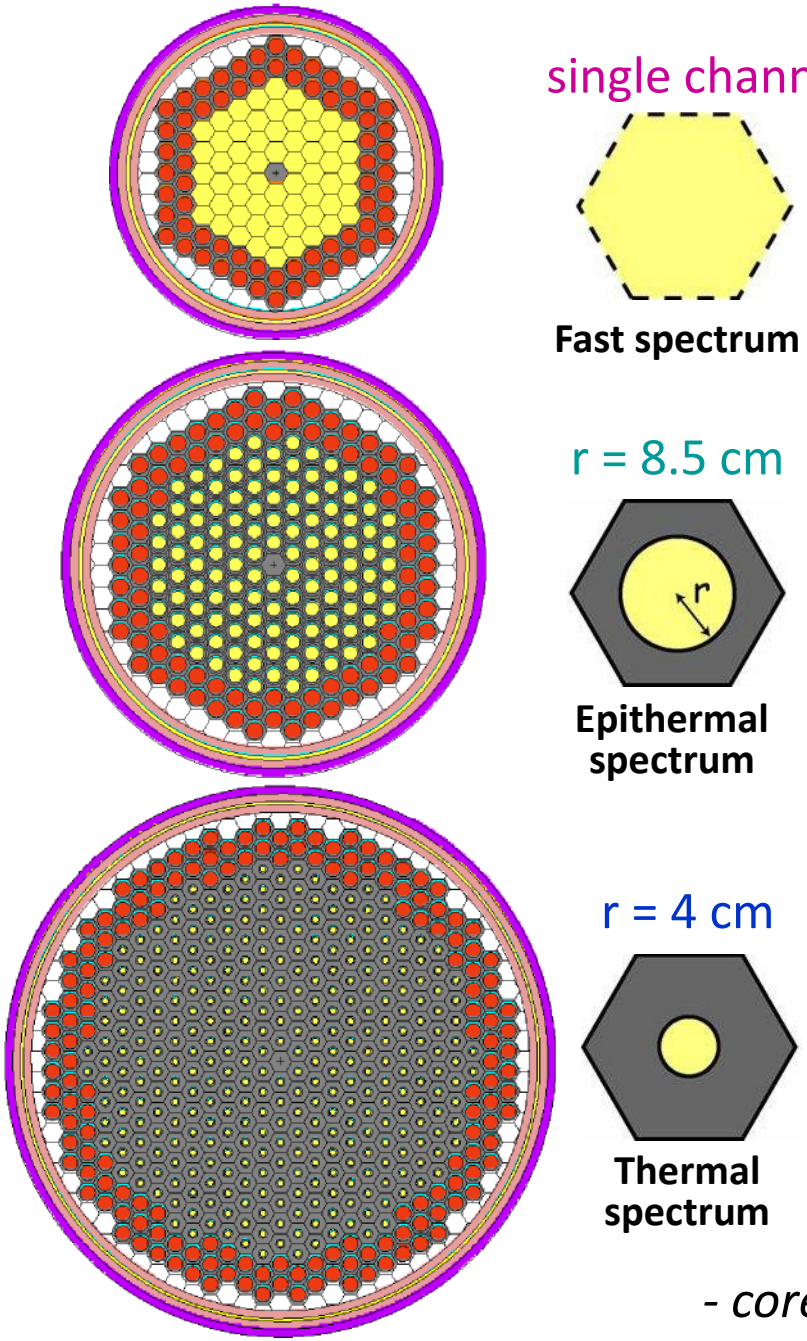
- ✓ Homogeneity of the fuel (no loading plan)
- ✓ Heat produced directly in the heat transfer fluid
- ✓ Possibility to reconfigure quickly and passively the geometry of the fuel (gravitational draining)
- ✓ Possibility to reprocess the fuel without stopping the reactor

Neutronic Optimization of MSR (Gen4 criteria) :

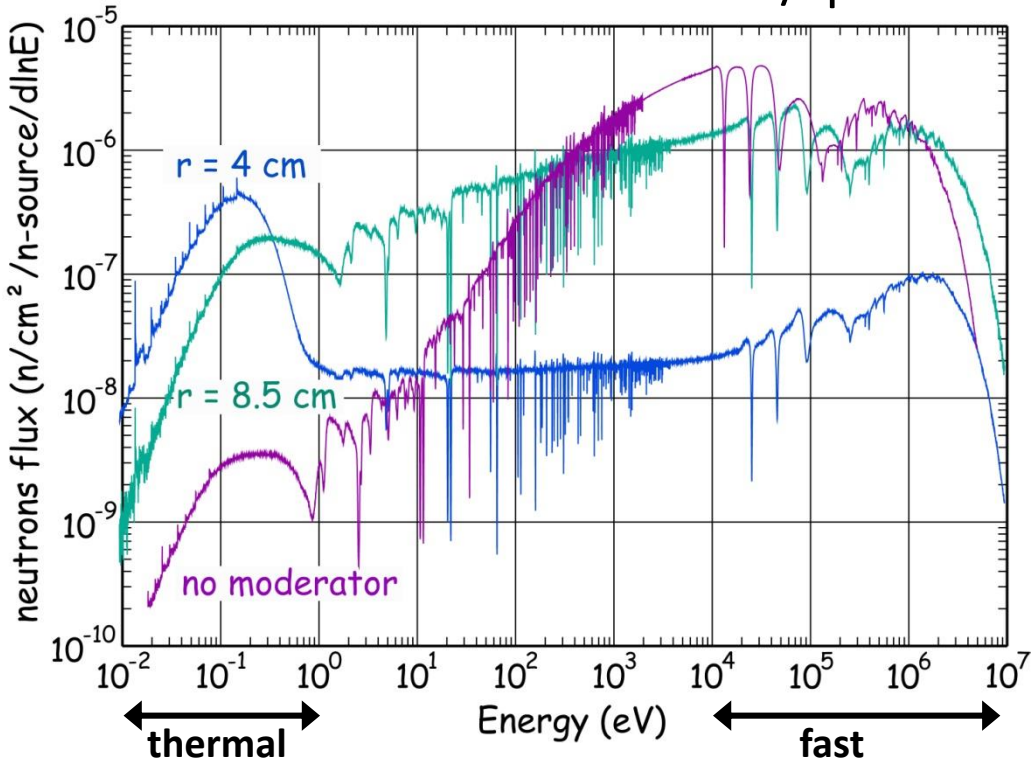
- **Safety: negative feedback coefficients**
- **Sustainability: reduce irradiation damages in the core**
- **Deployment: good breeding of the fuel + reduced initial fissile inventory**

Historical MSR Studies at CNRS
Influence of the neutron spectrum – Parametric studies

PhD thesis of
Ludovic MATHIEU



3 different moderation ratios/spectra:

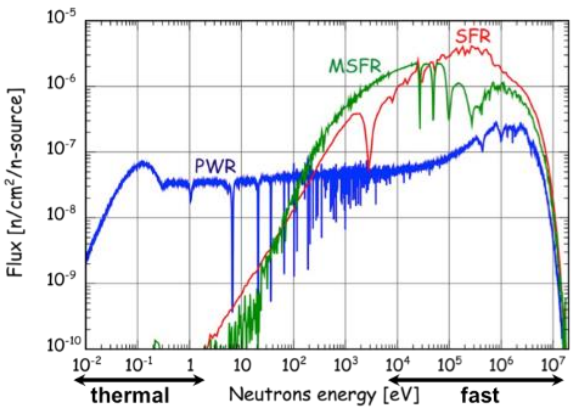


- core volume adjusted to keep the same salt volume -

Historical MSR Studies at CNRS

Thermal spectrum configurations

- positive feedback coefficient
- iso-breeder
- quite long graphite life-span
- low ^{233}U initial inventory



Epithermal spectrum configurations

- quite negative feedback coefficient
- iso-breeder
- very short graphite life-span
- quite low ^{233}U initial inventory

Fast spectrum configurations (no moderator)

- **very negative feedback coefficients**
- **very good breeding ratio**
- **no problem of graphite life-span**
- medium ^{233}U initial inventory

The Molten Salt Fast Reactor - MSFR

MSFR: Design and Fissile Inventory Optimization

Reactor Design and Fissile Inventory Optimization = Specific Power Optimization

2 parameters:
- The produced power
- The fuel salt volume and the core geometry

Liquid fuel and no solid matter inside the core ⇒ possibility to reach specific power much higher than in a solid fuel

3 limiting factors:

- The capacities of the heat exchangers in terms of heat extraction and the associated pressure drops (pumps) → *large fuel salt volume and small specific power*
- The neutronic irradiation damages to the structural materials (in Ni-Cr-W alloy) which modify their physicochemical properties. Three effects: displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium → *large fuel salt volume and small specific power*

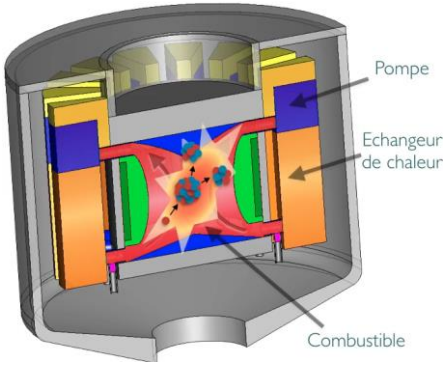
Voir présentation de M. Allibert

- The neutronic characteristics of the reactor in terms of burning efficiencies → *small fuel salt volume and large specific power* and of deployment capacities, i.e. breeding ratio (= ^{233}U production) versus fissile inventory → *optimum near 15-20 m^3 and 300-400 W/cm^3*

⇒ Reference MSFR configuration with 18 m^3 and 330 W/cm^3 corresponding to an initial fissile inventory of 3.5 tons per GWe

Concept of Molten Salt Fast Reactor (MSFR)

- ✓ Homogeneity of the fuel (no loading plan)
- ✓ Heat produced directly in the heat transfer fluid
- ✓ Possibility to reconfigure quickly and passively the geometry of the fuel (gravitational draining)
- ✓ Possibility to reprocess the fuel without stopping the reactor:



Neutronic Optimization of MSR (Gen4 criteria) :

- **Safety: negative feedback coefficients**
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2008: Definition of an innovative MSR concept based on a fast neutron spectrum, and called **MSFR (Molten Salt Fast Reactor)** by the GIF Policy Group

- ➢ **All feedback thermal coefficients negative**
- ➢ **No solid material in the high flux area: reduction of the waste production of irradiated structural elements and less in core maintenance operations**
- ➢ **Good breeding of the fissile matter thanks to the fast neutron spectrum**
- ➢ **Actinides burning improved thanks to the fast neutron spectrum**

R&D objectives

The renewal and diversification of interests in molten salts have led the MSR provisional SSC to shift the R&D orientations and objectives initially promoted in the original Generation IV Roadmap issued in 2002, in order to encompass in a consistent body the different applications envisioned today for fuel and coolant salts.

Two baseline concepts are considered which have large commonalities in basic R&D areas, particularly for liquid salt technology and materials behavior (mechanical integrity, corrosion):

- The Molten Salt Fast-neutron Reactor (MSFR) is a long-term alternative to solid-fuelled fast neutron reactors offering very negative feedback coefficients and simplified fuel cycle. Its potential has been assessed but specific technological challenges must be addressed and the safety approach has to be established.

- The AHTR is a high temperature reactor with better compactness than the VHTR and passive safety potential for medium to very high unit power.

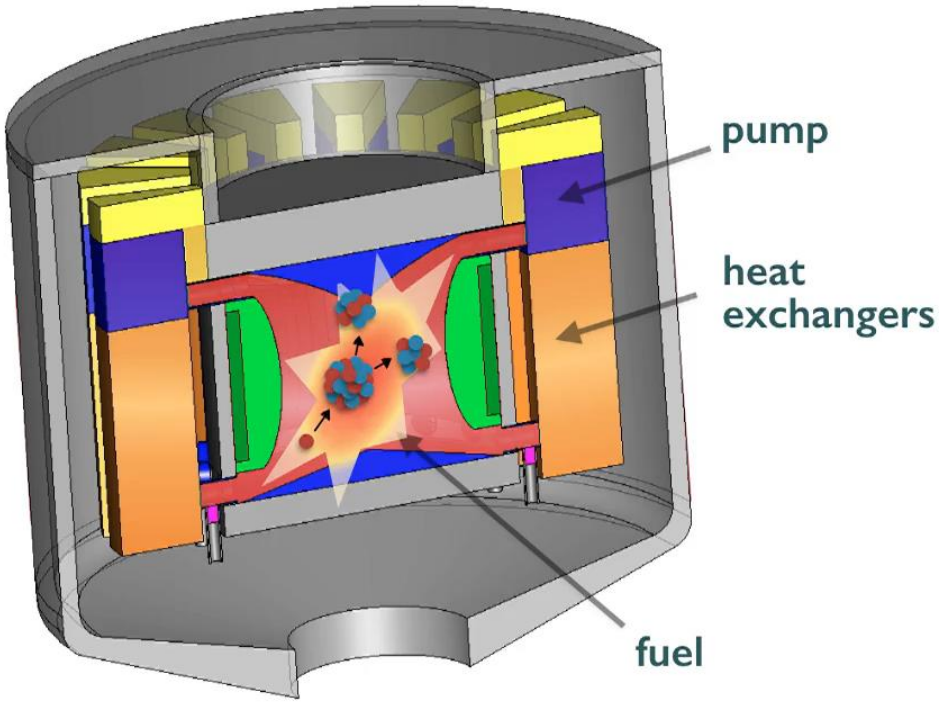
Description of the Molten Salt Fast Reactor (MSFR) system

General characteristics:

- Fuel = coolant
- Liquid circulating fuel
- Thermal yield: 45%
- Fast neutron spectrum
- Thorium fuel cycle

Three circuits:

Fuel salt circuit



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General characteristics:

- Liquid circulating fuel
- Fuel = coolant
- Thermal yield: 45%
- Fast neutron spectrum
- Thorium fuel cycle

Three circuits:

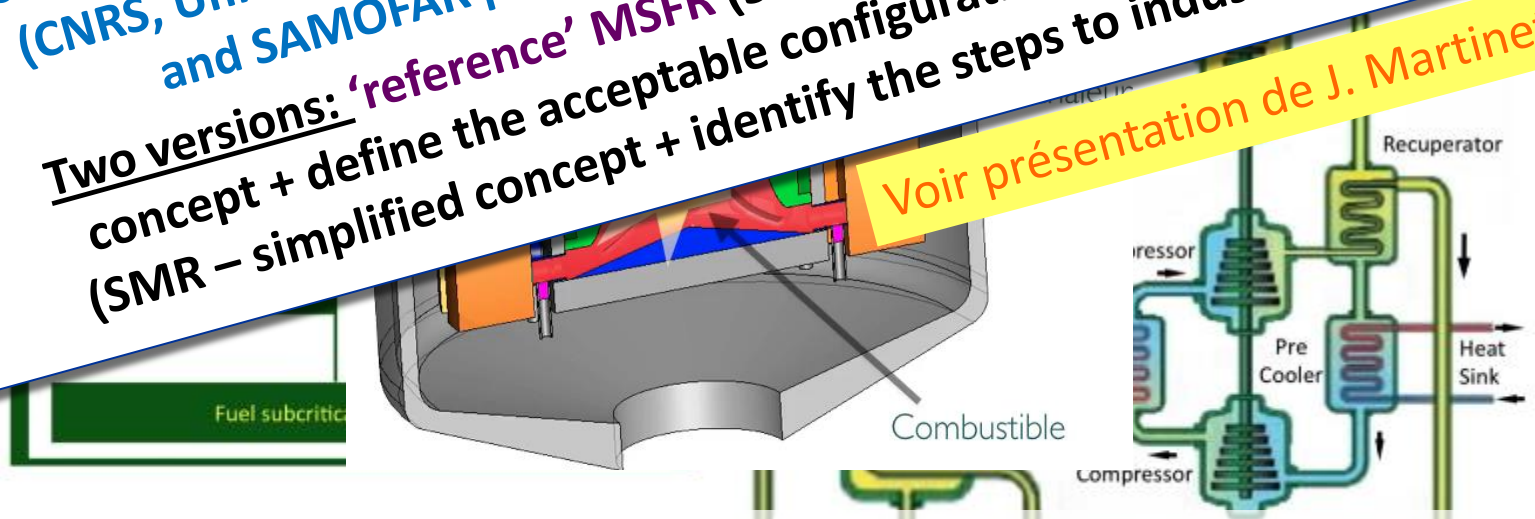
Fuel salt circuit

Interm

R&D activities: multi-disciplinary expertise (reactor physics, chemistry, safety, design, materials…) – Collaborations in a national (CNRS, Universities, AREVA, IRSN, EDF) / European (Euratom EVOL and SAMOFAR projects) / worldwide (GIF, IAEA) frame

Two versions: 'reference' MSFR (3 GWth – identify the limits of the concept + define the acceptable configurations) and Small-MSFR (SMR – simplified concept + identify the steps to industrialization)

Voir présentation de J. Martinet



M. Allibert, M. Aufiero, M. Brovchenko, S. Delpech, V. Ghetta, D. Heuer, A. Laureau, E. Merle-Lucotte, ***"Chapter 7 - Molten Salt Fast Reactors"****, Handbook of Generation IV Nuclear Reactors, Woodhead Publishing Series in Energy (2015)*

MSFR and the European project EVOL

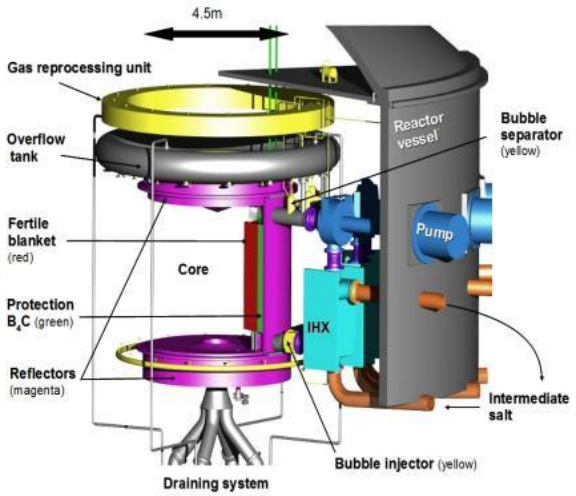
European Project "EVOL" Evaluation and Viability Of Liquid fuel fast reactor - ***FP7 (2011-2013): Euratom/Rosatom cooperation***

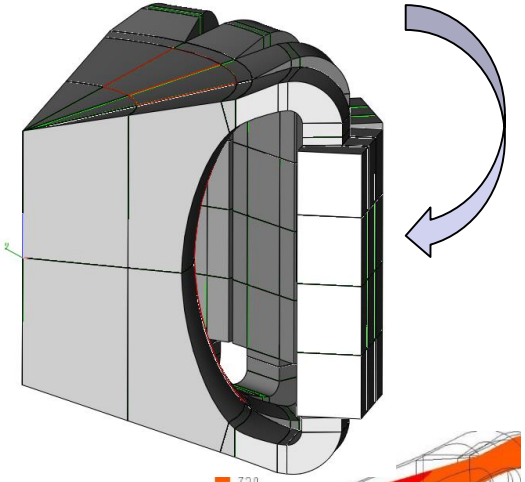
Objective : to propose a design of MSFR by end of 2013 given the best system configuration issued from physical, chemical and material studies

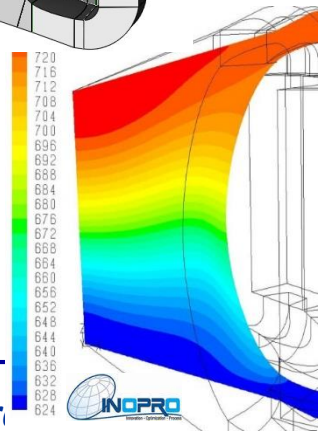
- **WP2: Design and Safety**
- **WP3: Fuel Salt Chemistry and Reprocessing**
- **WP4: Structural Materials**

Examples of outputs of the project:

- Optimized toroidal shape of the core
- Proposal for an optimized initial fuel salt composition
- Neutronic benchmark (comparison tools/ nuclear databases)
- First developments of a safety assessment method for MSR
- Recommendations for the choice of the core structural materials







12 European Partners: France (CNRS: Coordinator, Grenoble INP , INOPRO, Aubert&Duval), Netherlands (Technical Univ Delft), Germany (ITU, KIT-G, HZDR), Italy (Politecnico di Torino), UK (Oxford), Hungary (Tech Univ Budapest)
+ 2 observers since 2012: Politecnico di Milano and Paul Scherrer Institute

+ Coupled to the MARS (Minor Actinides Recycling in Molten Salt) project of ROSATOM (2011-2013)
Partners: RIAR (Dimitrovgrad), KI (Moscow), VNIITF (Snezinsk), IHTE (Ekateriburg), VNIKHT (Moscow) et MUCATEX (Moscow)

Concept of MSFR: Fuel processing

Voir présentation de S. Delpech et J. Serp

4th Generation reactors => Breeder reactors

Fuel processing mandatory to recover the produced fissile matter – Liquid fuel = reprocessing during reactor operation

Fission Products Extraction: Motivations

✓ Control physicochemical properties of the salt (control deposit, erosion and corrosion phenomena's)

✓ Keep good neutronic properties

Physical Separation (in core?)

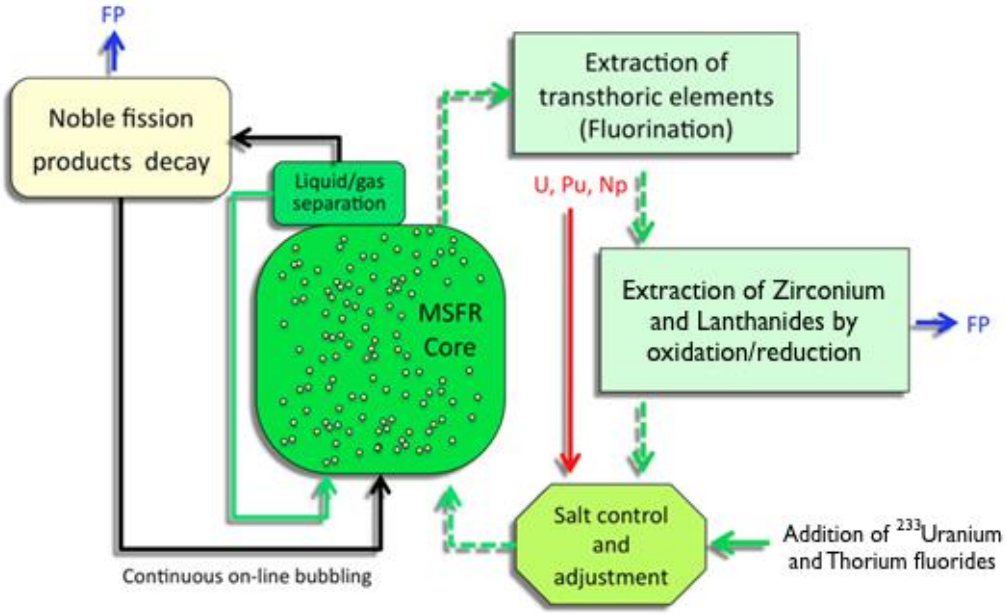
➢ Gas Processing Unit through bubbling extraction

➢ Extract Kr, Xe, He and particles in suspension

Chemical Separation (by batch)

➢ Pyrochemical processing Unit

➢ Located on-site, but outside the reactor vessel



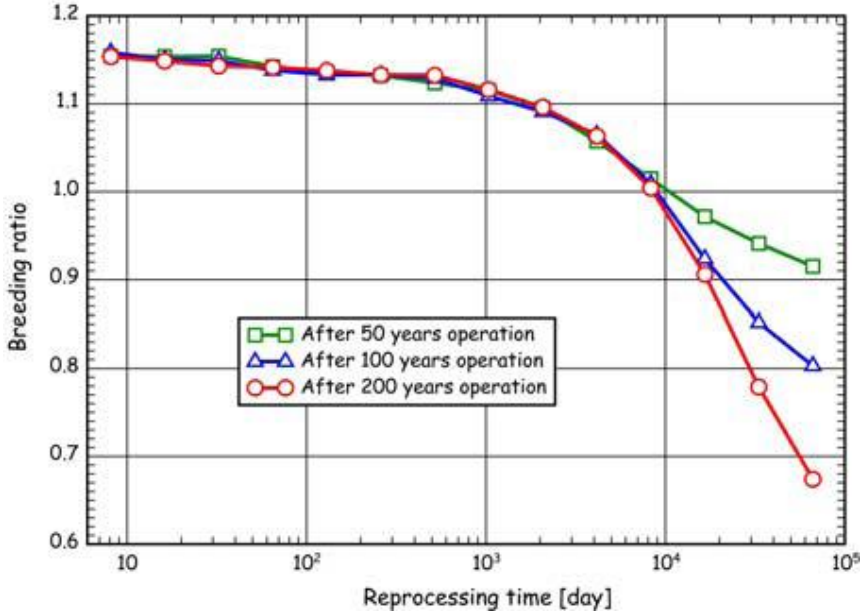
S. Delpech, E. Merle-Lucotte, D. Heuer, M. Allibert, V. Ghetta, C. Le-Brun, L. Mathieu, G. Picard, *"Reactor physics and reprocessing scheme for innovative molten salt reactor system"*, J. of Fluorine Chemistry, 130 Issue 1, p. 11-17 (2009)

Concept of MSFR: Fuel processing

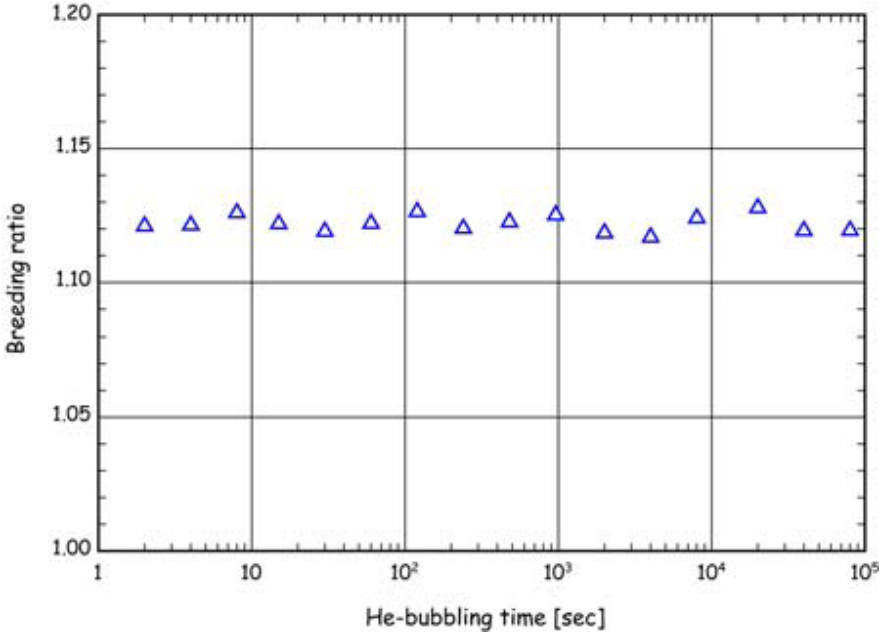
PhD thesis of Xavier DOLIGEZ

Batch chemical processing:

Element	Absorption (per fission neutron)
Heavy Nuclei	**0.9**
Alkalines	$< 10^{-4}$
Metals	0.0014
Lanthanides	0.006
Total FPs	**0.0075**



On-line (bubbling) processing:



Fast neutron spectrum

⇒ very low capture cross-sections

⇒ low impact of the processing (chemical and bubbling) on neutronics

⇒ Parallel studies of chemical and neutronic issues possible

Concept of MSFR: Starting modes and deployment capacities

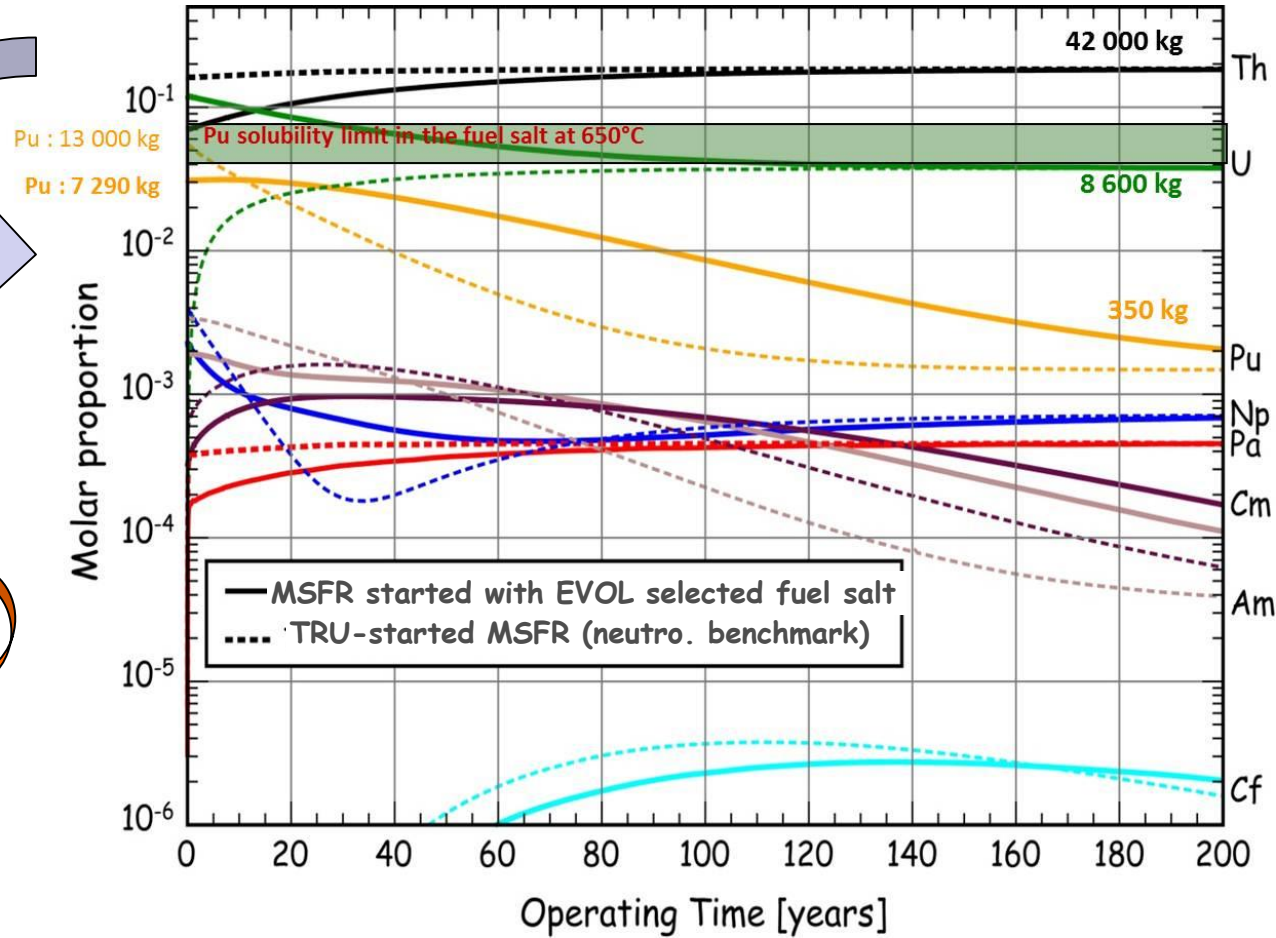
EVOL : Selection of the optimized fuel salt composition (deliverable 3.7)

Optimized initial composition of the fuel salt:
$LiF-ThF_4-UF_4-(TRU)F_3$ with (77.7-6.7-12.3-3.3 mol%) and U enriched at 13%
Density = 5085.6 - 0.8198*(T/K) - T(solid.) = 867 K

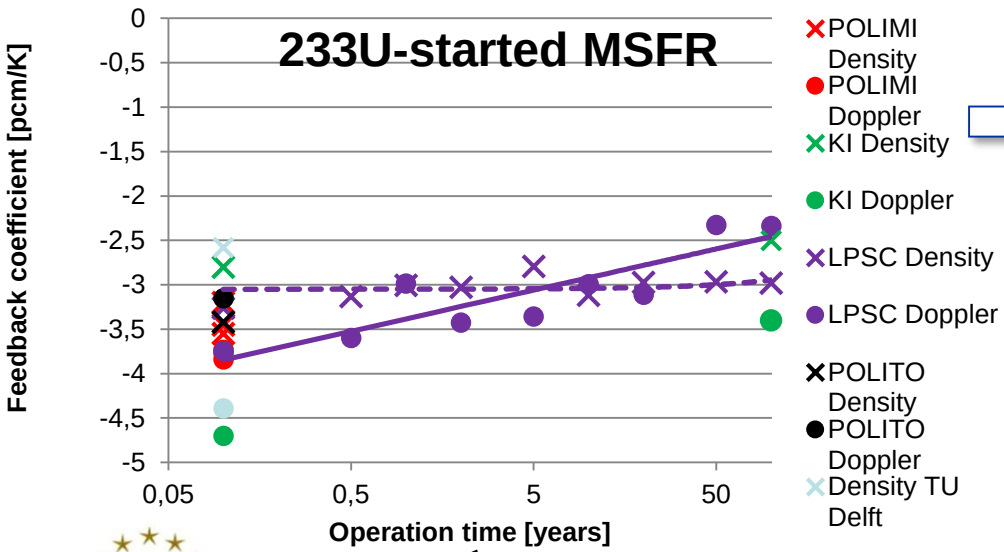
Neutronics, chemical and material behavior very satisfying

Used for scenarios studies to close the current fuel cycle and launch Gen4 MSRs

Voir présentation de D. Heuer

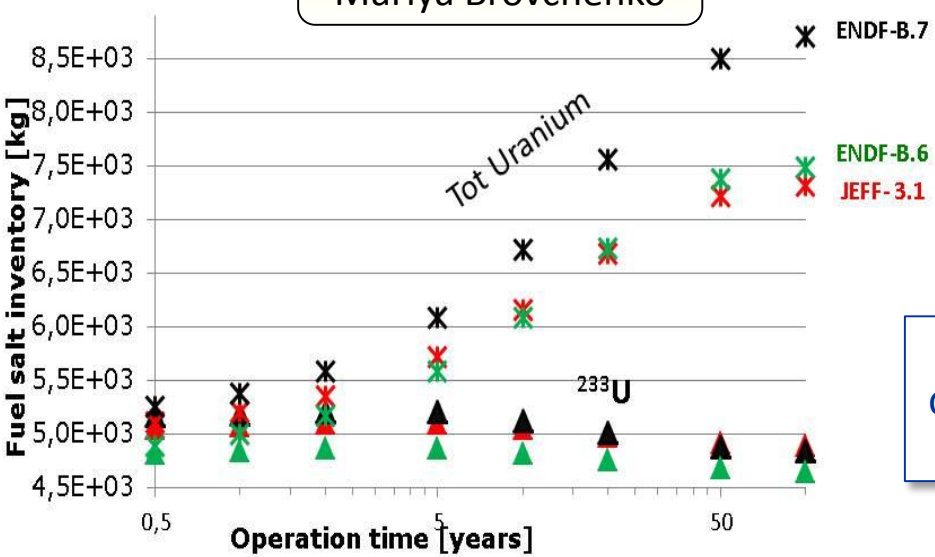


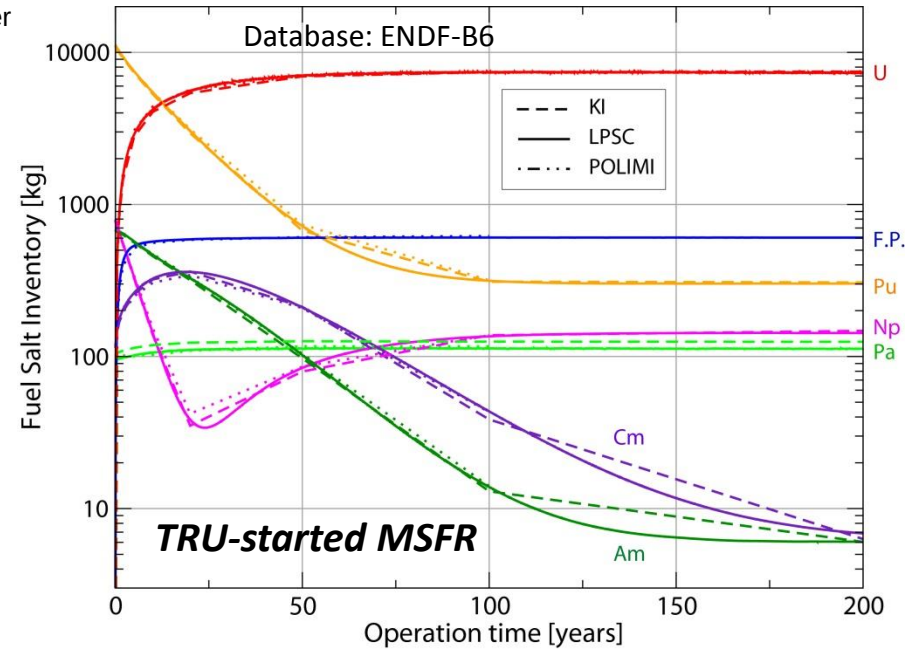
Concept of MSFR: safety coefficients and inventories



Excellent (largely negative) feedback coefficients, ∀ the simulation tool or the database used

PhD thesis of Mariya Brovchenko



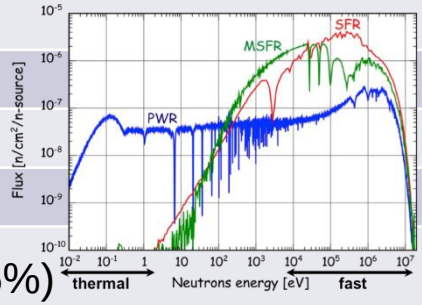


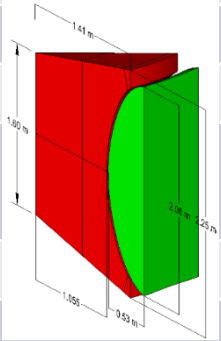
Very good agreement between the different simulation tools – Large impact of the nuclear database

“Reference MSFR” neutronic characteristics: from EVOL to SAMOFAR

Review of previous studies ⇨ list of constraints leading to the following proposal:

Parameter	Value
Thermal/electric power	3000 MWth / ~1300 MWe
Fuel salt temperature rise in the core (°C)	100
Fuel molten salt - Initial composition	$LiF-ThF_4-^{233}UF_4$ or $LiF-ThF_4-^{enr}UF_4-(Pu-MA)F_3$ with 77.5 mol% LiF
Fuel salt melting point (°C)	585
Mean fuel salt temperature (°C)	725
Fuel salt density (g/cm^3)	4.1
Fuel salt dilation coefficient (g.cm^{-3}/°C)	8.82 10^{-4}
Fertile blanket salt - Initial composition (mol%)	$LiF-ThF_4$ (77.5%-22.5%)
Breeding ratio (steady-state)	1.1
Total feedback coefficient (pcm/°C)	-8
Toroidal core dimensions (m)	Radius: 1.06 to 1.41 Height: 1.6 to 2.26
Fuel salt volume (m^3)	18 (1/2 in the core)
Total fuel salt cycle in the fuel circuit	3.9 s
Intermediate fluid	fluoroborate ($8NaF-92NaBF_4$), FLiNaK, $LiF-ZrF_4$, FLiBe
Structural materials	Ni-based alloy (Hastelloy-N)





Concept of Molten Salt Fast Reactor (MSFR)

SAMOFAR Project – Horizon2020
Safety Assessment of a MOlten salt FAst Reactor

4 years (2015-2019), 3,5 M€

Partners: TU-Delft (leader), CNRS, JRC-ITU, CIRTEN (POLIMI, POLITO), IRSN, AREVA, CEA, EDF, KIT + PSI + CINVESTAV

SAMOFAR will deliver the experimental proof of the following **key safety features**:
The **freeze plug** and draining of the fuel salt
New materials and new coatings to materials
Measurement of safety related data of the fuel salt
The dynamics of **natural circulation** of (internally heated) fuel salts
The **reductive extraction processes** to extract lanthanides and actinides from the fuel salt

5 technical work-packages:

WP1 Integral safety approach and system integration

WP2 Physical and chemical properties required for safety analysis

WP3 Proof of concept of key safety features

WP4 Numerical assessment of accidents and transients

WP5 Safety evaluation of the chemical processes and plant

Concept of MSFR: SAMOFAR WP1 ”Integral safety approach and system integration”

SAMOFAR

Del. n°	Deliverable title	Lead benef.	Delivery date
D1.1	Description of initial reference design and identification of safety aspects	CNRS	Month 6
D1.2	Identifying safety related physico-chemical and material data	JRC	Month 6
D1.3	Development of a power plant simulator	CNRS	Month 24
D1.4	Safety issues of normal operation conditions, including start, shut-down and load-following	CIRTEN	Month 30
D1.5	Development on an integral safety assessment methodology for MSR	IRSN	Month 36
D1.6	Identification of risks and phenomena involved, identification of accident initiators and accident scenarios	CIRTEN	Month 36
D1.7	Improved Integral power plant design (reactor core and chemical plant) to maximize safety and proposal for safety demonstrator	CNRS	Month 48

Specificities of the MSFR impacting the safety analysis

PhD theses of Mariya Brovchenko, Axel Laureau and Delphine Gérardin

- Liquid fuel
  - ✓ Molten fuel salt acts as reactor fuel and coolant
  - ✓ Relative uniform fuel irradiation
  - ✓ A significant part of the fissile inventory is outside the active area where fissions occur (but in the core vessel)
  - ✓ Fuel
- No cont
  - ✓ Re ... cients, co
  - ✓ No ... tion, et
- Fuel sal
  - ✓ Emergency shutdown obtained by draining quickly the molten salt from the fuel circuit
  - ✓ Changing the fuel geometry allows for adequate shutdown margin and cooling
  - ✓ Fuel draining can be done passively or by operator action in 2 dedicated systems (normal operation and emergency draining systems)

- ❖ Design definition (core and draining system at least)
- ❖ Development of simulation tools dedicated (more generic)
- ❖ Definition of the normal operation procedures
- ❖ Safety evaluation: accident initiators? Accident scenarios?
- ❖ Safety approach: severe accident? Barriers? Reactivity control?

M. Brovchenko, D. Heuer, E. Merle-Lucotte, M. Allibert, V. Ghetta, A. Laureau, P. Rubiolo, *"Design-related Studies for the Preliminary Safety Assessment of the Molten Salt Fast Reactor"*, Nuclear Science and Engineering, **175**, 329–339 (2013)

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  - ✓ Molten fuel salt acts as reactor fuel and coolant
  - ✓ Relative uniform fuel irradiation
  - ✓ A significant part of the fissile inventory is outside the active area where fissions occur (but in the core vessel)
  - ✓ Fuel
- No cont
  - ✓ Re ... cients, co
  - ✓ No ... ion, et
- Fuel sal
  - ✓ Emergency shutdown obtained by draining quickly the molten salt from the fuel circuit
  - ✓ Changing the fuel geometry allows for adequate shutdown margin and cooling
  - ✓ Fuel draining can be done passively or by operator action in 2 dedicated systems (normal operation and emergency draining systems)

- ❖ Design definition (core and draining system at least)
- ❖ Development of simulation tools dedicated (more generic)
- ❖ Definition of the normal operation procedures
- ❖ Safety evaluation: accident initiators? Accident scenarios?
- ❖ Safety approach: severe accident? Barriers? Reactivity control?

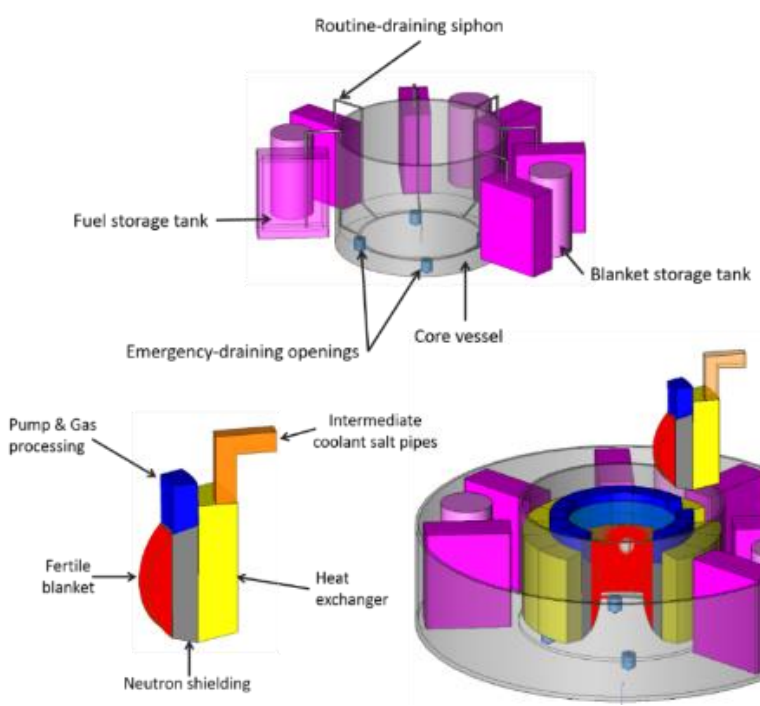
M. Brovchenko, D. Heuer, E. Merle-Lucotte, M. Allibert, V. Ghetta, A. Laureau, P. Rubiolo, *"Design-related Studies for the Preliminary Safety Assessment of the Molten Salt Fast Reactor"*, Nuclear Science and Engineering, **175**, 329–339 (2013)

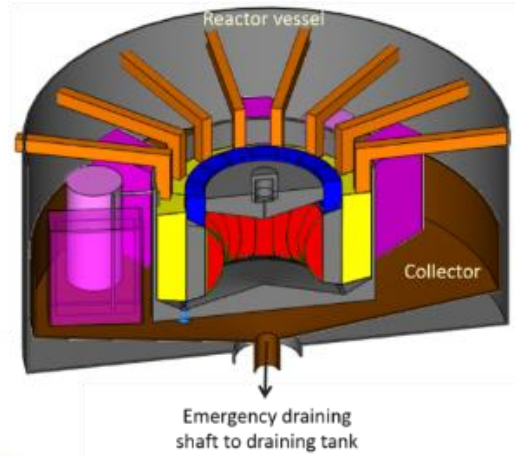
Design aspects impacting the MSFR safety analysis

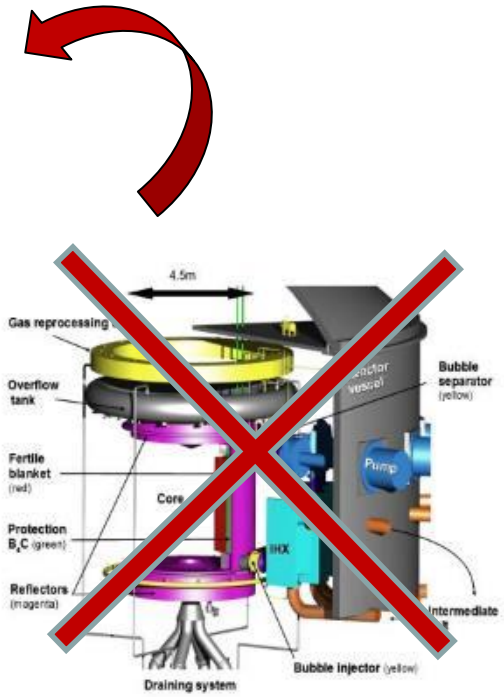
LOLF accident (Loss of Liquid Fuel) → no tools available for quantitative analysis but qualitatively:

- Fuel circuit: complex structure, multiple connections
- Potential leakage: collectors connected to draining tank

→ Proposition of an '<u>Integrated MSFR design</u>' to suppress pipes/leakage







Specificities of the MSFR impacting the safety analysis

PhD theses of Mariya Brovchenko, Axel Laureau and Delphine Gérardin

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Voir présentation de A. Laureau

M. Brovchenko, D. Heuer, E. Merle-Lucotte, M. Allibert, V. Ghetta, A. Laureau, P. Rubiolo, *"Design-related Studies for the Preliminary Safety Assessment of the Molten Salt Fast Reactor"*, Nuclear Science and Engineering, **175**, 329–339 (2013)

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Voir présentation de S. Beils et D. Gérardin

M. Brovchenko, D. Heuer, E. Merle-Lucotte, M. Allibert, V. Ghetta, A. Laureau, P. Rubiolo, *"Design-related Studies for the Preliminary Safety Assessment of the Molten Salt Fast Reactor"*, Nuclear Science and Engineering,**175**, 329–339 (2013)

Demonstration steps and Demonstrator of MSFR

Sizing of the facilities:

Small size: ~1liter - chemistry and corrosion – off-line processing

Pyrochemistry: basic chemical data, processing, monitoring

Medium size: ~100 liters – hydrodynamics, noble FP extraction, heat exchanges

Process analysis, modeling, technology tests

Full size experiment: ~1 m^3 salt / loop – validation at loop scale

Validation of technology integration and hydrodynamics models

3 levels of radio protection:

- ✓ Inactive simulant salt ⇨ Standard laboratory

  Hydrodynamics, material, measurements, model validation
- ✓ Low activity level (Th, depleted U) ⇨ Standard lab + radio protect

  Pyrochemistry, corrosion, chemical monitoring
- ✓ High activity level ($^{enriched}U$, ^{233}U, Pu, MA) ⇨ Nuclear facility

  Fuel salt processing: Pyrochemistry, , Actinides recycling

Some PhD Thesis in France on MSR

Gabriela DURAN, ***"Étude du comportement de l'uranium et de l'iode dans le mélange de fluorures fondus LiF-ThF$_4$ à 650°C"***, PhD Thesis, Université Paris Saclay, France (2017)

Axel LAUREAU, ***"Développement de modèles neutroniques pour le couplage thermohydraulique du MSFR et le calcul de paramètres cinétiques effectifs"***, PhD Thesis, Grenoble Alpes University, France (2015)

Davide RODRIGUES, ***"Solvatation du thorium par les fluorures en milieu sel fondu à haute température : application au procédé d'extraction réductrice pour le concept MSFR"***, PhD Thesis, Paris Sud University (2015)

Mariya BROVCHENKO, ***"Etudes préliminaires de sûreté du réacteur à sels fondus MSFR"***, PhD Thesis, Grenoble Institute of Technology, France (2013)

Xavier DOLIGEZ, ***"Influence du retraitement physico-chimique du sel combustible sur le comportement du MSFR et sur le dimensionnement de son unité de retraitement",*** PhD Thesis, Grenoble Institute of Technology and EDF, France (2010)

Elsa MERLE-LUCOTTE, ***"Le cycle Thorium en réacteurs à sels fondus peut-il être une solution au problème énergétique du XXIème siècle ? Le concept de TMSR-NM",*** Habilitation à Diriger les Recherches, Grenoble INP, France (2008)

Ludovic MATHIEU, ***"Cycle Thorium et Réacteurs à Sel Fondu: Exploration du champ des Paramètres et des Contraintes définissant le Thorium Molten Salt Reactor",*** PhD Thesis, Grenoble Institute of Technology and EDF, France (2005)

Jorgen FINNE, ***"Chimie des mélanges de sels fondus - Application à l'extraction réductrice d'actinides et de lanthanides par un métal liquide"***, PhD Thesis, EDF-CEA-ENSCP, Paris, France (2005)

Fabien PERDU, ***"Contributions aux études de sûreté pour des filières innovantes de réacteurs nucléaires",*** PhD Thesis, Grenoble Institute of Technology, France (2003)

Alexis NUTTIN, ***"Potentialités du concept de réacteur à sels fondus pour une production durable d'énergie nucléaire basée sur le cycle thorium en spectre épithermique",*** PhD Thesis, Grenoble I University and EDF, France (2002)

David LECARPENTIER, ***"Le concept AMSTER, aspects physiques et sûreté"***, EDF and CNAM, Paris, France (2001)

Available on http://lpsc.in2p3.fr/index.php/fr/38-activites-scientifiques/physique-des-reacteurs-nucleaires/183-msfr-bibliographie or 'MSFR LPSC' in google search

Some documents mentioning the MSRF

MSR-Safety White Paper, Gen4 International Forum, SSC-MSR, under review (2017)

M. Allibert, M. Aufiero, M. Brovchenko, S. Delpech, V. Ghetta, D. Heuer, A. Laureau, E. Merle-Lucotte, ***"Chapter 7 - Molten Salt Fast Reactors",*** Handbook of Generation IV Nuclear Reactors, Woodhead Publishing Series in Energy (2015)

"Introduction of Thorium in the Nuclear Fuel Cycle", Nuclear Science 2015, NEA website https://www.oecd-nea.org/science/pubs/2015/7224-thorium.pdf (2015)

J. Serp, M. Allibert, O. Beneš, S. Delpech, O. Feynberg, V. Ghetta, D. Heuer, D. Holcomb, V. Ignatiev, J.L. Kloosterman, L. Luzzi, E. Merle-Lucotte, J. Uhlíř, R. Yoshioka, D. Zhimin, ***"The molten salt reactor (MSR) in generation IV: Overview and Perspectives",*** Prog. Nucl. Energy, 1-12 (2014)

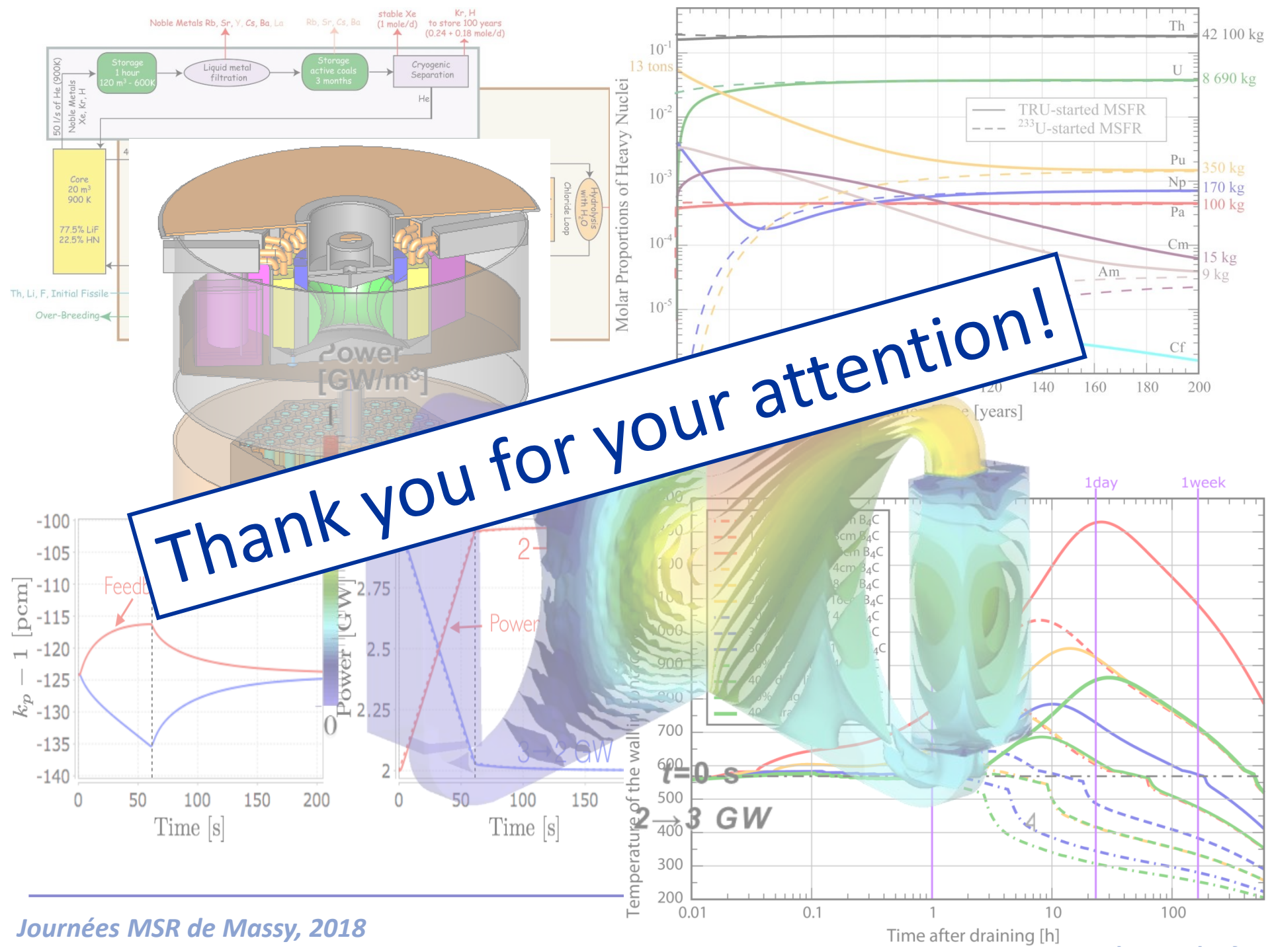
H. Boussier, S. Delpech, V. Ghetta, D. Heuer, D.E. Holcomb, V. Ignatiev, E. Merle-Lucotte, J. Serp, ***"The Molten Salt Reactor in Generation IV: Overview and Perspectives"***, Proceedings of the Generation4 International Forum Symposium, San Diego, USA (2012)

CEA, Rapport sur la gestion durable des matières nucléaires - Tome 4 : Les autres filières à neutrons rapides de 4ème génération (2012)

C. Renault, S. Delpech, E. Merle-Lucotte, R. Konings, M. Hron, V. Ignatiev, ***"The molten salt reactor: R&D status and perspectives in Europe"***, Poceedings of FISA2009: 7th European Commission conference on EURATOM research and training in reactor systems, Prague, Tchéquie (2009)

SNETP
SUSTAINABLE NUCLEAR ENERGY TECHNOLOGY PLATFORM

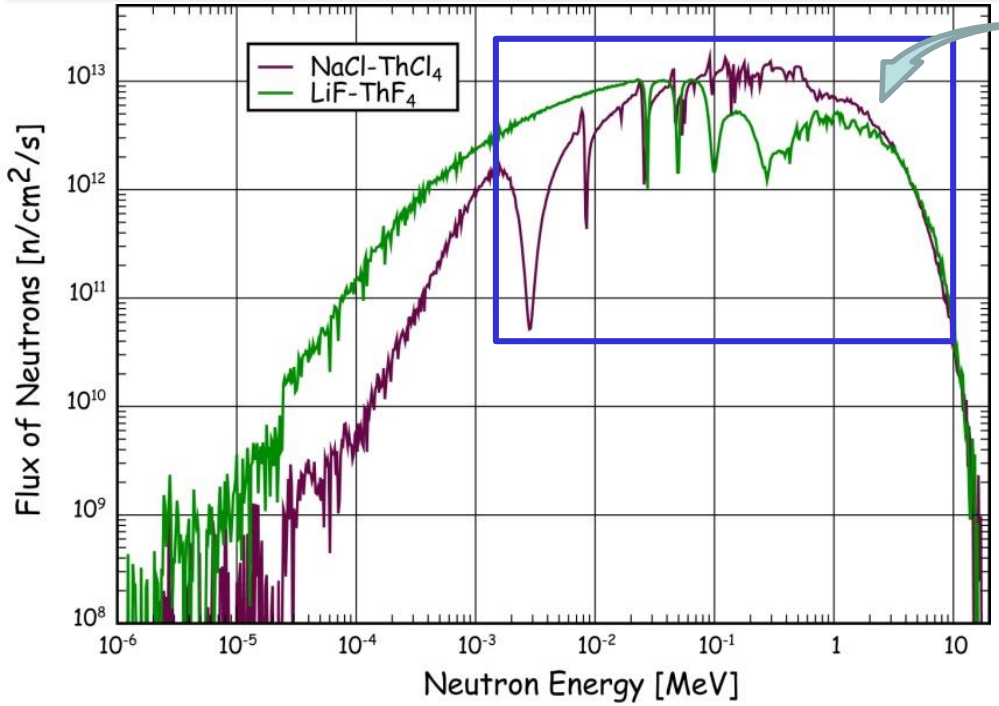
See also the annex on Molten Salt Reactor Systems of the Strategic Research Agenda (published in January 2012)
Agenda of the SNETP (Sustainable Nuclear Energy Technology Platform of Europe) here: http://*www.snetp.eu/www/snetp/.../sra_annex-MSRS.pdf*

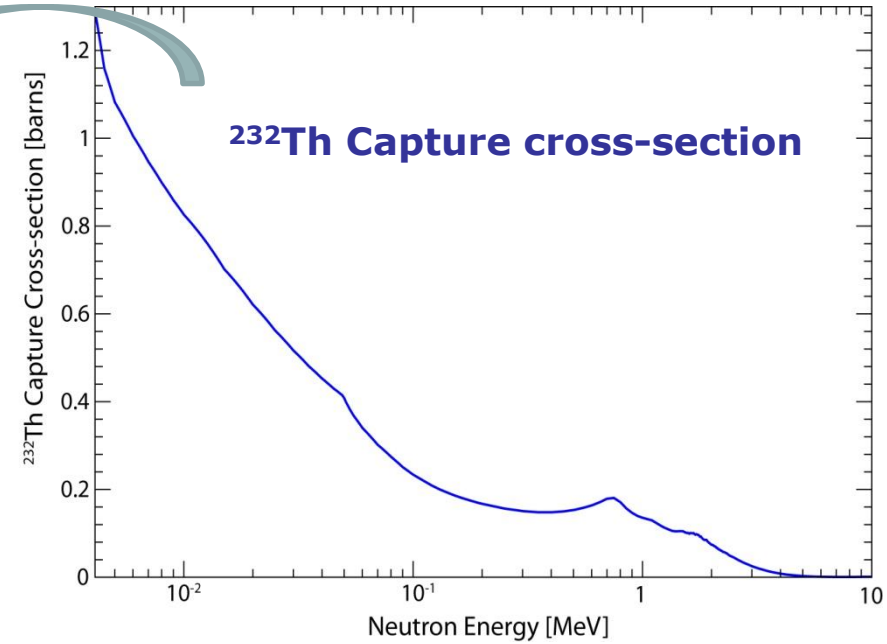

Noble Metals Rb, Sr, Y, Cs, Ba, La
Rb, Sr, Cs, Ba
stable Xe (1 mole/d)
Kr, H to store 100 years (0.24 + 0.18 mole/d)
Storage 1 hour 120 m³ - 600K
Liquid metal filtration
Storage active coals 3 months
Cryogenic Separation
He
50 l/s of He (900K)
Noble Metals Xe, Kr, H
Core 20 m³ 900 K
77.5% LiF 22.5% HN
Th, Li, F, Initial Fissile
Over-Breeding
Chloride Loop
Hydrolysis with H2O
Power [GW/m³]
Molar Proportions of Heavy Nuclei
13 tons
Th
42 100 kg
U
8 690 kg
TRU-started MSFR
233U-started MSFR
Pu
350 kg
Np
170 kg
Pa
100 kg
Cm
15 kg
Am
9 kg
Cf
[years]
Thank you for your attention!
Feed
Power
Time [s]
Power [GW]
3 → 2 GW
1day
1week
t=0 s
2 → 3 GW
Temperature of the wall
Time after draining [h]

Liquid-fueled reactors: fluoride or chloride? *Chemistry*

Chemical characteristics
Chloride salts less studied that fluoride salts for nuclear applications
Chloride salts: some have a lower melting temperature (50/100K) and larger actinides solubilities
Chloride salts do not dissolve oxides
Chloride salts are more difficult to dehydrate → more H_2O contamination and thus more corrosion risk
More processes of separative chemistry for chloride salts

Liquid-fueled reactors: fluoride or chloride? *Breeding*

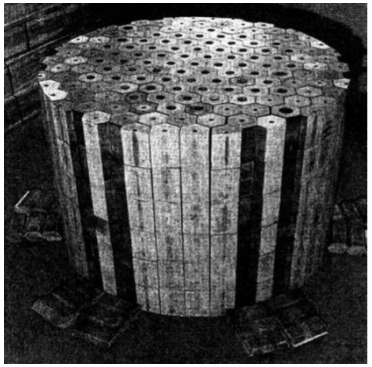




Parameter	Fluoride Salt	Chloride Salt
Thorium capture cross-section in core (barn)	0.61	0.315
Thorium amount in core (kg)	42 340	47 160
Thorium capture rate in core (mole/day)	11.03	8.48
Thorium capture cross-section in blanket (barn)	0.91	0.48
Thorium amount in the blanket (kg)	25 930	36 400
Thorium capture rate in the blanket (mole/day)	1.37	2.86
^{233}U initial inventory (kg)	5720	6867
Neutrons per fission ν in core	2.50	2.51
^{233}U capture cross-section in core (barn)	0.495	0.273
^{233}U fission cross-section in core (barn)	4.17	2.76
Capture/fission ratio α (spectrum-dependent)	0.119	0.099
Total breeding ratio	1.126	1.040

Molten Salt Reactors – Historical Studies

Historical studies of MSR: Oak Ridge Nat. Lab. - USA



- 1954 : **Aircraft Reactor Experiment (ARE)**
  Operated during 1000 hours
  Power = 2.5 MWth

- 1964 – 1969: **Molten Salt Reactor Experiment (MSRE)**
  Experimental Reactor
  Power: 7.4 MWth
  Temperature: 650°C
  U enriched 30% (1966 - 1968)
  ^{233}U (1968 – 1969) - ^{239}Pu (1969)
  No Thorium inside

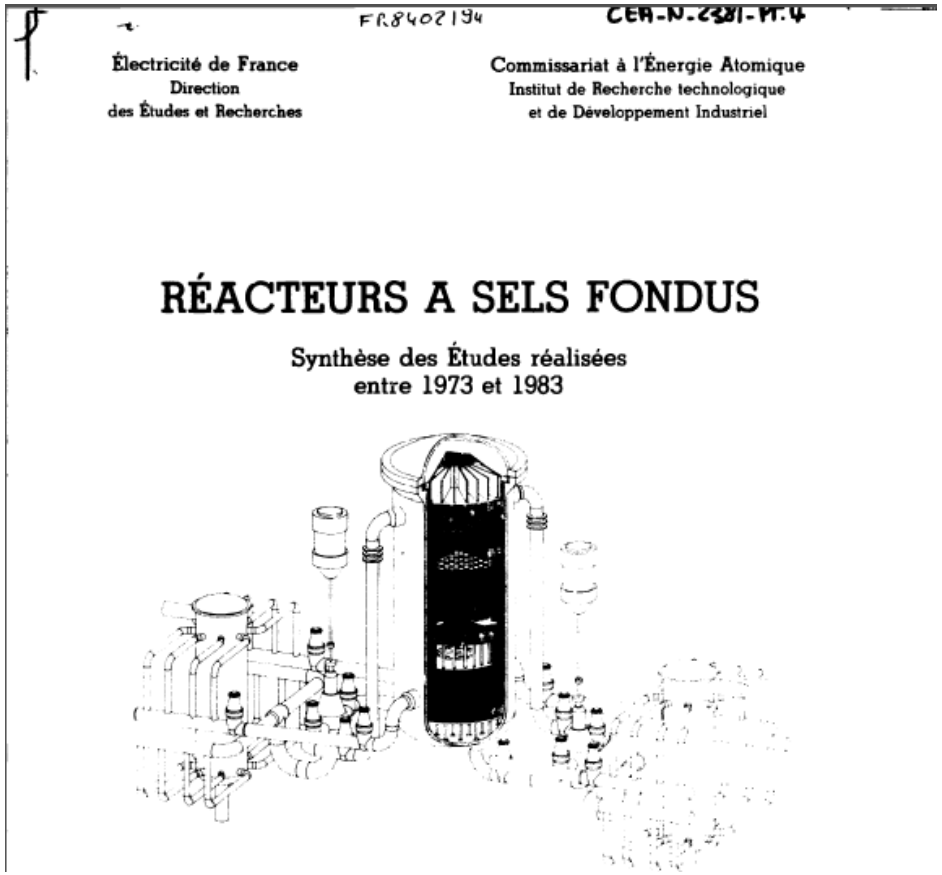
- 1970 - 1976: **Molten Salt Breeder Reactor (MSBR)**
  Never built
  Power: 2500 MWth
  Thermal neutron spectrum

Molten Salt Reactors - Renewal of the concept

- Thorims-NES5 then FUJI-AMSB in Japan since the 80's

  Reactor of very low specific power fed with ^{233}U produced in sub-critical reactors

- Resumption of the MSBR's studies by CEA and EDF – 70's to 80's



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Électricité de France
Direction
des Études et Recherches

Commissariat à l'Énergie Atomique
Institut de Recherche technologique
et de Développement Industriel

RÉACTEURS A SELS FONDUS

Synthèse des Études réalisées
entre 1973 et 1983

Molten Salt Reactors - Renewal of the concept

Past studies on Molten Salt Reactors at EDF R&D since 2000

At the begining of the 2000'

1. Study of actinide Burner on thorium support AMSTER (adaptation of the MSBR) – 2001

From 2001 to 2005 Thorium breeder in thermal spectrum considered as a better option for sustainalitity.

1. In-house research : neutronic / processing / corrosion
2. Support to CNRS and CEA research
3. Contribution to European projects
   MOST (2001), ALISIA (2007), SAMOFAR (2015)

Since 2005 non moderated Thorium breeder as most promizing option for sustainability

Contact person: David LECARPENTIER, EDF R&D

Molten Salt Reactors - Renewal of the concept

- Thorims-NES5 then FUJI-AMSB in Japan since the 80's
  Reactor of very low specific power fed with ^{233}U produced in sub-critical reactors
- Resumption of the MSBR's studies by CEA and EDF since the 90's
- TIER-1 project of C. Bowman in the 90's
  Pu burner (LWR's spent fuel assemblies dissolved in liquid fuel) in sub-critical reactors
- TASSE (CEA) project in the 90's
  Plutonium burner (liquid fuel) in sub-critical reactors
- AMSTER (EDF) project in the 90's
  Plutonium burner then breeder reactor in Thorium cycle
- REBUS (EDF), MOSART (Kurchakov Institute), SPHINX (Czech Republic)
  Projects of actinide burners
- MOST Network, FP5, 2001-2004
  European network having assessed the studies, the experiments and the state of knowledge concerning molten salt reactors
- ALISIA (Assessment of Liquid Salts for Innovative Applications), FP6, 2007-2008
  European Action - Lead authors : O.Bene C. Cabet, S. Delpech, P. Hosnedl, V. Ignatiev, R. Konings, D. Lecarpentier, O. Matal, E. Merle-Lucotte, C. Renault, J. Uhlir, 6st Euratom Framework Prog.

MSR - Renewal of the concept – CNRS studies

- Participation to the project TIER I of C. Bowman (1998)

- Re-evaluation of the MSBR from 1999 to 2002

  PhD thesis of Alexis NUTTIN

  Use of a probabilistic neutronic code (MCNP)

  Development of an in-house evolution code for materials (REM)

  Coupling of the neutronic code with the evolution code

- From the Thorium Molten Salt Reactor to the Molten Salt Fast Reactor

  Breeder in the Thorium fuel cycle and Actinide Burner Reactor

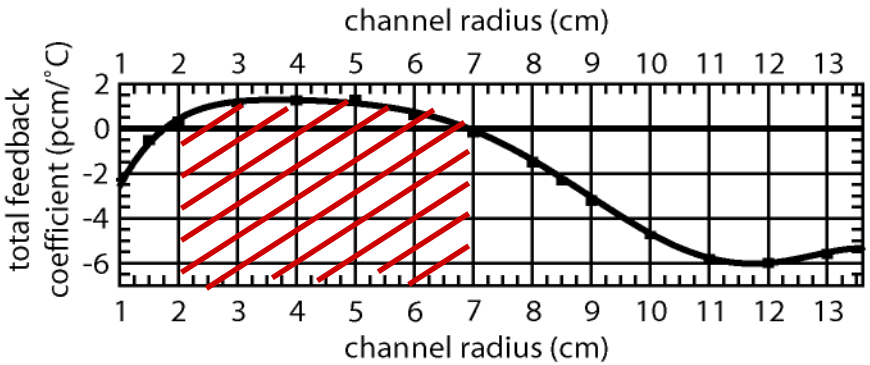
  Developed to solve the problems of the MSBR project

  - Bad (null to positive) feedback coefficients
  - Positive void coefficient
  - Unrealistic reprocessing
  - Problems specific to the graphite moderator
    - Lifespan
    - Reprocessing and storage
    - Fire risk

Historical MSR Studies at CNRS
Influence of the neutron spectrum on the core behavior

Influence studied through
4 core characteristics:

- Total feedback coefficient



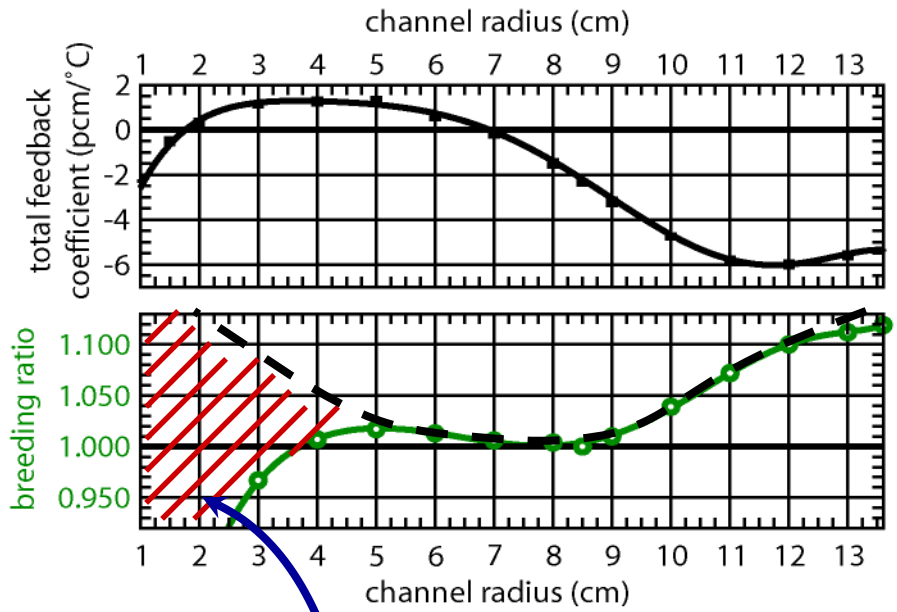
Total feedback coefficient:
Ok if <0 at least

Historical MSR Studies at CNRS
Influence of the neutron spectrum on the core behavior

Influence studied through
4 core characteristics:

- Total feedback coefficient

- Breeding ratio:



$^{232}Th/^{233}U$ cycle: neutron balance better in thermal or fast spectrum

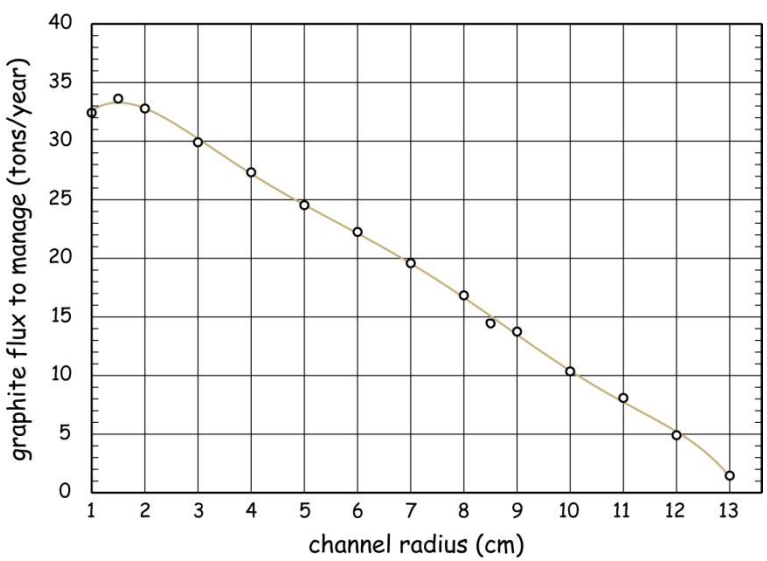
+

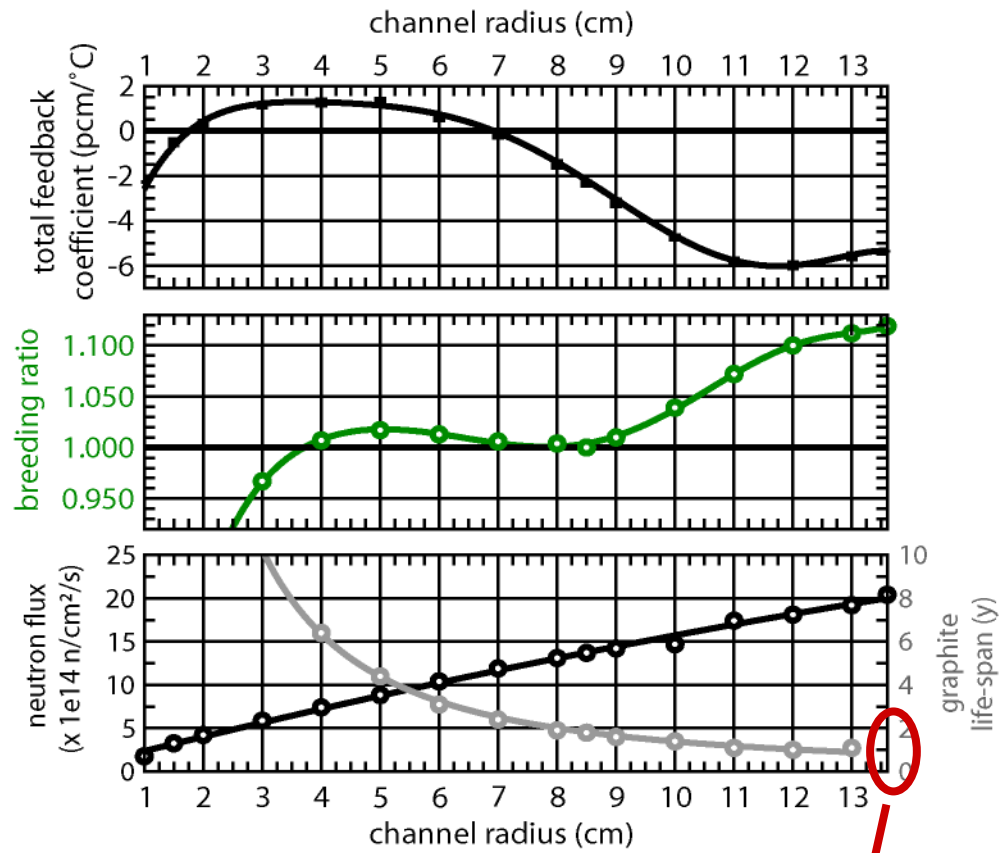
neutron captures in moderator in thermal spectrum

Historical MSR Studies at CNRS
Influence of the neutron spectrum on the core behavior

Influence studied through
4 core characteristics:

- Total feedback coefficient
- Breeding ratio
- Neutron flux / Graphite lifespan





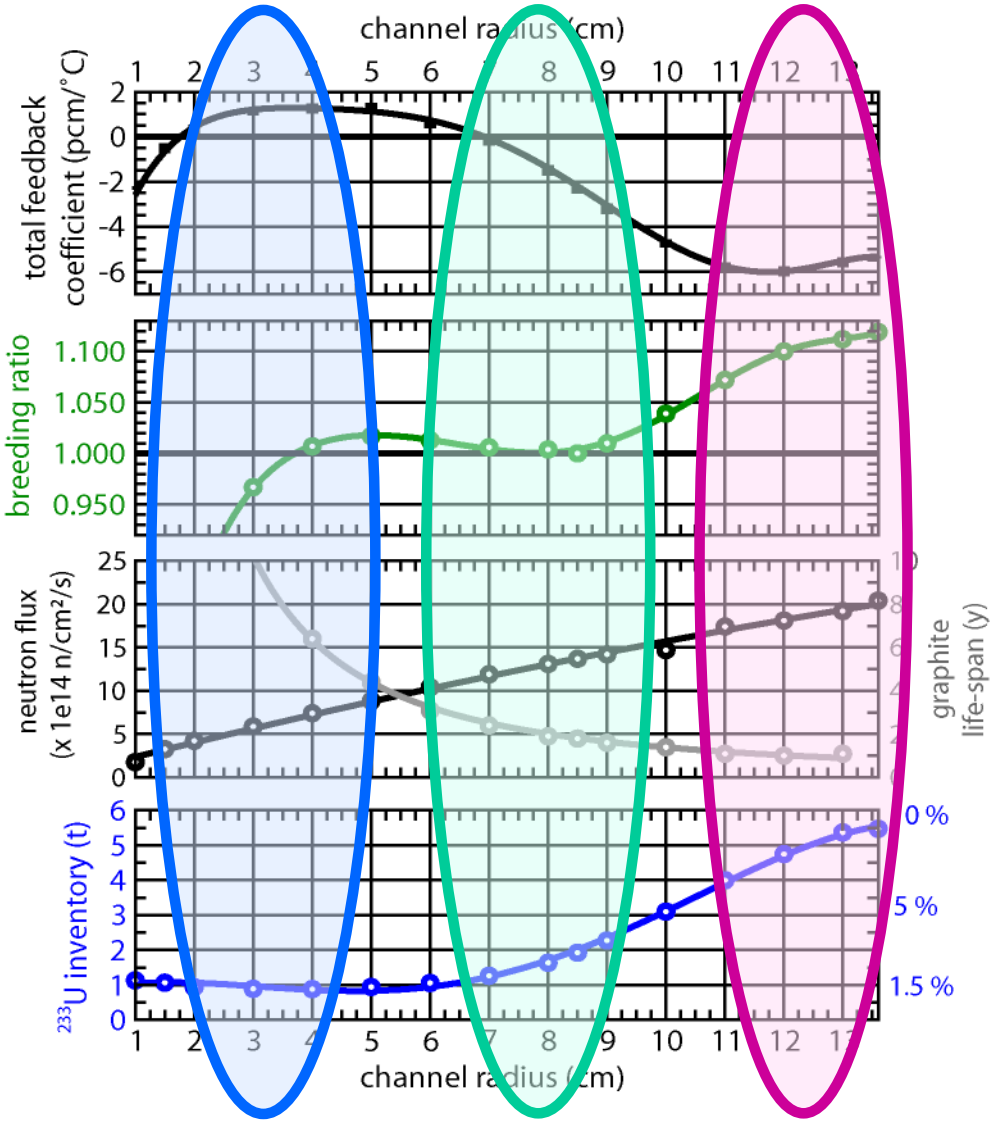
+ $L_{graphite} \propto \frac{1}{\phi}$

no graphite moderator

Historical MSR Studies at CNRS
Influence of the neutron spectrum on the core behavior

Three types of configuration:

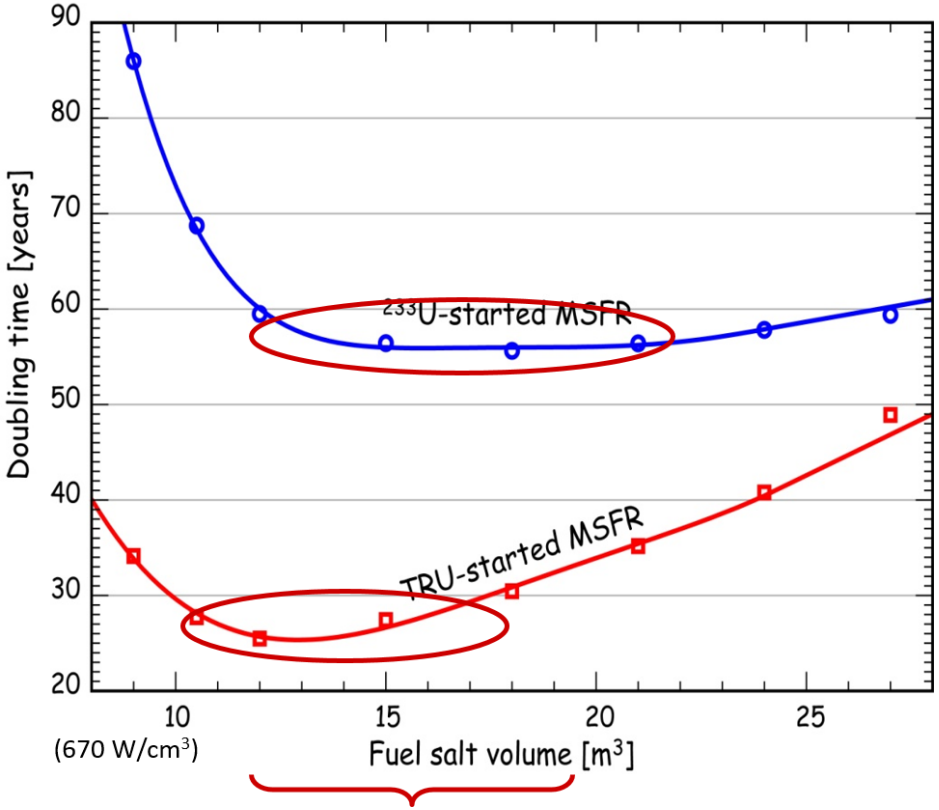
- thermal (r = 3-6 cm)
- epithermal (r = 6-10 cm)
- fast (r > 10 cm)



MSFR: Design and Fissile Inventory Optimization

Breeding and deployment capacities

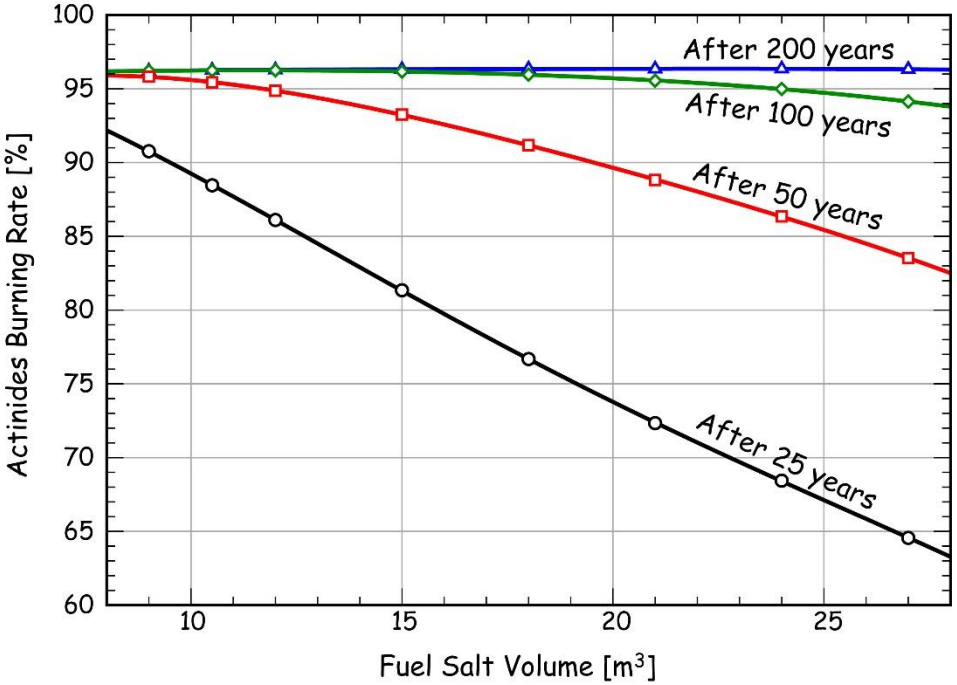
⇒ Deployment capacities: favors the medium volumes (10 m³ to 20 m³)



Initial fissile inventory of 3 to 4.5 tons / GWe reachable

Burning efficiency

⇒ Small fuel salt volumes favoured: > 90% of TRUs burned after only 25 years of operation, and ~ 96% after 50 years of operation



R&D Activities in Reactor Physics, Design and Safety at CNRS

Concepts considered:

- **Large scale MSFR:** called "**reference MSFR**" and studied since around 10 years first at CNRS and now in European and national programs – Objective: defining a large power reactor to identify the challenges of such a concept + define the acceptable configurations
- **Small-size MSFR:** called "**S-MSFR**" and studies foreseen in the coming years – Objective: see how to simplify the MSFR concept + define the steps and a R&D roadmap up to industrialization of such a reactor

R&D Common thematics

- **Neutronics and reactor physics studies** based on multiphysics (code ref = TFM-OpenFOAM coupling code) and evolutive Monte-Carlo calculations + a system code (basic principle simulator) under development
- **Safety analysis and evaluations** using classical safety tools (FFMEA, MLD, LoD) + **Resistance to proliferation** preliminary analysis
- **Experimental** studies of the **salt operation**, the **salt control and processing**

R&D Activities in Reactor Physics, Design and Safety at CNRS

Concepts considered:

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R&D progress

- **Studies at a preliminary stage**
- **Initial fuel composition, preliminary design**

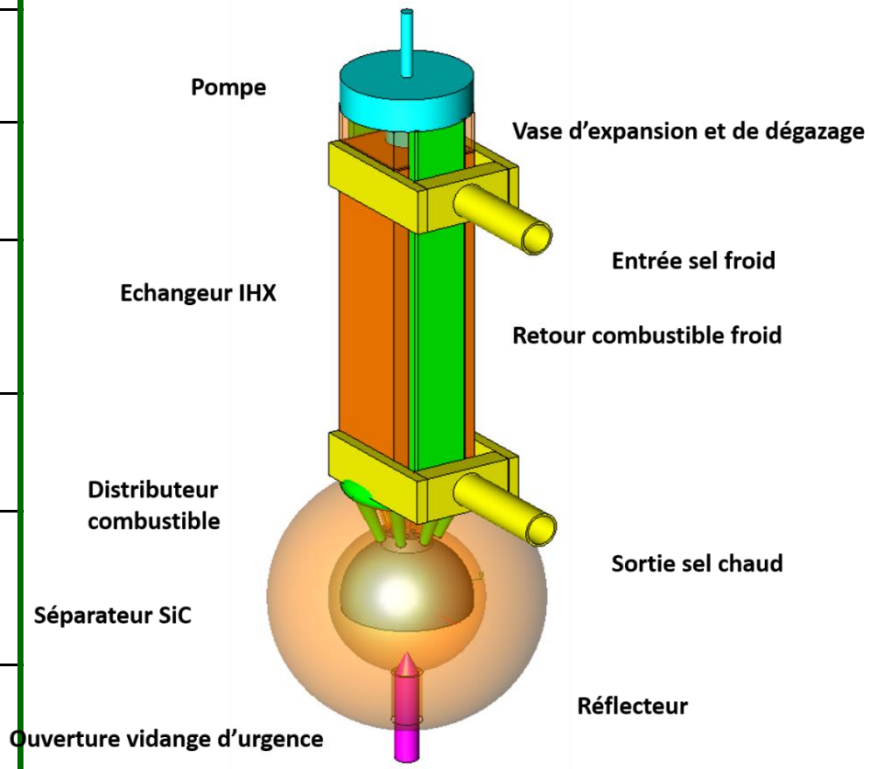
Perspectives and research frame

- **Frame : NEEDS French interdisciplinary program, collaboration with AREVA, other collaborations?**
- **Perspectives: PhD thesis started in October 2017, realistic SMR design + fuel cycle assessment + transient calculations and safety analysis**

Small Modular Reactor – S-MSFR

financé par IDEX Université Grenoble Alpes

Power	**100 MW to 300 MW**
Mean fuel salt temperature	**675 °C**
Fuel temperature increase in the core	**30 °C**
Initial fuel salt composition	**75% LiF-(Heavy Nuclei)F_4 – in Th/^{233}U and U/Pu fuel cycle**
Core size	**Internal diameter ~1.3 m External diameter ~2.3 m**
Fuel salt volume	**2 to 4 m^3 1.1 to 2.3 in core 0.9 to 1.7 in external circuit**
Total fuel cycle in the fuel circuit	**3 - 5 s**



PhD thesis of Johann Martinet

Objectives: ease the constructability while reducing the costs

Idea: operation during around 30 years with the same fuel salt without chemical processing with a simplified concept (heat exchangers simplified, classical nuclear materials, no fertile blanket nor draining tank, …)

Small Modular Reactor – S-MSFR

Preliminary studies of the fissile inventory and breeding capacities in the Th fuel cycle

	No radial blanket and H/D=1	No radial blanket and H/D=1
Power [MW_{th}]	**100**	**200**
Initial ^{233}U load [kg]	654	654
Fuel reprocessing of 1l/day		
Feeding in ^{233}U [kg/an]	**11.38**	**23.38**
Breeding ratio	-29.83%	-30.64%
Total ^{233}U needed [kg]	**1013.87**	**1388.37**

Around 650kg of ^{233}U to start

Under-breeder reactor

Fuel reprocessing of 4l/day		
Feeding in ^{233}U [kg/an]	**11.20**	**22.58**
Breeding ratio	-29.37%	-29.59%
Total ^{233}U needed [kg]	**1001.86**	**1353.13**

Low impact of the chemical reprocessing rate (not mandatory for the demonstrator / SMR)

Small Modular Reactor – S-MSFR

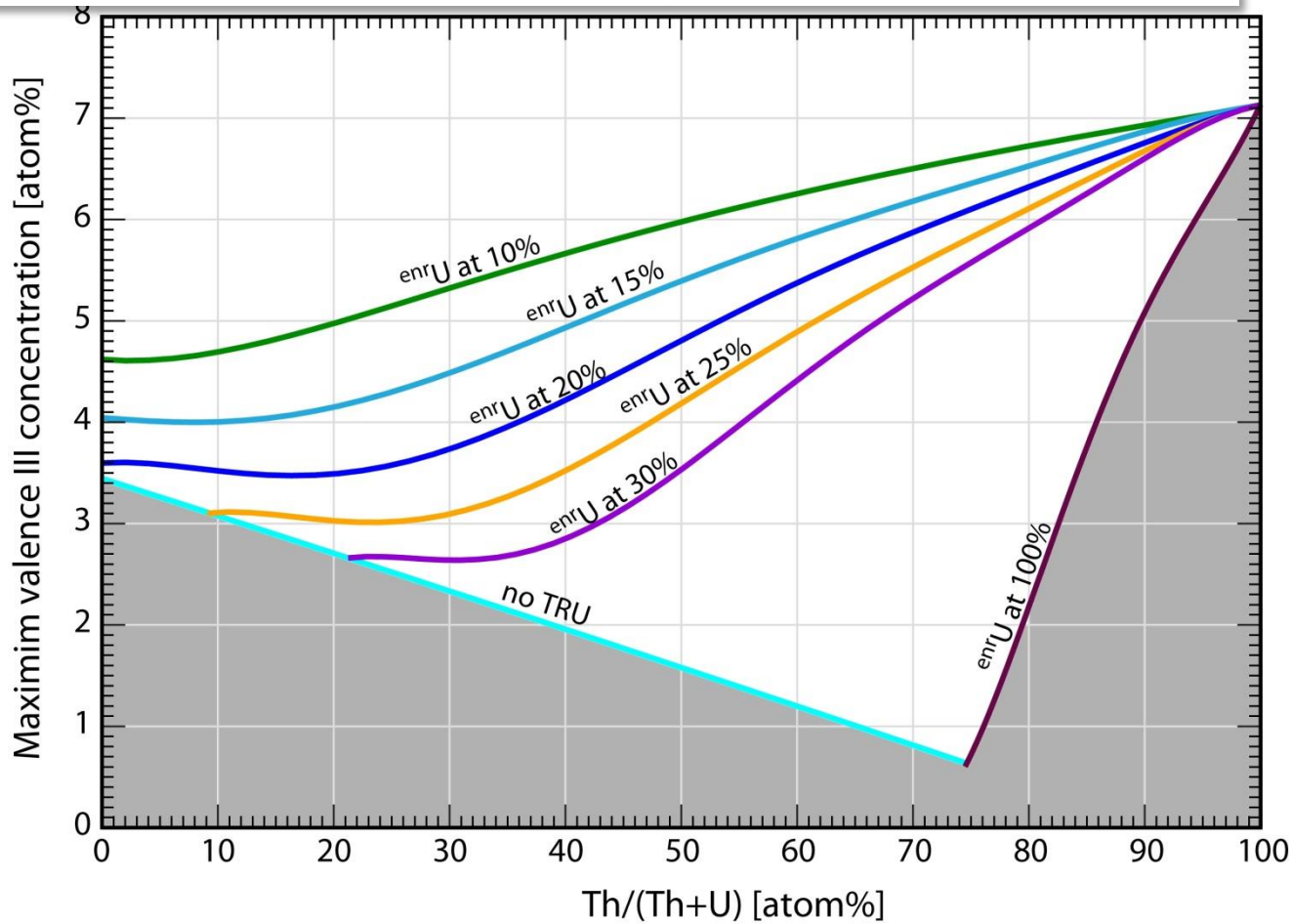
Preliminary studies of the fissile inventory and breeding capacities in the Th fuel cycle

	No radial blanket and H/D=1	No radial blanket and H/D=1	Radial blanket and H/D=1	Radial blanket and H/D=1
Power [MW_{th}]	**100**	**200**	**100**	**200**
Initial ^{233}U load [kg]	654	654	667	667
Fuel reprocessing of 1l/day				
Feeding in ^{233}U [kg/an]	11.38	23.38	1.72	4.70
Breeding ratio	**-29.83%**	**-30.64%**	**-4.52%**	**-6.16%**
Total ^{233}U needed [kg]	1013.87	1388.37	738.83	835.16
Breeding ratio (radial + axial fertile blankets)			***1.81%***	***-0.04%***
Fuel reprocessing of 4l/day				
Feeding in ^{233}U [kg/an]	11.20	22.58	1. 48	3.58
Breeding ratio	**-29.37%**	**-29.59%**	**-3.88%**	**-4.69%**
Total ^{233}U needed [kg]	1001.86	1353.13	722.50	794.21
Breeding ratio (radial + axial fertile blankets)			***2.49%***	***1.54%***

Addition of axial + radial fertile blankets ⇨ small modular breeder MSFR

Small Modular Reactor – S-MSFR

Preliminary studies of the fissile inventory and breeding capacities in the Th fuel cycle



Small Modular Reactor – S-MSFR

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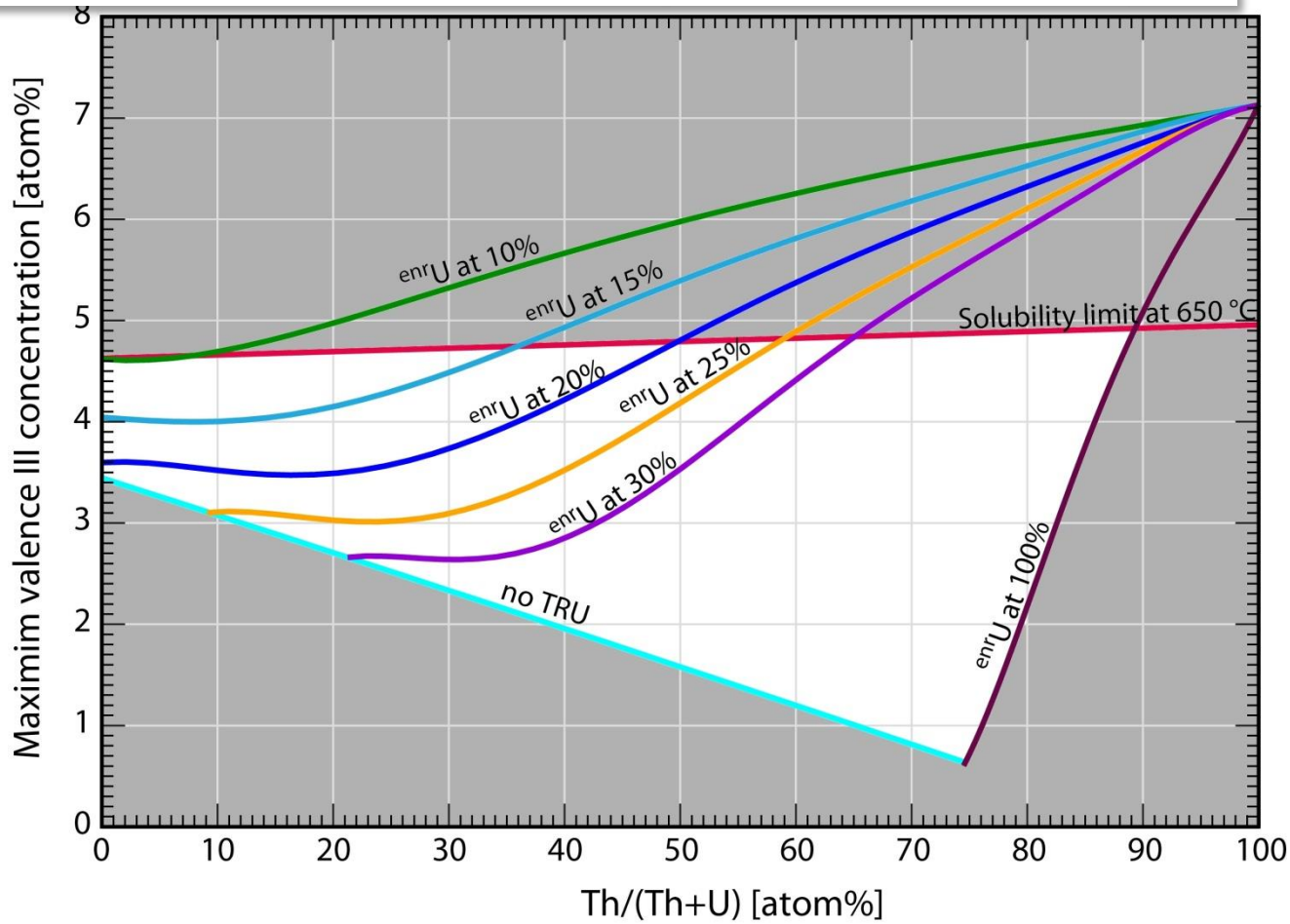
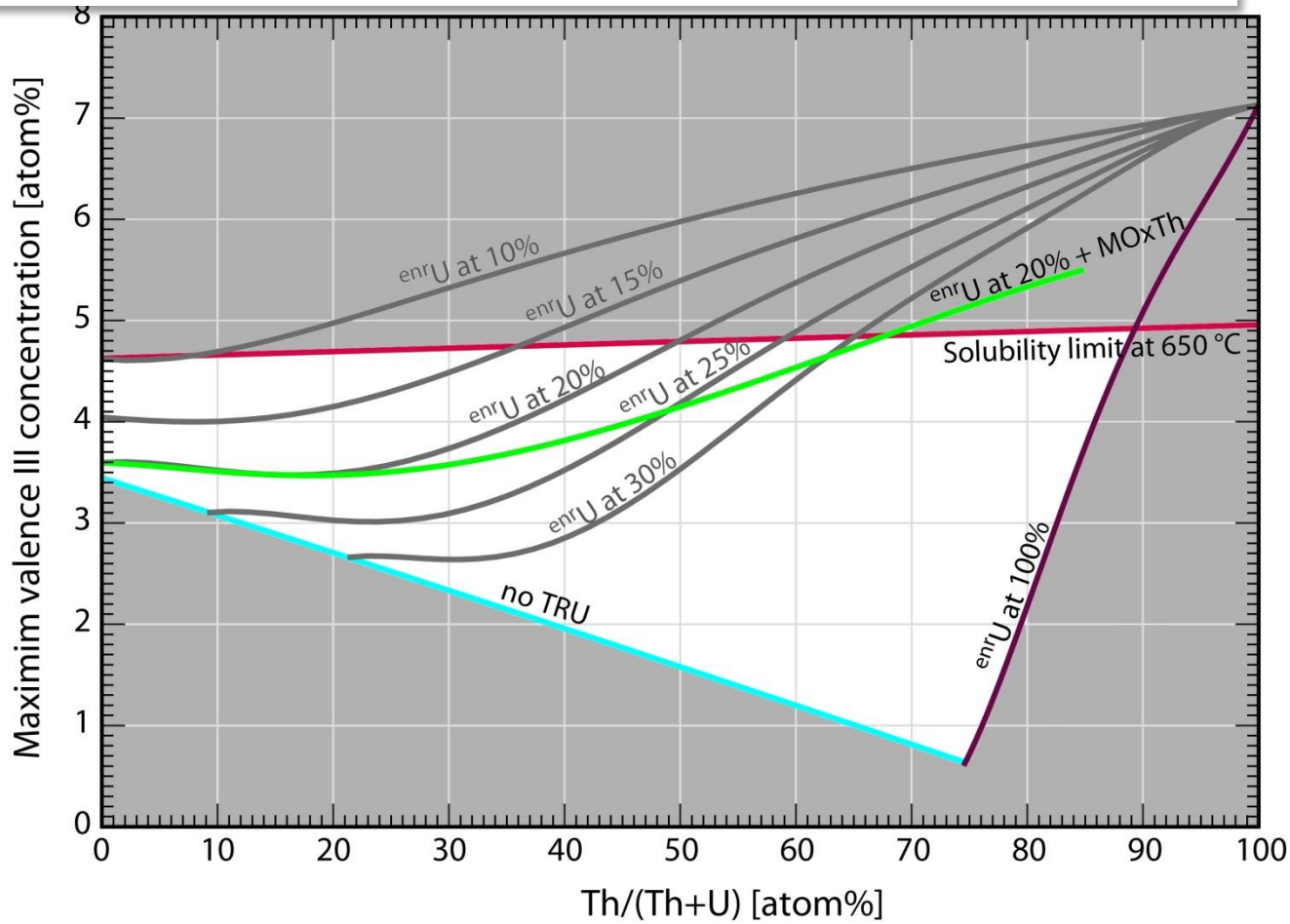


✓ enrichedU mixed with transuranic elements possible with U enrichment of 15% - 20%

Small Modular Reactor – S-MSFR

Preliminary studies of the fissile inventory and breeding capacities in the Th fuel cycle

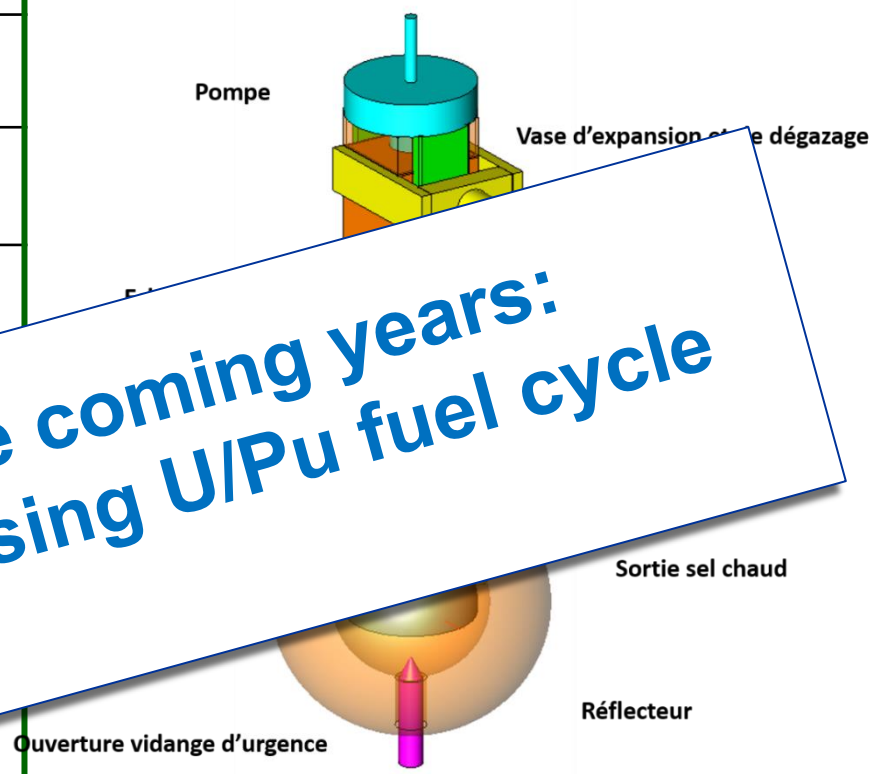


- ✓ ^{enriched}U mixed with transuranic elements possible with U enrichment of 15% - 20%
- ✓ Uranium enriched at 20% mixed with irradiated MOx-Th with a ratio of Th/(Th+U) = 20 to 65%

Small Modular Reactor – S-MSFR

financé par IDEX Université Grenoble Alpes

Power	**100 MW to 300 MW**
Mean fuel salt temperature	**675 °C**
Fuel temperature increase in the core	**30 °C**
Initial fuel salt composition	**75% LiF-(Heavy Nuclei)F_4 – in Th/^{233}U and U/Pu fuel cy**
Core size	**Interm**
Fu	
Total circui	



R&D studies in the coming years:
Study of a S-MSFR using U/Pu fuel cycle

PhD thesis of Johann Martinet

<u>Objectives:</u> ease the constructability while reducing the costs

<u>Idea:</u> operation during around 30 years with the same fuel salt without chemical processing with a simplified concept (heat exchangers simplified, classical nuclear materials, residual heat, no fertile blanket nor draining tank, …)

R&D Activities in Reactor Physics, Design and Safety at CNRS

Concepts considered:

- **Large scale MSFR: called "reference MSFR" and studied since around 10 years first at CNRS and now in European and national programs – Objective: defining a large power reactor to identify the challenges of such a concept + define the acceptable configurations**
- **Small-size MSFR: called "S-MSFR" and studies foreseen in the coming years – Objective: see how to simplify the MSFR concept + define the steps and a R&D roadmap up to industrialization of such a reactor**

R&D progress

- **Advanced studies (reactor physics, design, safety, chemistry, components…) including experimental facilities (2 loops under operation)**
- **See EVOL and SAMOFAR results**

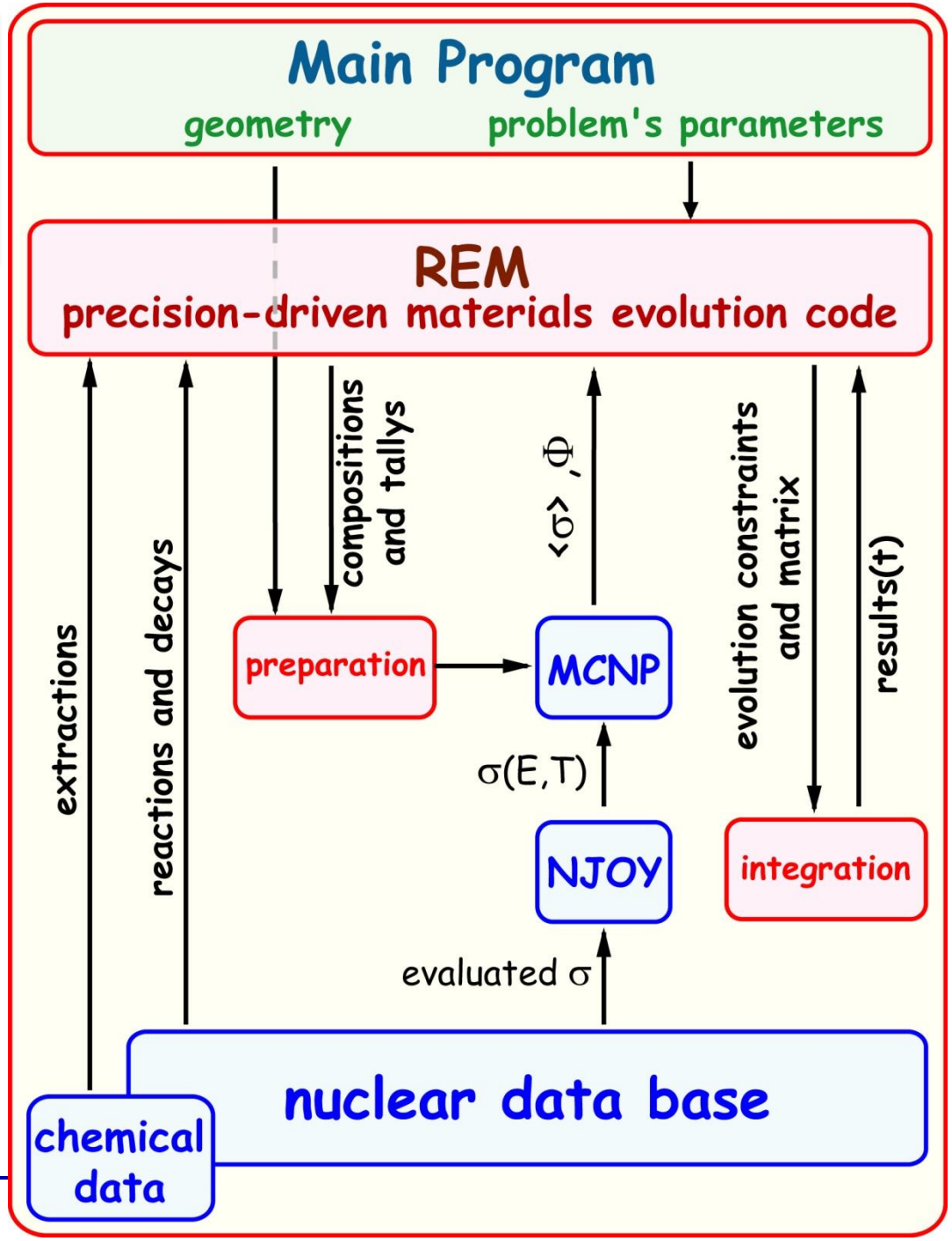
Perspectives and research frame

- **Frame: SAMOFAR European Project and national projects (France, Switzerland)**
- **Perspectives: new European project (advanced simulations for safety), complete the safety study including all systems/components, define the instrumentation, control and inspection, demonstration steps, definition of the demonstrator**

Tools for the Simulation of Reactor Evolution:

Evolving Monte-Carlo calculations

Coupling of the in-house code REM for materials evolution with the probabilistic code MCNP for neutronic calculations



Tools for the Simulation of Reactor Evolution:

Evolving Monte-Carlo calculations

PhD thesis of Xavier DOLIGEZ

Bateman Equation for nucleus i (in the whole system):

$$\frac{\partial N_i^{\,B}}{\partial t} = \sum_{j \neq i} \left\{ \left(\lambda_{j \to i} + X_j < \sigma_{j \to i} \phi > \right) N_j^{\,B} \right\} + \sum_{C \neq B} \lambda_{Chem}^{C \to B} N_i^{\,C} - \lambda_i N_i^{\,B} - < \sigma_i \phi > N_i^{\,B} - \sum_{C \neq B} \lambda_{Chem}^{B \to C} N_i^{\,B}$$

production by radioactive decay

production by nuclear reaction

with

'B' = location of nucleus i in the sub-system B

'B → C' = transfer from sub-system B to sub-system C

disappearance by radioactive decay

disappearance by nuclear reaction

⇩

Calculation of the evolution of matter in each part of the system

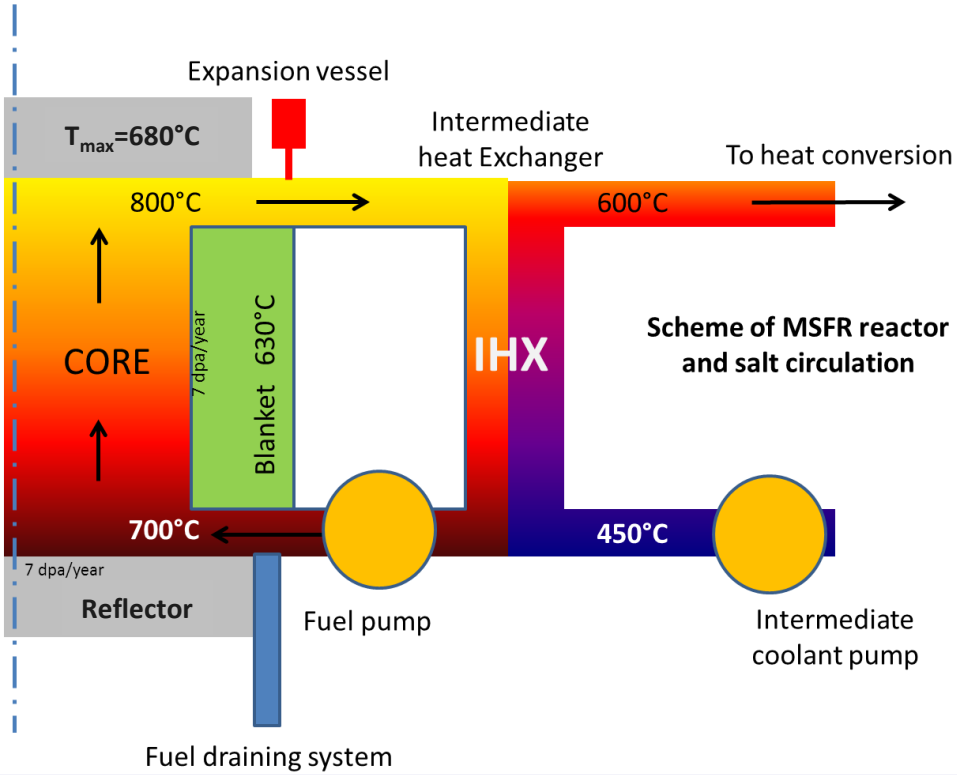
⇒ Determination of isotopes concentrations, gamma or neutron flux + the residual heat (fundamental data for radioprotection) everywhere (core, reprocessing unit, bubbling system, draining tanks…) – Basis of the design and of the safety and radioprotection assessment of the whole system

Concept of MSFR: choice of the structural materials (Ni-based alloys)

Example of material proposed in the EVOL project (with Tungsten):

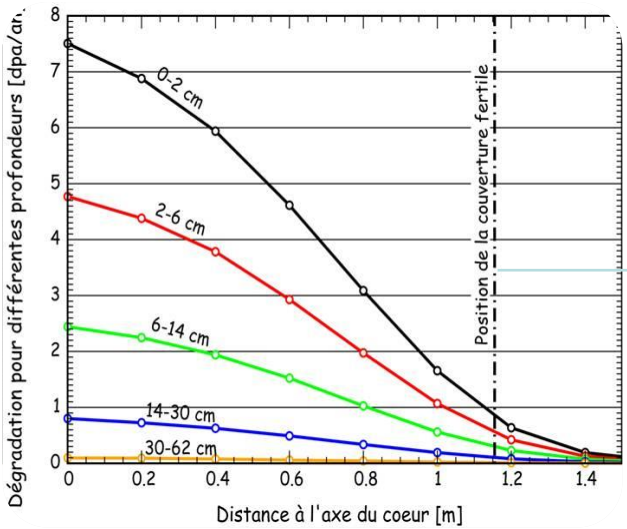
Ni	W	Cr	Mo	Fe	Ti	C	Mn	Si	Al	B	P	S
79.432	9.976	8.014	0.736	0.632	0.295	0.294	0.257	0.252	0.052	0.033	0.023	0.004

Neutronic irradiation damages to the structural materials (modify their physicochemical properties) = displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium, activation – At high temperatures

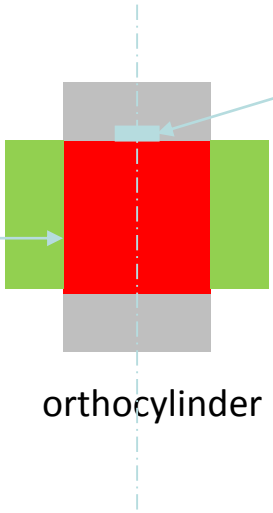


Concept of MSFR: irradiation damages

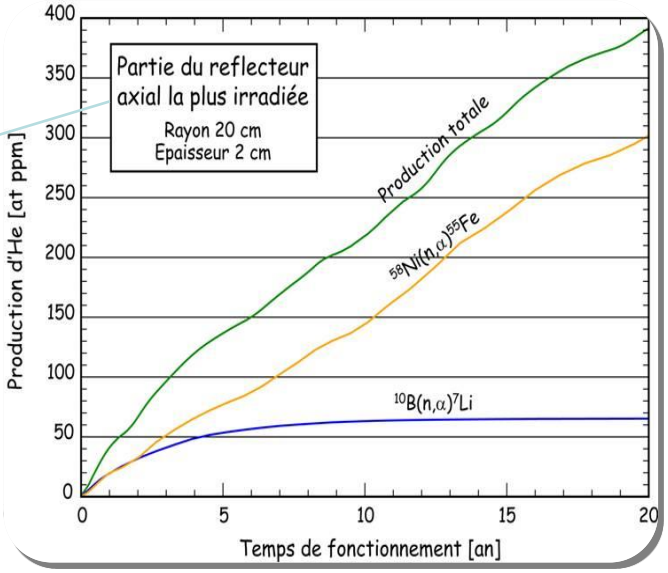
Displacement per Atom (dpa)



dpa in the structural materials (depth & distance from core axis)



orthocylinder



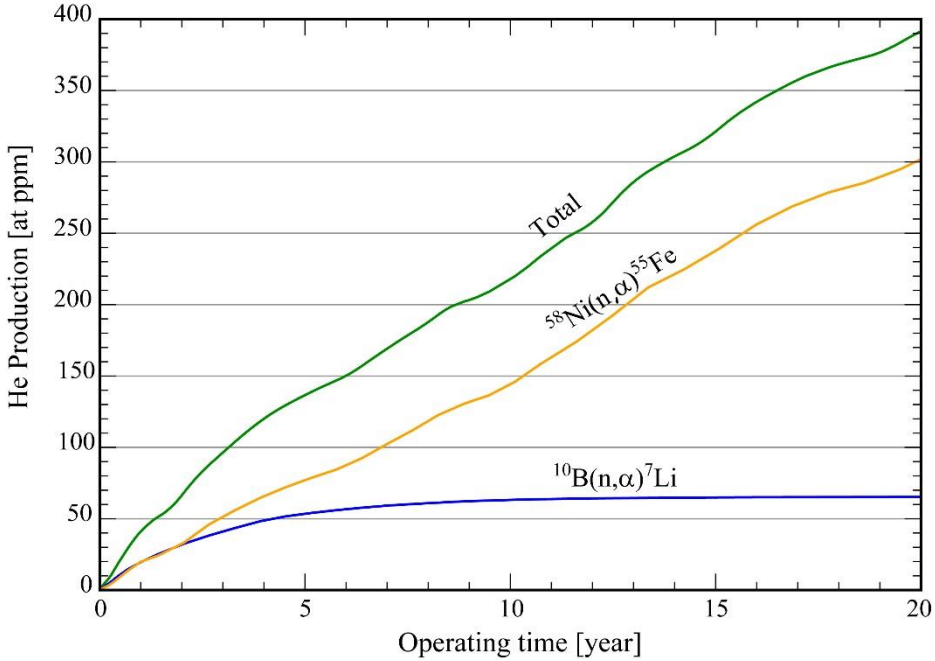
He accumulation (ppm) on the most irradiated part of the reflectors with Ni-based alloys (no He diffusion)

Radiation damages concentrated on the core axis and concern a limited material depth

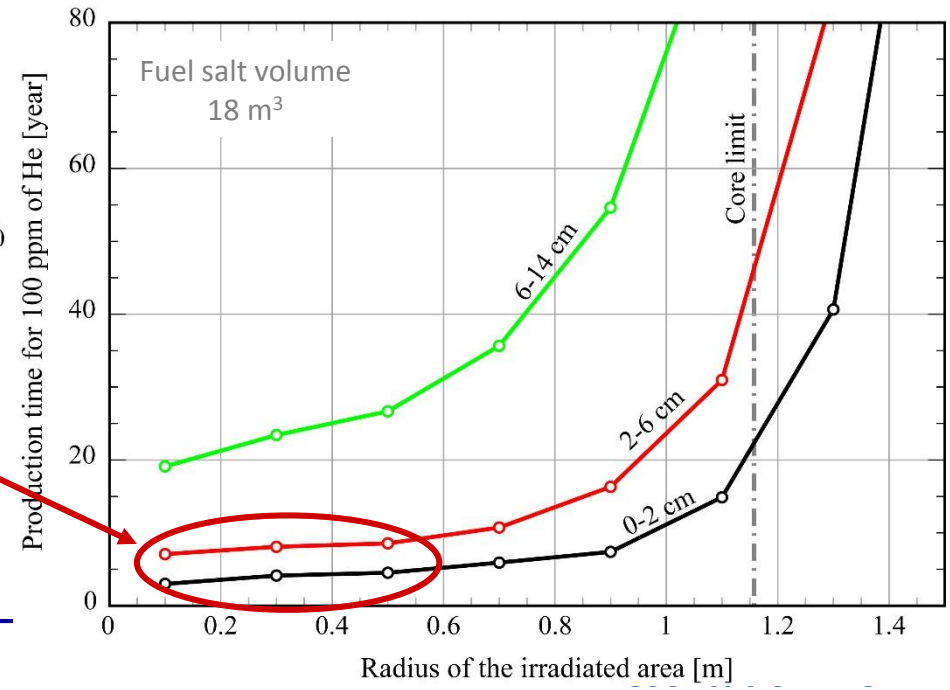
Concept of MSFR: irradiation damages

Helium production in the structural materials

Ni	W	Cr	Mo	Fe	Ti	C	Mn	Si	Al	B	P	S
79.432	9.976	8.014	0.736	0.632	0.295	0.294	0.257	0.252	0.052	0.033	0.023	0.004



Main contribution to Helium production in the most irradiated area (radius 20 cm /thickness 2 cm) for a fuel salt volume of 18 m^3 due to ^{58}Ni



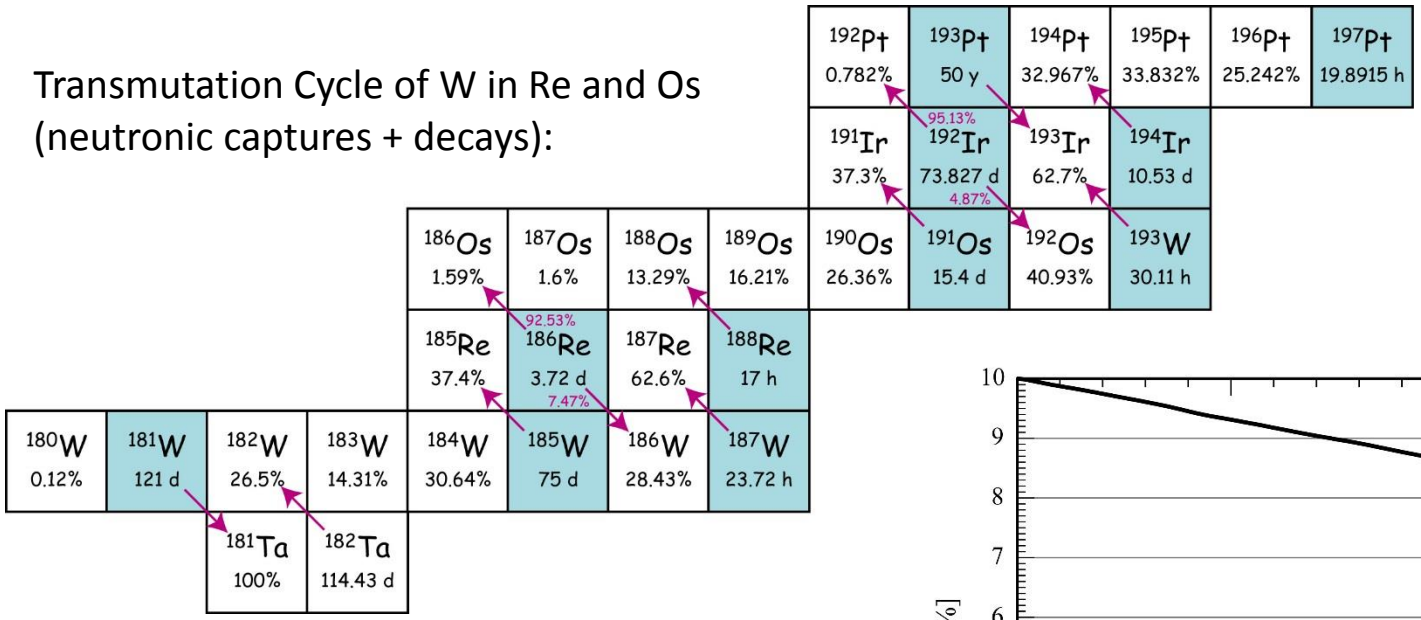
⇒ Regular replacements of these area to be planned (first 10cm only) or addition of another material to protect the surface of these reflectors?

Concept of MSFR: irradiation damages

Transmutation of the Tungsten contained in the alloy into Rhenium and Osmium

Ni	W	Cr	Mo	Fe	Ti	C	Mn	Si	Al	B	P	S
79.432	9.976	8.014	0.736	0.632	0.295	0.294	0.257	0.252	0.052	0.033	0.023	0.004

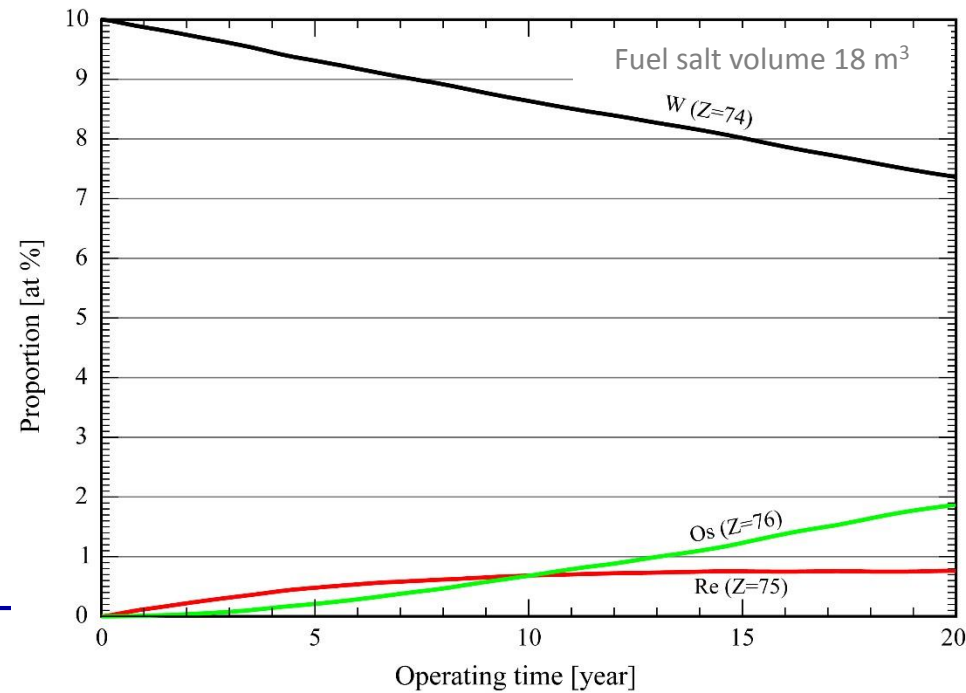
Transmutation Cycle of W in Re and Os (neutronic captures + decays):



W, Re and Os contents of the most irradiated area for a fuel salt volume of 18m^3:

- Value of the acceptable limit?
- Impact on the structural materials resistance?

→ Alloy without Tungsten



Concept of MSFR: irradiation damages

Ni	W	Cr	Mo	Fe	Ti	C	Mn	Si	Al	B	P	S
79.432	9.976	8.014	0.736	0.632	0.295	0.294	0.257	0.252	0.052	0.033	0.023	0.004

Neutronic irradiation damages to the structural materials (modify their physicochemical properties) = displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium, activation

Structural elements: layers	Displacements per atom	He production	Tungsten transmutation
0-2.5 cm	6.8 dpa/year	12 ppm / year	0.11 at% /year
2.5-7.5 cm	3.5 dpa/year	6 ppm / year	0.07 at% /year

To be experimentally studied: He production (maximal acceptable amount, diffusion effects?) + Effects on the long-term resistance of structural materials due to W transmutation + Effects of high temperature on structural materials

Conclusions:

- Irradiation damages low + Limits unknown
- Irradiation damages limited to the first 10 cm (replaced 3-4 times or use a thin layer of SiC for example as thermal protection)
- Materials not under large mechanical stress