Introduction to the Physics of the Molten Salt Fast Reactor

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For the 'MSFR Team' of LPSC - M. ALLIBERT, M. BROVCHENKO, V. GHETTA, D. HEUER, A. LAUREAU, E. MERLE-LUCOTTE, P. RUBIOLO

With the support of the IN2P3 institute and the PACEN and NEEDS Programs of CNRS, and of the EVOL Euratom FP7 Project

Liquid fuelled-reactors

Which constraints for a liquid fuel?

- Melting temperature not too high
- High boiling temperature
- Low vapor pressure
- 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

Best candidates = **fluoride salt** (LiF – 99.995% of ^{7}Li)

Molten Salt Reactors

Neutronic properties of F not favorable to the U/Pu fuel cycle

Thorium / ^{233}U Fuel Cycle

Advantages of a Liquid Fuel

- ✓ 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)

- One configuration optimized for the electricity production managing the criticality
- An other configuration allowing 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 physicochem. properties
- No reactivity reserve (fertile/fissile matter adjusted during reactor operation)

Liquid fuelled-reactors: MSR

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Molten Salt Reactors

Neutronic properties of F not favorable to the U/Pu fuel cycle

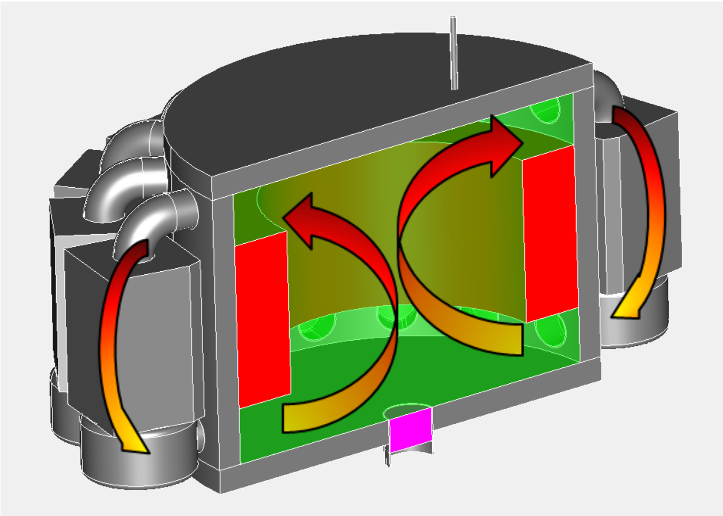
Thorium /^{233}U Fuel Cycle

What is a MSFR ?

Molten Salt Reactor (molten salt = liquid fuel also used as coolant)

Based on the Thorium fuel cycle

With no solid (i.e. moderator) matter in the core $\Rightarrow$ Fast neutron spectrum



From MSR to Molten Salt Fast Reactor (MSFR)

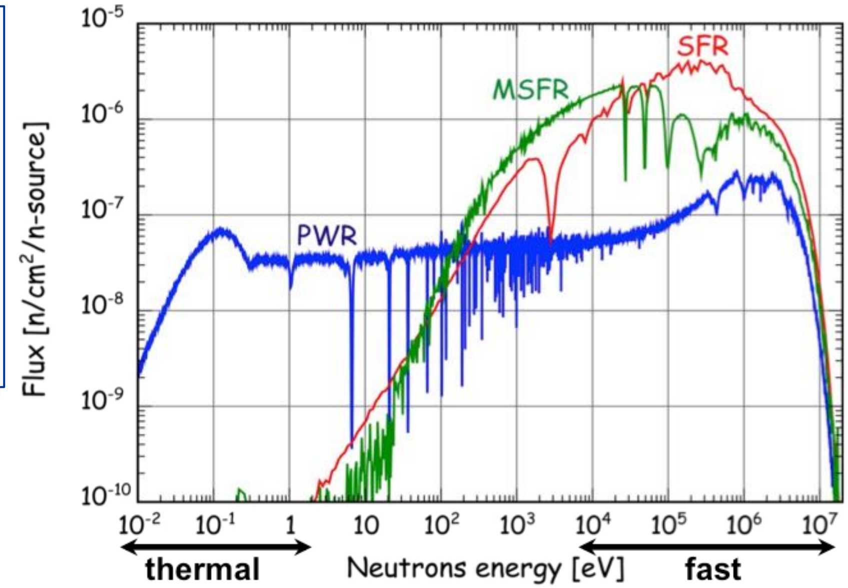
Neutronic Optimization of MSR (Gen4 criteria) :

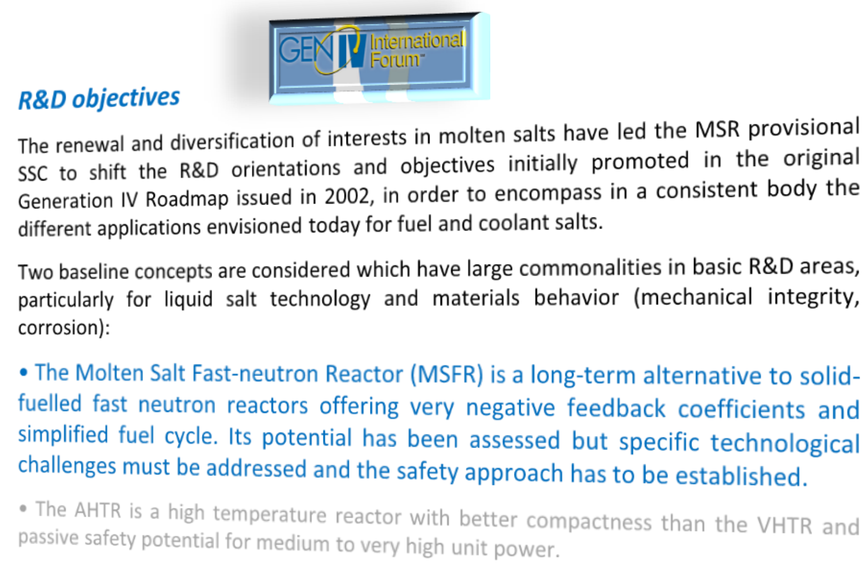
- **Safety: negative feedback coefficients**
- **Sustainability: reduce irradiation damages in the core**
- **Deployment: good breeding of the fuel + reduced initial fissile inventory**

PhD Thesis of L. Mathieu

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.

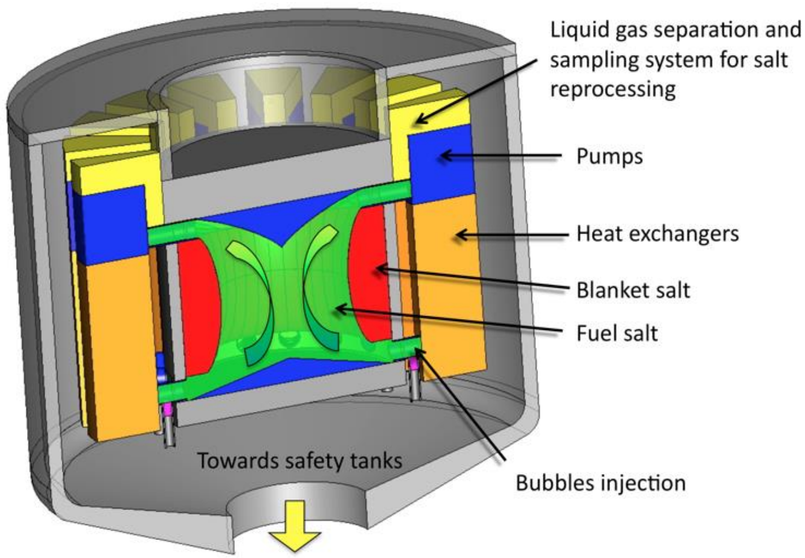
Two baseline concepts are considered which have large commonalities in basic R&D areas, particularly for liquid salt technology and materials behavior (mechanical integrity, corrosion):

- 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.

The concept of Molten Salt Fast Reactor (MSFR)

Thermal power	**3000 MWth**
Mean fuel salt temperature	**750 °C**
Fuel salt temperature rise in the core	**100 °C**
Fuel molten salt - Initial composition	**77.5% LiF and 22.5% [ThF_4+ (Fissile Matter)F_4] with Fissile Matter = ^{233}U / $^{enriched}U$ / Pu+MA**
Fuel salt melting point	**565 °C**
Fuel salt density	**4.1 g/cm^3**
Fuel salt dilation coefficient	**8.82 10^{-4} / °C**
Fertile blanket salt - Initial composition	**LiF-ThF_4 (77.5%-22.5%)**
Breeding ratio (steady-state)	**1.1**
Total feedback coefficient	**-5 pcm/K**
Core dimensions	**Diameter: 2.26 m Height: 2.26 m**
Fuel salt volume	**18 m^3 (½ in the core + ½ in the external circuits)**
Blanket salt volume	**7.3 m^3**
Total fuel salt cycle	**3.9 s**

Design of the 'reference' MSFR



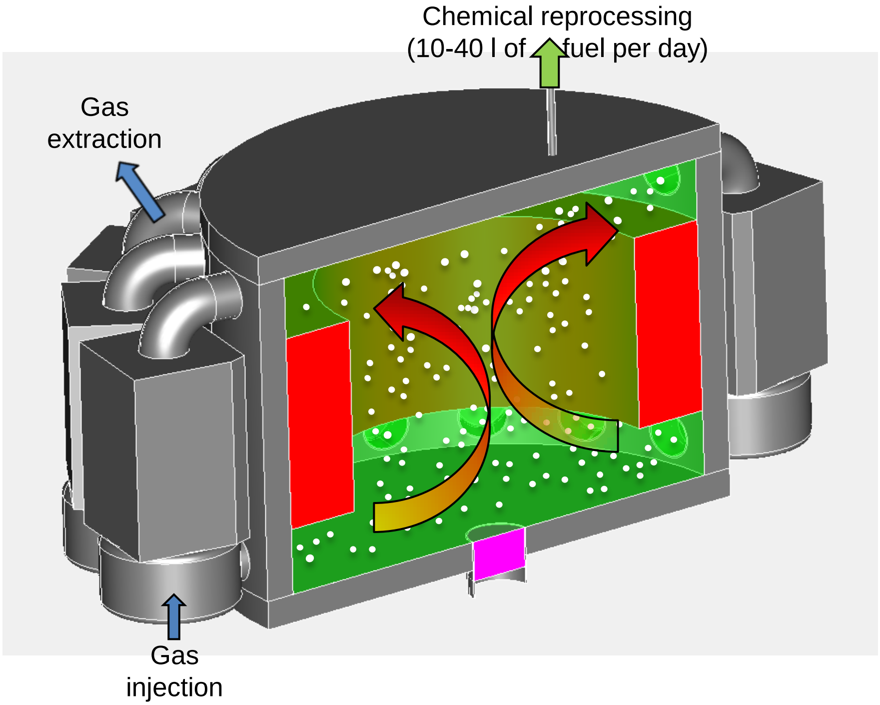
Optimization Criteria:

Initial fissile matter (^{233}U, Pu, enriched U), salt composition, fissile inventory, reprocessing, waste management, deployment capacities, heat exchanges, structural materials, design…

MSFR: R&D collaborations

4th Generation reactors => Breeder reactors

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



MSFR: R&D collaborations

4th Generation reactors => Breeder reactors

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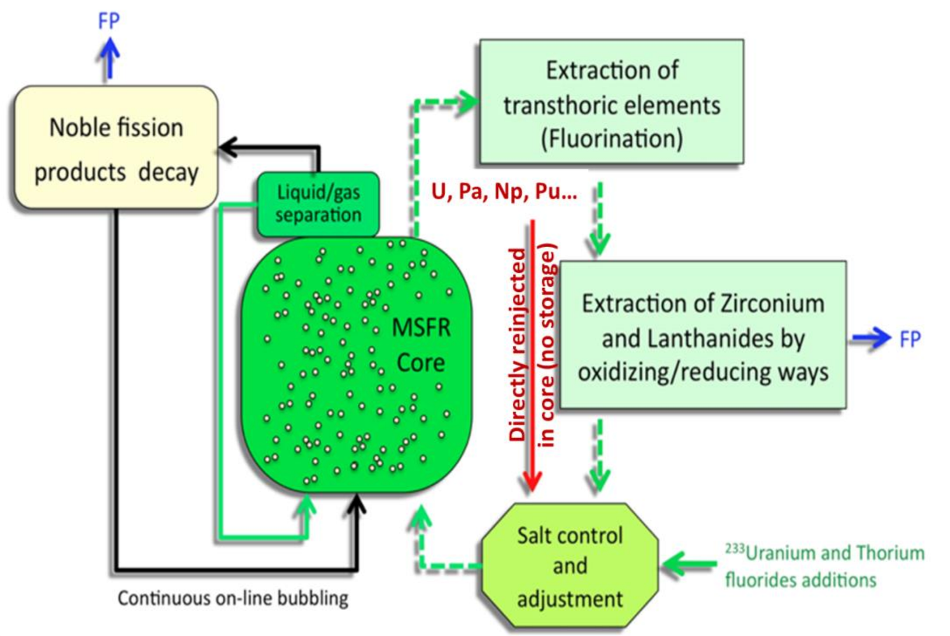
Conclusions of the studies: **very low impact of the reprocessings (chemical and bubbling) on the neutronic behavior of the MSFR thanks to the fast neutron spectrum** = neutronic and chemical (physico-chemical properties of the salt) studies driven in parallel

PhD Thesis of X. Doligez

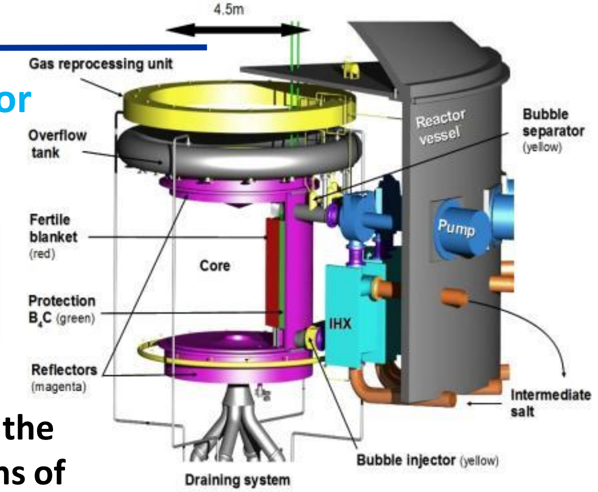


Studies requiring multidisciplinary expertise (reactor physics, chemistry, safety, materials, design…)

Collaboration at different levels:

- **World: Generation 4 International Forum**
- **Europe: Collaborative Project EVOL Euratom/Rosatom + SNETP SRIA Annex**
- **National: IN2P3/CNRS and interdisciplinary programs PACEN and NEEDS (CNRS, CEA, IRSN, AREVA, EdF), structuring project 'CLEF' of Grenoble INP**

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

- **Recommendations for the design of the core and fuel heat exchangers**
- **Definition of a safety approach dedicated to liquid-fuel reactors - Transposition of the defence in depth principle - Development of dedicated tools for transient simulations of molten salt reactors**
- **Determination of the salt composition - Determination of Pu solubility in LiF-ThF4 - Control of salt potential by introducing Th metal**
- **Evaluation of the reprocessing efficiency (based on experimental data) – FFFER project**
- **Recommendations for the composition of structural materials around the core**

EVOL

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

12 European Partners: France (CNRS: Coordinateur, Grenoble INP , INOPRO, Aubert&Duval), Pays-Bas (Université Techno. de Delft), Allemagne (ITU, KIT-G, HZDR), Italie (Ecole polytechnique de Turin), Angleterre (Oxford), Hongrie (Univ Techno de Budapest)
+ 2 observers since 2012 : Politecnico di Milano et Paul Scherrer Institute

+ Coupled to the MARS (Minor Actinides Recycling in Molten Salt) project of ROSATOM (2011-2013)

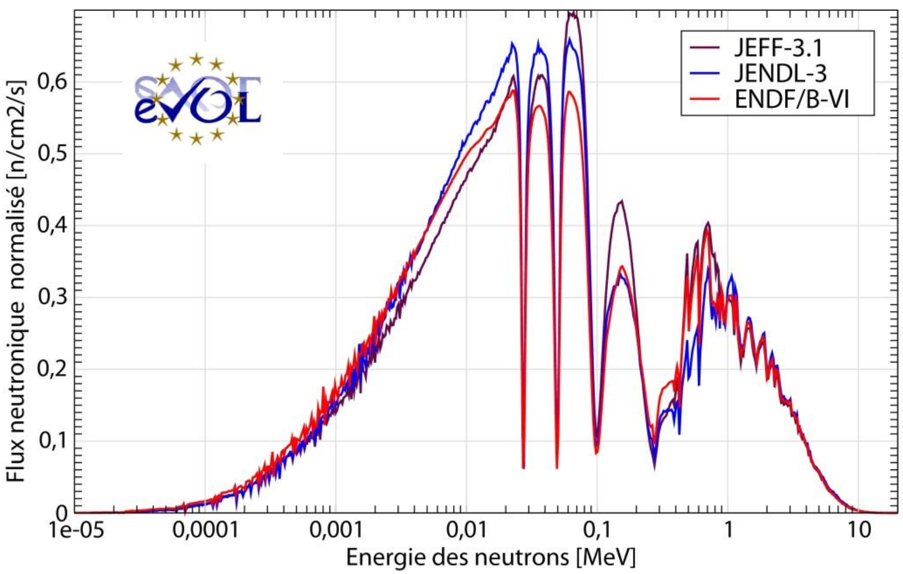
Partners: RIAR (Dimitrovgrad), KI (Moscow), VNIITF (Snezinsk), IHTE (Ekateriburg), VNIKHT (Moscow) et MUCATEX (Moscow)

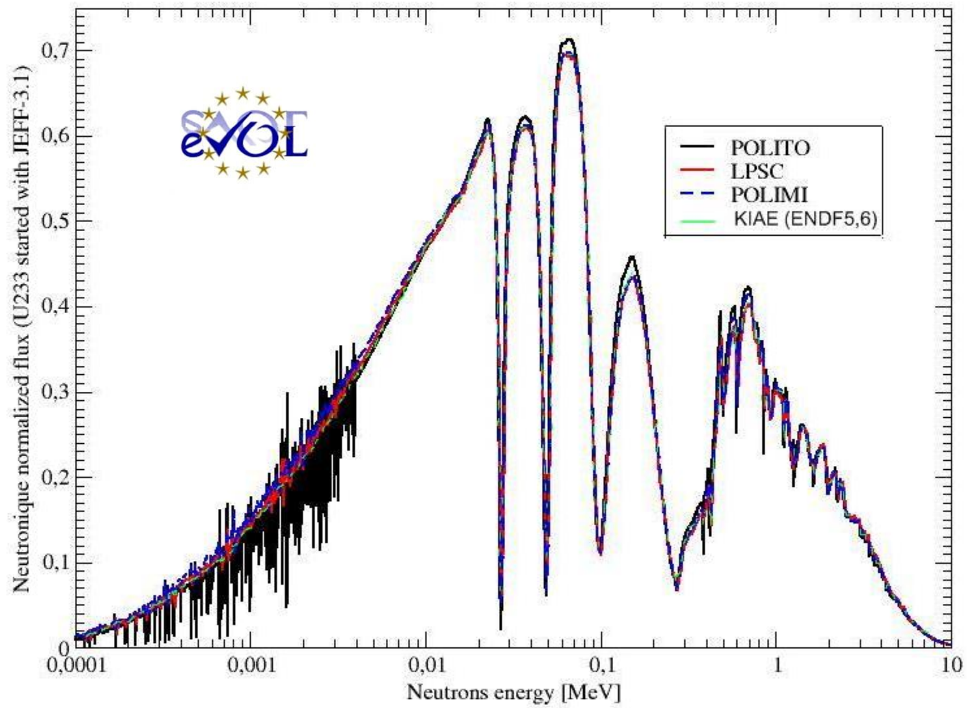
MSFR optimization: neutronic benchmark (EVOL)

LPSC-IN2P3 calculations performed with a Monte-Carlo neutronic tool (MCNP) coupled to a material evolution code (REM)

Initial Fuel Salt Composition – EVOL Benchmark				
^{233}U-started MSFR		TRU-started MSFR		
Th	^{233}U	Th	Actinides	
38 281 kg 19.985 %mol	4 838 kg 2.515 %mol	30 619 kg 16.068 %mol	Pu	11 079 kg 5.628 %mol
			Np	789 kg 0.405 %mol
			Am	677 kg 0.341 %mol
			Cm	116 kg 0.058 %mol

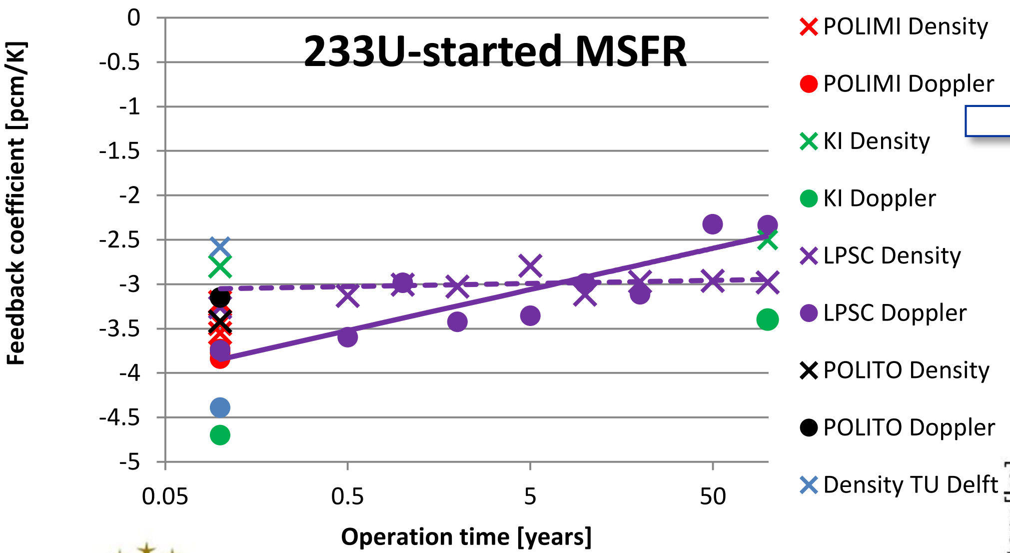
PhD Thesis of M. Brovchenko



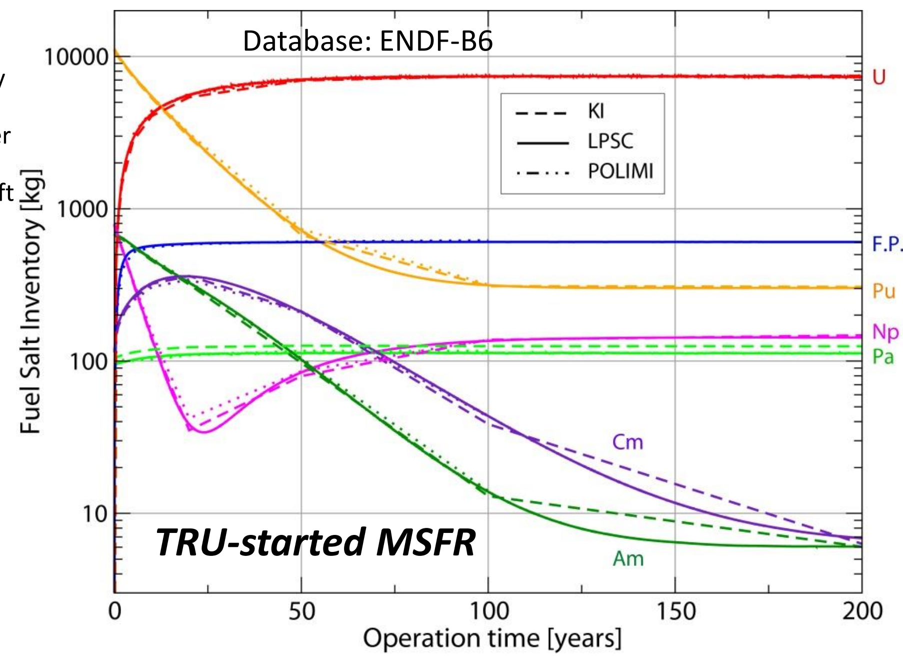


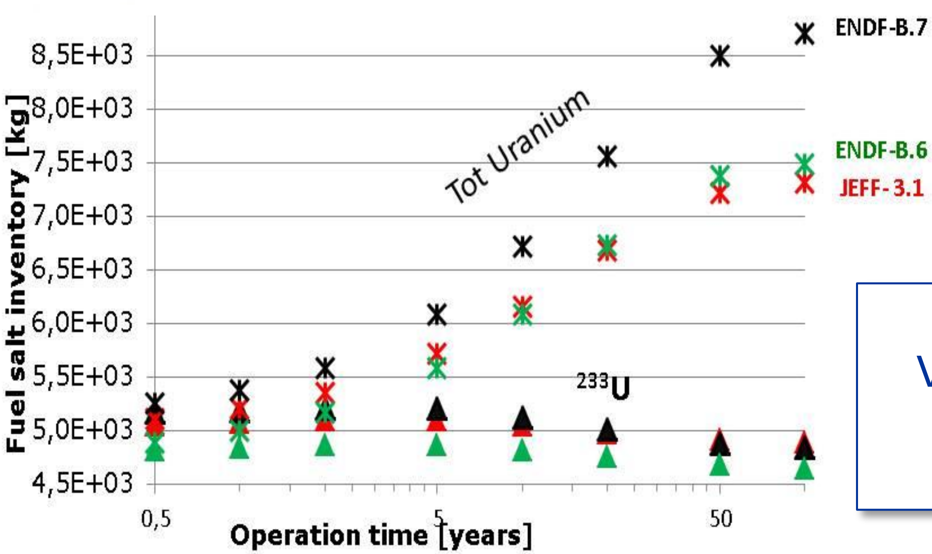
<u>Static calculations (BOL here):</u>
Good agreement between the different simulation tools – High impact of the nuclear database

MSFR optimization: neutronic benchmark (EVOL)



Largely negative feedback coefficients, ∀ the simulation tool or the database used





<u>Evolution calculations:</u>
Very good agreement between the different simulation tools – High impact of the nuclear database

MSFR optimization: initial fissile matter

Which initial fissile load to start a MSFR?

- Start directly ^{233}U produced in Gen3+ or Gen4 (included MSFR) reactors

- Start directly with enriched U: U enrichment < 20% (prolif. Issues)

- Start with the Pu of current LWRs mixed with other TRU elements:
solubility limit of valence-III elements in LiF

- Mix of these solutions: Thorium as fertile matter +

➢ ^{233}U + TRU produced in LWRs

➢ MOx-Th in Gen3+ / other Gen4

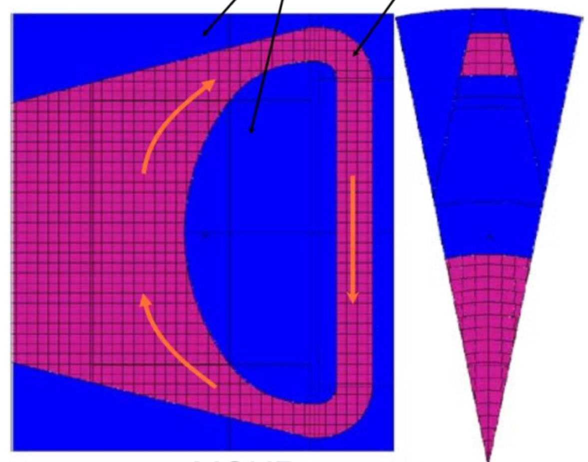
➢ Uranium enriched (e.g. 13%) + TRU currently produced

[kg per GWe]	^{233}U started MSFR	TRU (Pu UOx) started MSFR	Enriched U (13%) + TRU started MSFR	Th Pu-MOx started MSFR
Th 232	25 553	20 396	10 135	18 301
Pa 231				20
U 232				1
U 233	3 260			2 308
U 234				317
U 235			1 735	45
U 236				13
U 238			11 758	
Np 237		531	335	54
Pu 238		229	144	315
Pu 239		3 902	2 464	1 390
Pu 240		1 835	1 159	2 643
Pu 241		917	579	297
Pu 242		577	364	1 389
Am 241		291	184	1 423
Am 243		164	104	354
Cm 244		69	44	54
Cm 245		6	4	

MSFR optimization: thermal-hydraulic studies

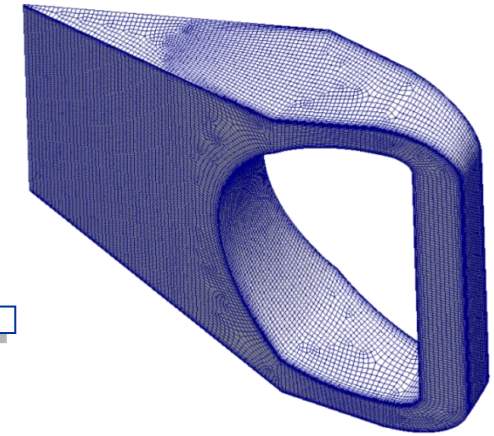
PhD Thesis of A. Laureau

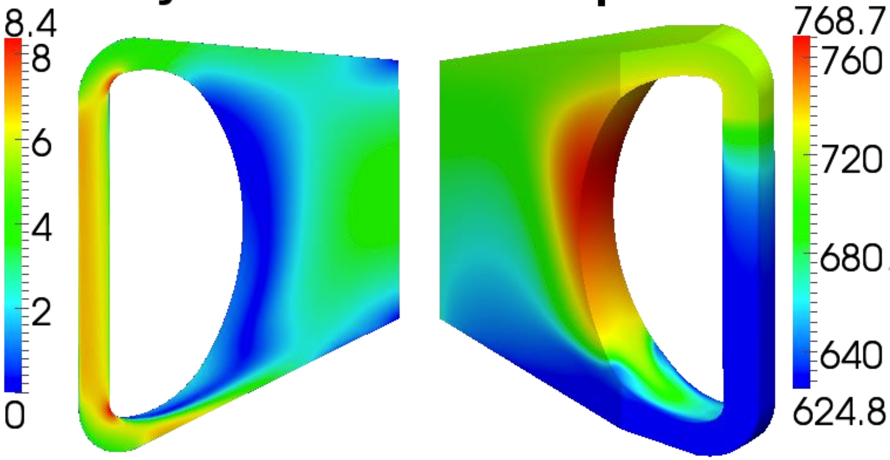
Steady state neutronic / thermal-hydraulic coupling dedicated to liquid fuel reactor

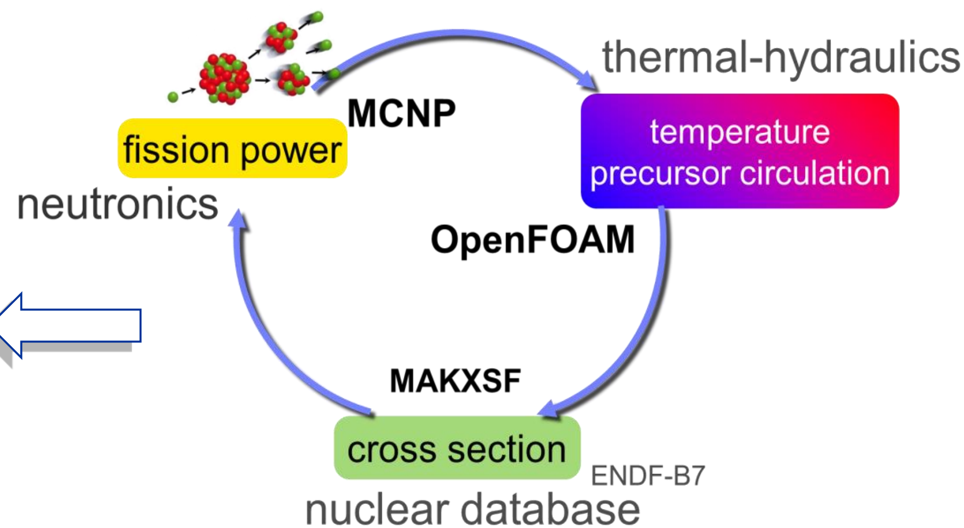


MCNP geometry

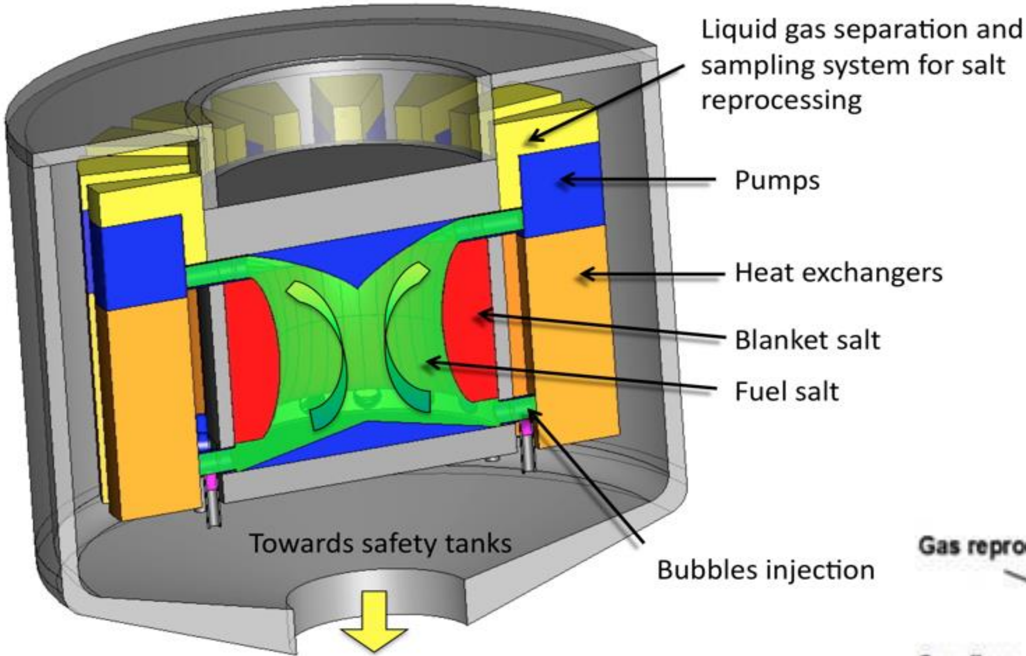
CFD mesh - 1/16 core 300 k cells







Molten Salt Fast Reactor (MSFR): fuel circuit



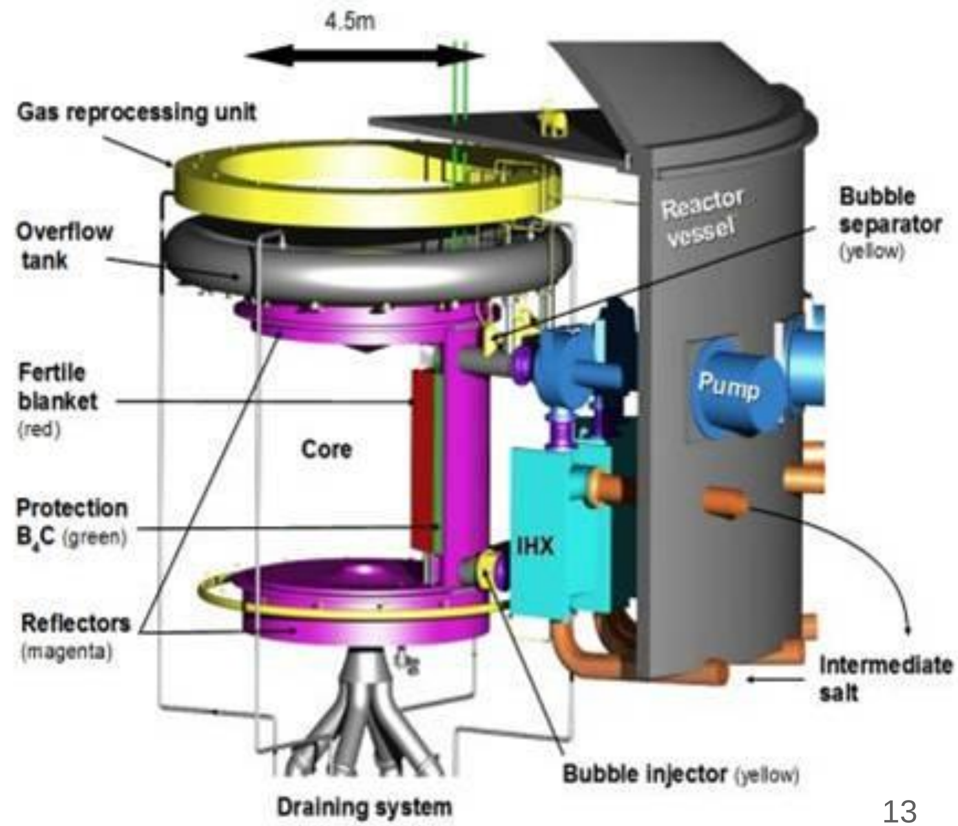
Core (active area):

No inside structure

Outside structure: Upper and lower Reflectors, Fertile Blanket Wall

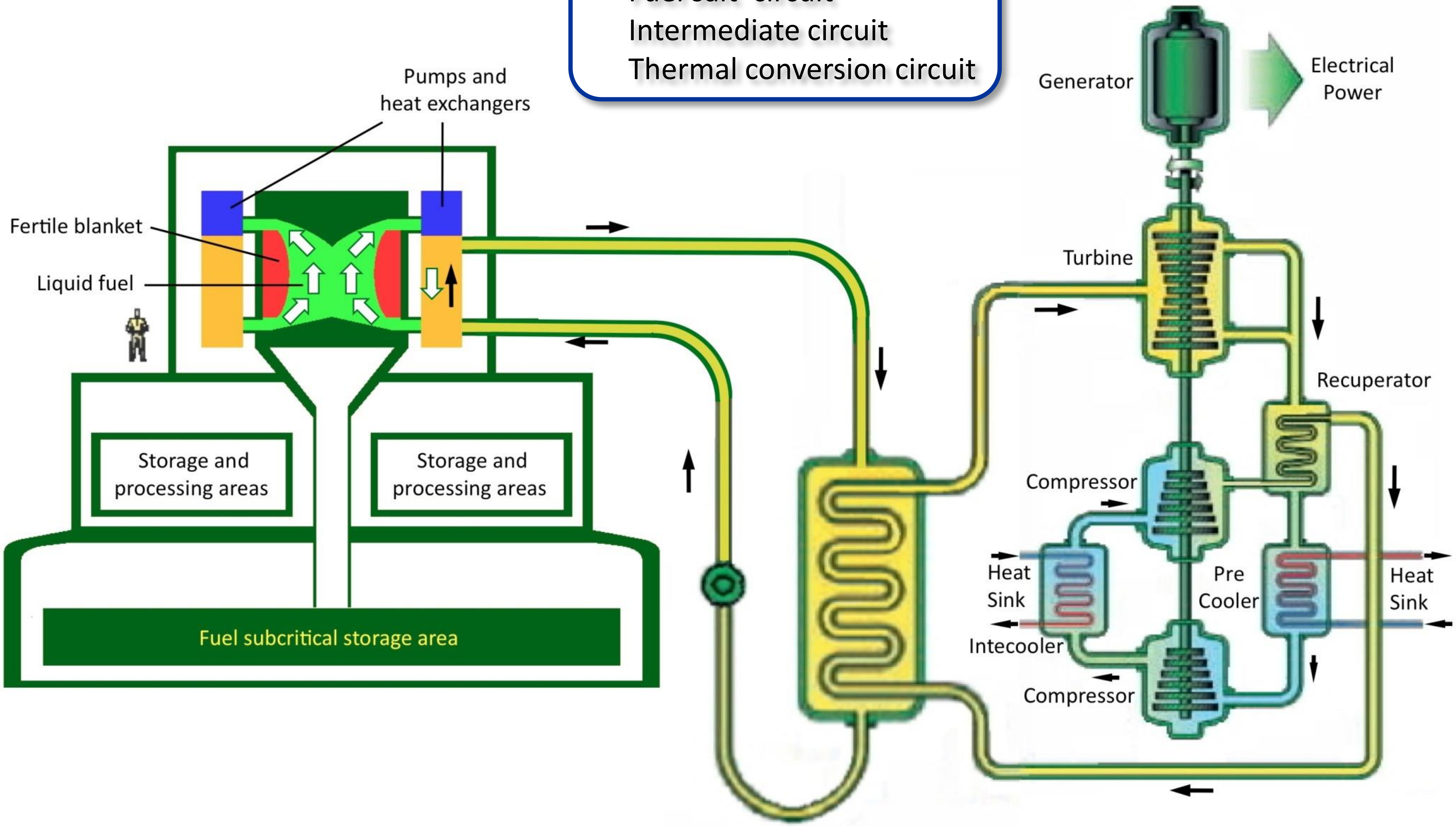
+ 16 external recirculation loops:

- Pipes (cold and hot region)
- Bubble Separator
- Pump
- Heat Exchanger
- Bubble Injection



Molten Salt Fast Reactor (MSFR)

Three circuits:
- Fuel salt circuit
- Intermediate circuit
- Thermal conversion circuit



MSFR and Safety Evaluation

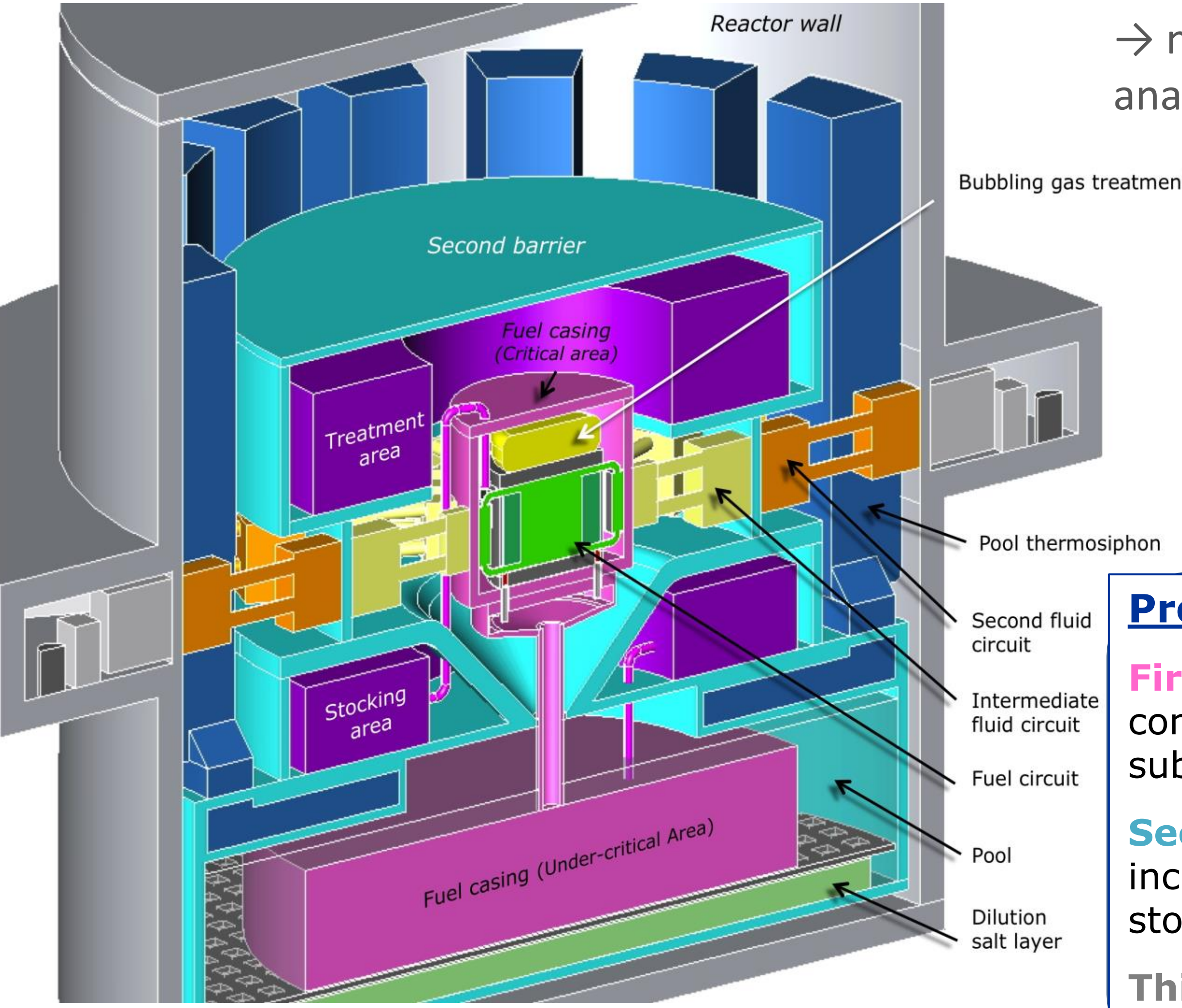
Design aspects impacting the MSFR safety analysis

- 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 core
  - ✓ Fuel reprocessing and loading during reactor operation

- No control rods in the core
  - ✓ Reactivity is controlled by the heat transfer rate in the HX + fuel salt feedback coefficients, continuous fissile loading, and by the geometry of the fuel salt mass
  - ✓ No requirement for controlling the neutron flux shape (no DNB, uniform fuel irradiation, etc.)

- Fuel salt draining
  - ✓ Cold shutdown is obtained by draining 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

MSFR and Safety Evaluation



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

⇩

Proposed Confinement barriers:

First barrier: fuel envelop, composed of two areas: critical and sub-critical areas

Second barrier: reactor vessel, also including the reprocessing and storage units

Third barrier: reactor wall, corresponding to the reactor building

MSFR and Safety Evaluation

Safety analysis: objectives

- **Develop a safety approach dedicated to MSFR**
  - **Based on current safety principles** e.g. defense-in-depth, multiple barriers, the 3 safety functions (reactivity control, fuel cooling, confinement) etc. but adapted to the MSFR.
  - Integrate both **deterministic and probabilistic** approaches
  - Specific approach dedicated to **severe accidents**:
    - Fuel liquid during normal operation
    - Fuel solubility in water (draining tanks)
    - Source term evaluation
- **Build a reactor risk analysis model**
  - Identify the **initiators and high risk scenarios** that require detailed transient analysis
  - Evaluate the risk due to the **residual heat and the radioactive inventory** in the whole system, including the reprocessing units (chemical and)
  - Evaluate some potential design solutions (**barriers**)
  - Allow reactor designer to estimate impact of design changes (***design by safety***)

MSFR and Safety Evaluation: example of accidental scenario

PhD Thesis of M. Brovchenko

Initiators (failure mode)	Dangerous Phenomena	Transient	Consequences	Concept adaptation
Identification + occurrence probability	Accident classification	Physical study of the reactor	Identified risks? Loss of barriers?	Prevention barrier? Protection? Damages limitation?

MSFR and Safety Evaluation: example of accidental scenario

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Initiators (failure mode) → Identification + occurrence probability

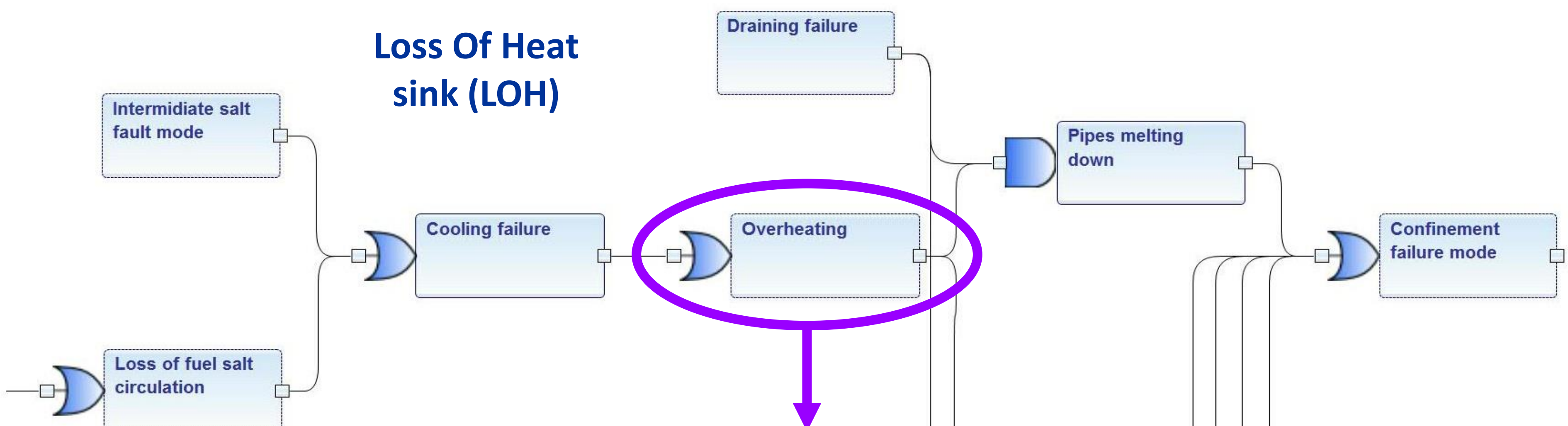
Dangerous Phenomena → Accident classification

Transient → Physical study of the reactor

Consequences → Identified risks? Loss of barriers?

Concept adaptation → Prevention barrier? Protection? Damages limitation?

Loss Of Heat sink (LOH)



Different transients depending on initial failure

MSFR and Safety Evaluation: example of accidental scenario

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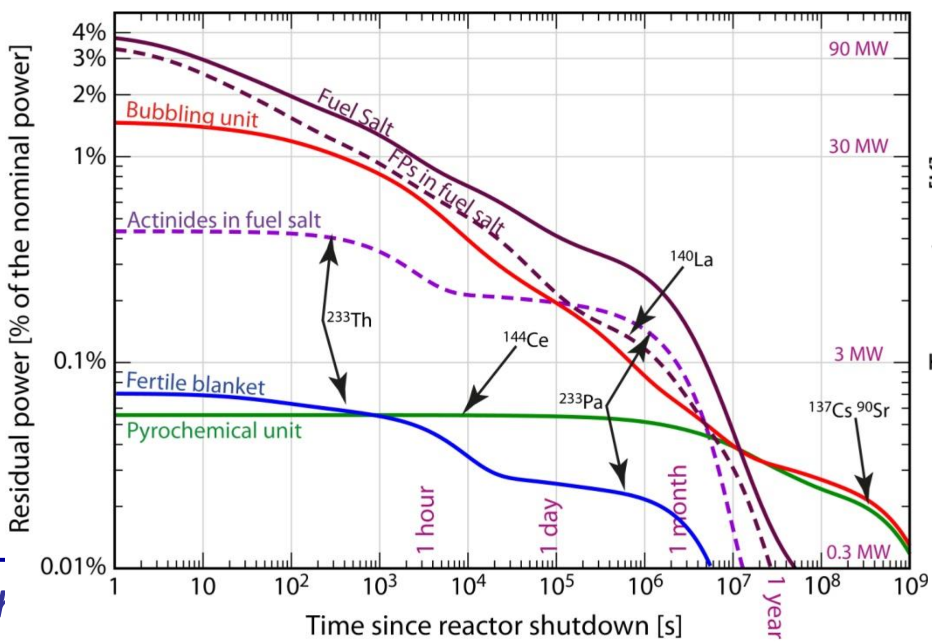
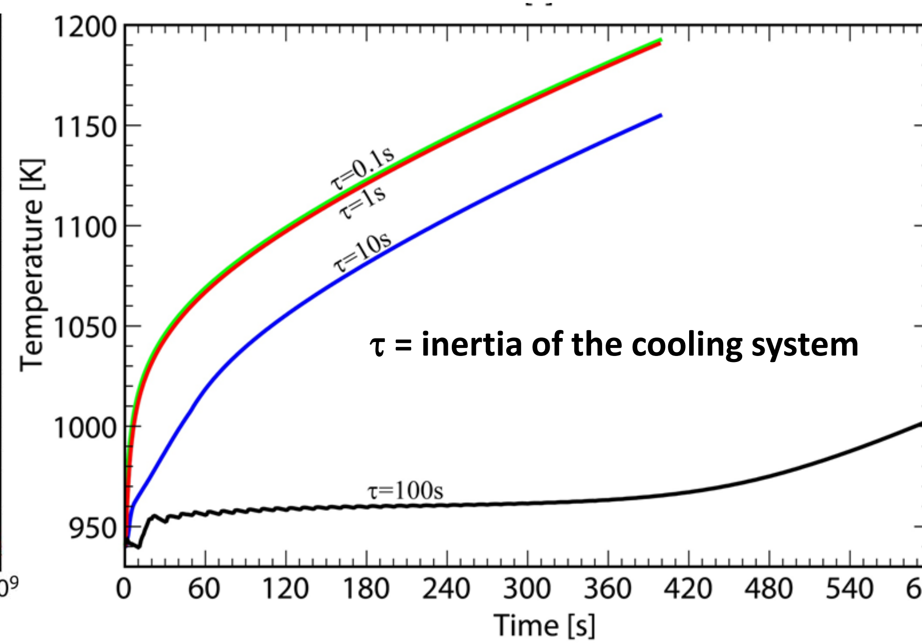
Scenario = passive decrease of the chain reaction (thermal feedback coefficients)
+ increase of the fuel salt temperature due to residual heat





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Risks identified:

- Continuous heating due to the residual power (physics)
- Increase of temperature : impact of the pump inertia (technology)

Protection:

- Draining of the fuel salt
- Thermal protection on the walls?

'Design by Safety' approach

Quantitatively: Risk = Probability x Severity

Accident probabilities and severity difficult to quantify at the current preliminary design stage

Demonstration 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

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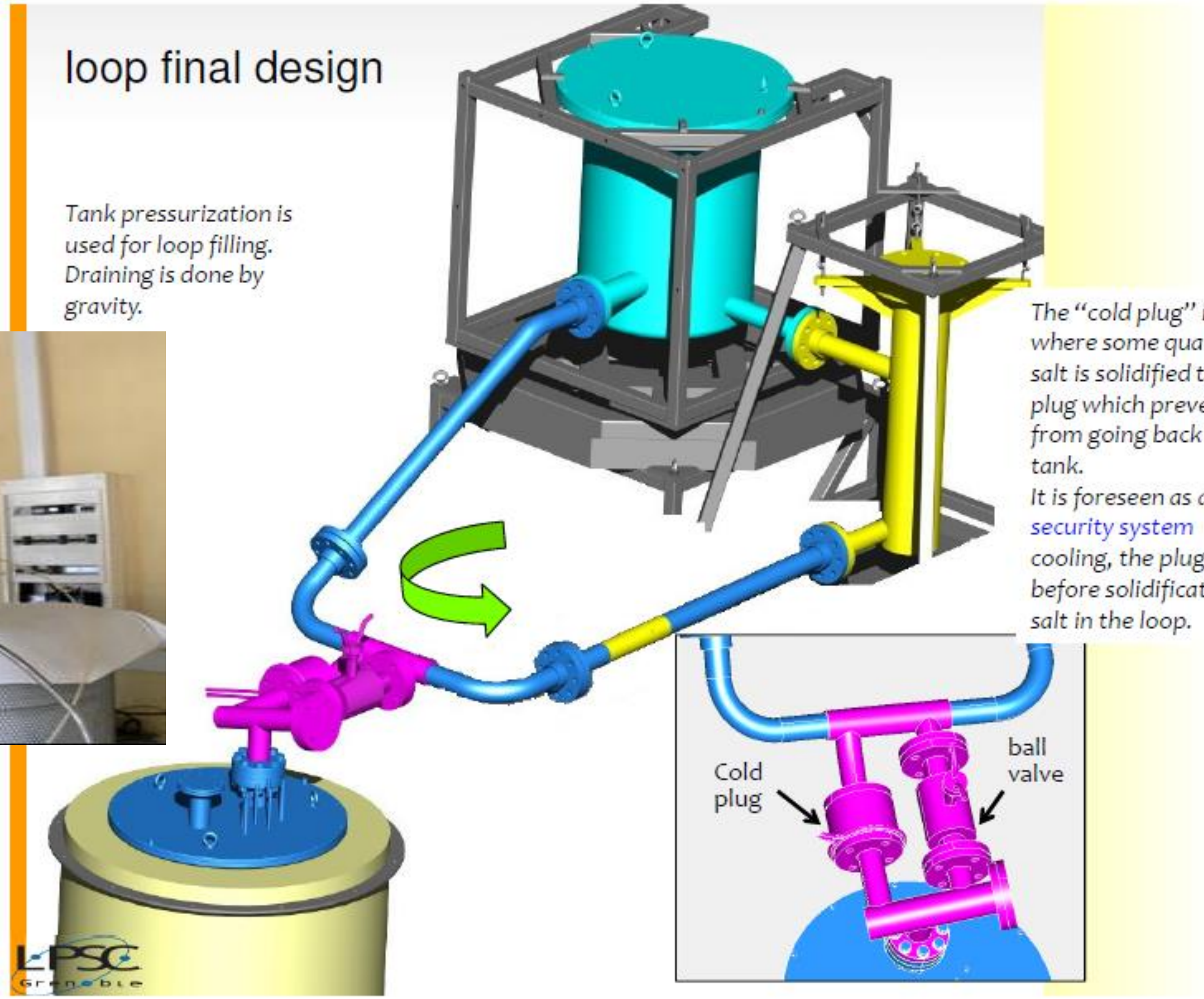
Demonstration and Demonstrator of MSFR: the FFFER facility

The Forced Fluoride Flow Experiment

Reproduces the gases and particles
extractions at 1/10th flow scale in simulant salt



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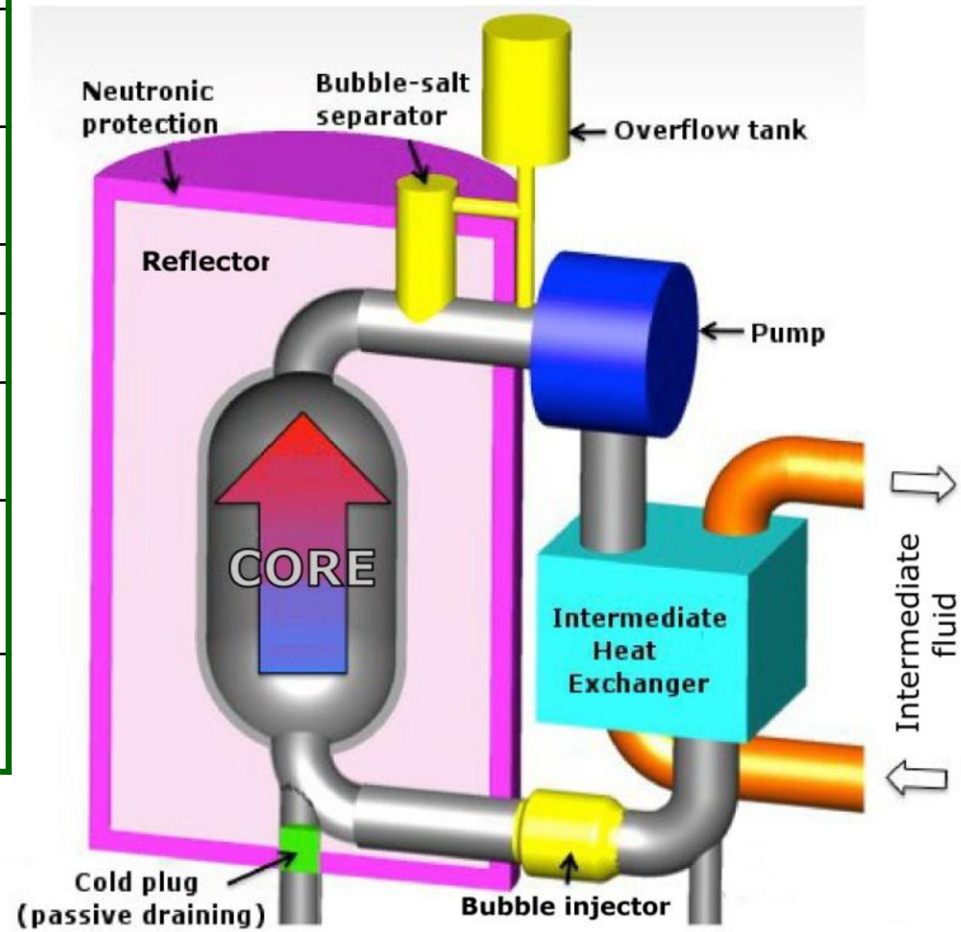
Power Demonstrator of the MSFR

Thermal power	**100 MWth**
Mean fuel salt temperature	**725 °C**
Fuel salt temperature rise in the core	**30 °C**
Fuel Molten salt initial composition	**75% LiF-ThF$_4$-^{233}UF$_4$ or LiF-ThF$_4$-(^{enriched}U+MOx-Th)F$_3$**
Fuel salt melting point	**565 °C**
Fuel salt density	**4.1 g/cm^3**
Core dimensions	**Diameter: 1.112 m** **Height: 1.112 m**
Fuel Salt Volume	**1.8 m^3** **1.08 in core** **0.72 in external circuits**
Total fuel salt cycle in the fuel circuit	**3.5 s**

Demonstrator characteristics representative of the MSFR

From the power reactor to the demonstrator:
Power / 30 and Volume / 10



6 external loops

MSFR : Conclusions and Perspectives

Summary: Definition of an innovative Molten Salt configuration with a Fast Neutron Spectrum, based firstly on reactor physics studies and including now more largely system developments (chemistry, thermal-hydraulics, materials, safety, design…)

Perspectives

⇨ Where?

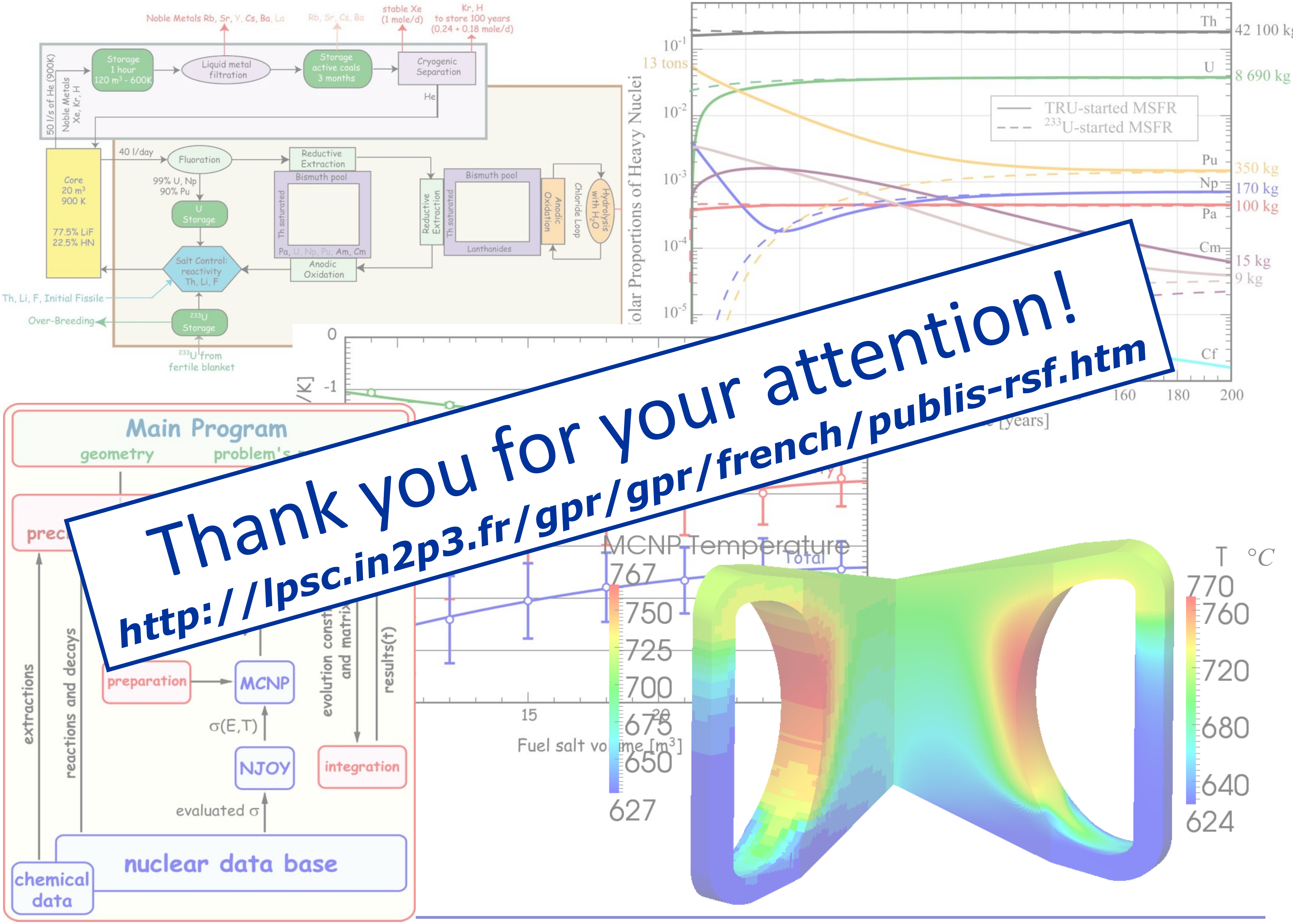
- National programs: CNRS (IN2P3…) and multidisciplinary program NEEDS – Collaborations with IRSN (and EdF/AREVA?) + Structuring project CLEF of Grenoble INP
- European project EVOL (FP7) with Rosatom: finished end 2013 – Next project in Horizon 2020?
- International: MSR MoU (GIF) to be signed by ROSATOM - Other collaborations (China , Japan, USA…)?

⇨ Optimization of the system and symbiotic safety/design studies

- Multi-physics and multi-scale coupling tool for a global simulation of the system
- Design of the reactor, draining and processing systems (including materials, components…)
- Risk analysis and safety approach dedicated to MSFR
- Define the demonstration steps and experimental facilities

Thank you for your attention!

http://lpsc.in2p3.fr/gpr/gpr/french/publis-rsf.htm



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MSFR: choice of the liquid fluid

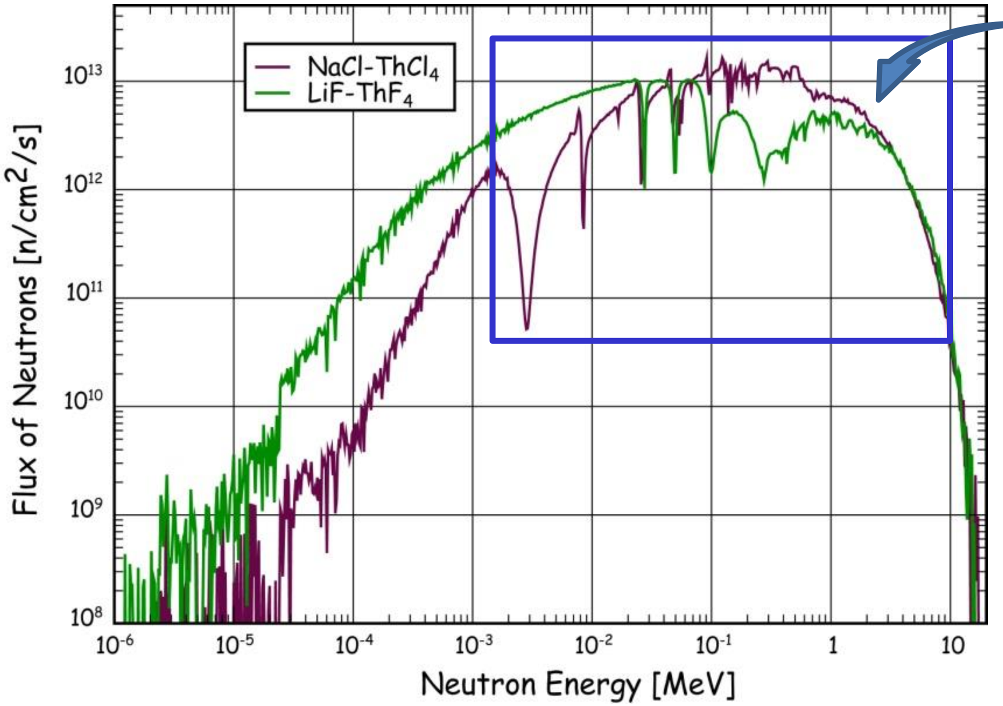
Element produced	Problem	Fluoride Salt	Chloride Salt
^{36}Cl produced via $^{35}Cl(n,\gamma)^{36}Cl$ and $^{37}Cl(n,2n)^{36}Cl$	Radioactivity - $T_{1/2}$ = 301000y		10 moles / y (373 g/year)
^{3}H produced via $^{6}Li(n,\alpha)$ t and $^{6}Li(n,t)$ α	Radioactivity - $T_{1/2}$ = 12 years	55 moles / y (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

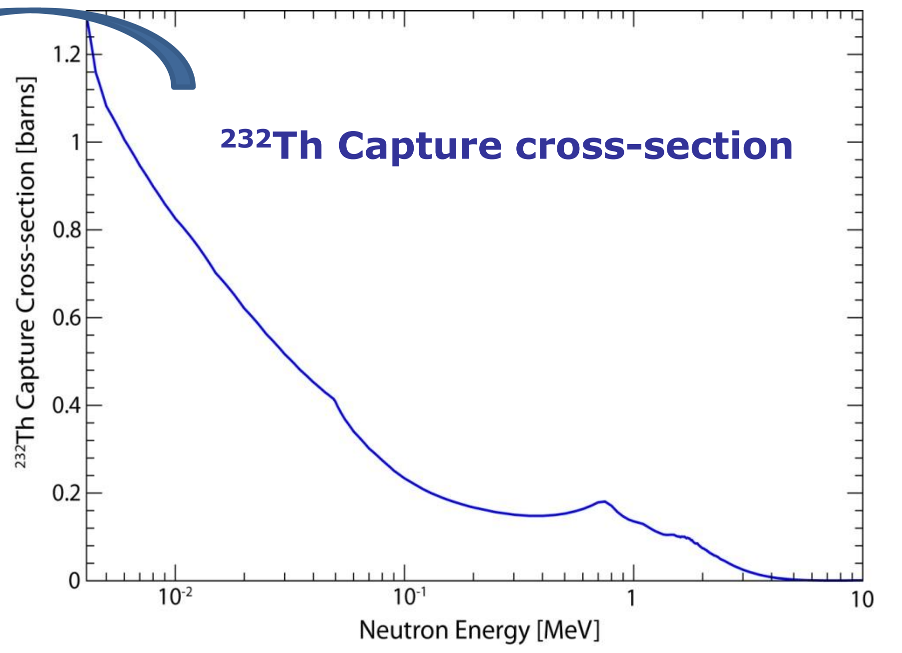
Combination of both neutronic and chemical considerations

⇩

MSFR based on a molten LiF fuel salt

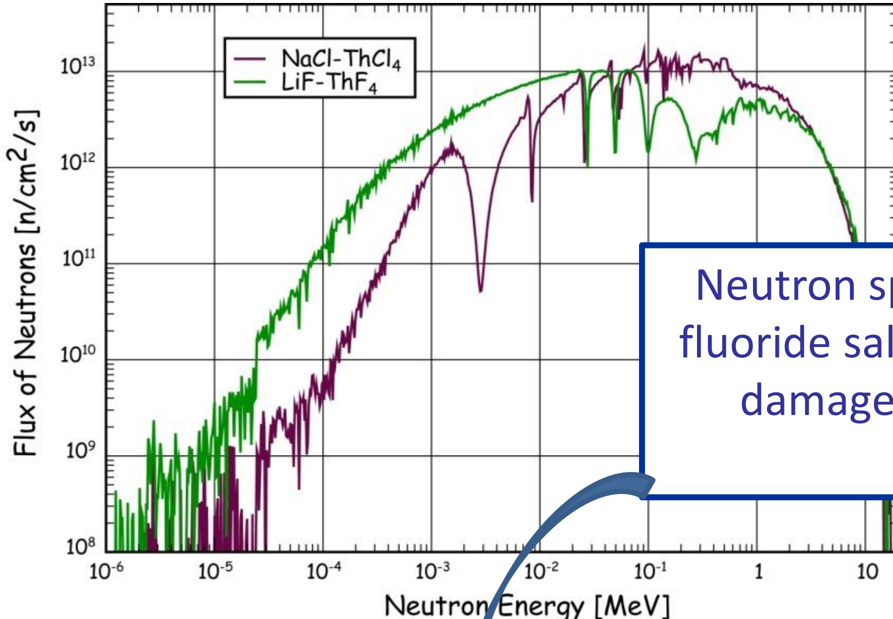
MSFR: choice of the liquid fluid



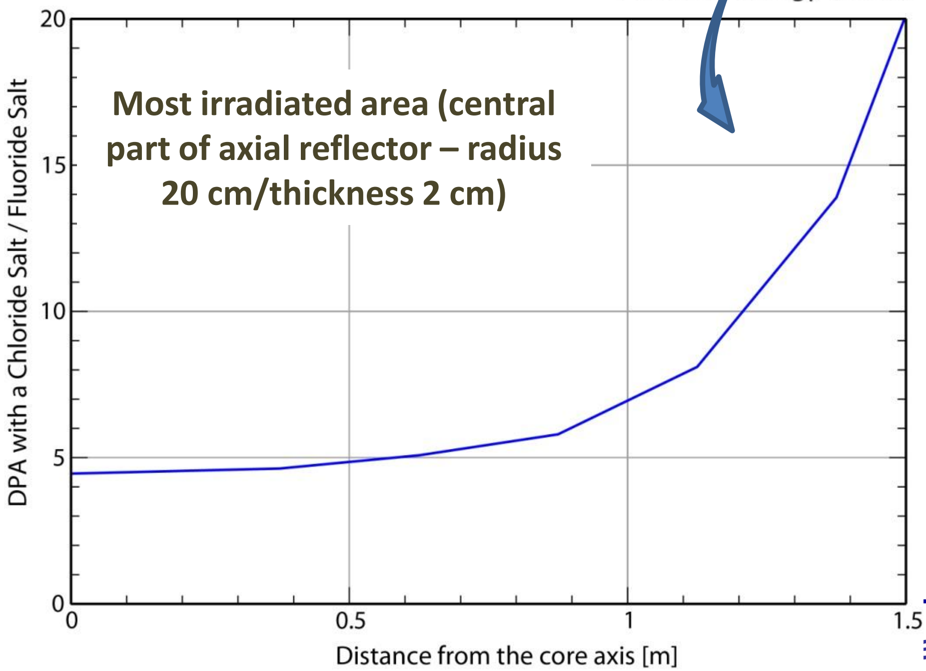


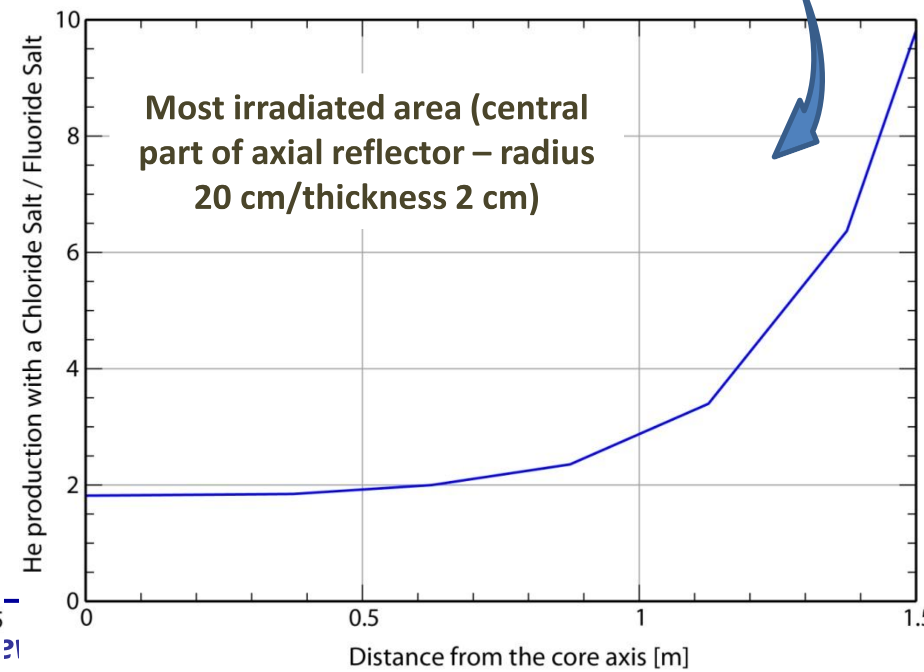
Parameter	Fluoride Salt	Chloride Salt
Thorium capture cross-section in core (barn)	0.61	0.315
Thorium amount in core (kg)	42 340	47 160
Thorium capture rate in core (mole/day)	11.03	8.48
Thorium capture cross-section in blanket (barn)	0.91	0.48
Thorium amount in the blanket (kg)	25 930	36 400
Thorium 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

MSFR: choice of the liquid fluid



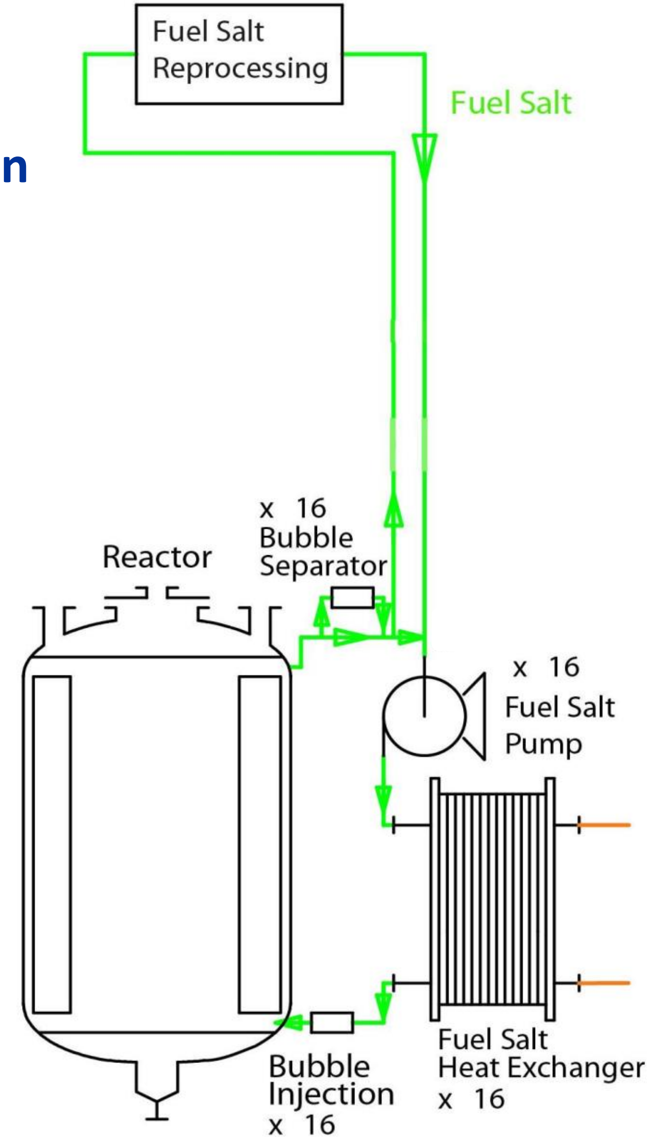
Neutron spectrum less fast with fluoride salt = reduced irradiation damages (both DPA and He production)





MSFR: conceptual design of the salt heat exchangers

"Fuel Salt Loop" = Includes all the systems **in contact** with the fuel salt during **normal operation**



Core:

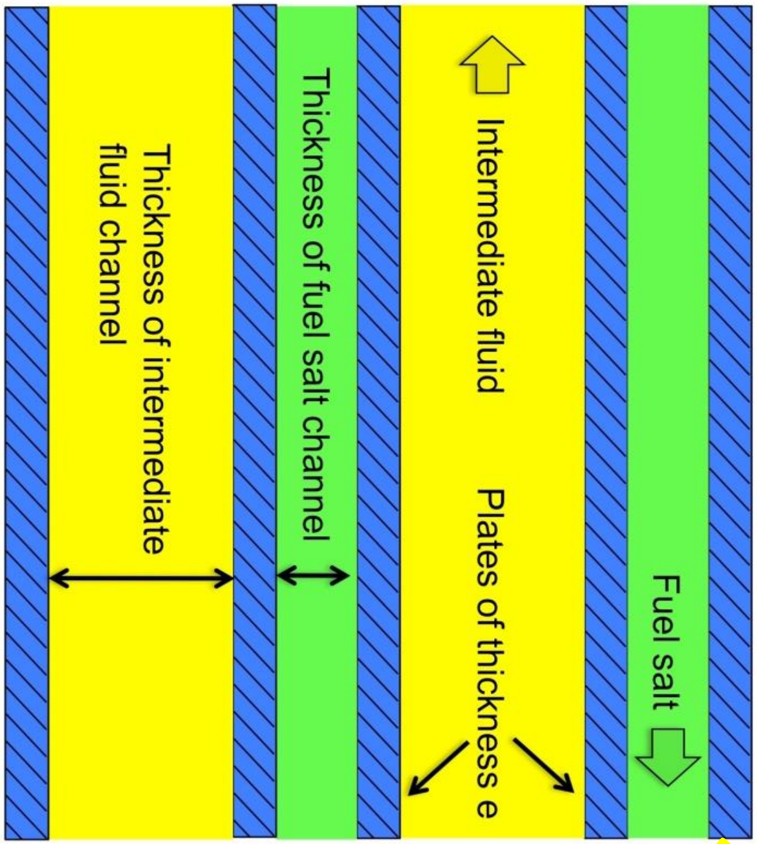
No inside structure

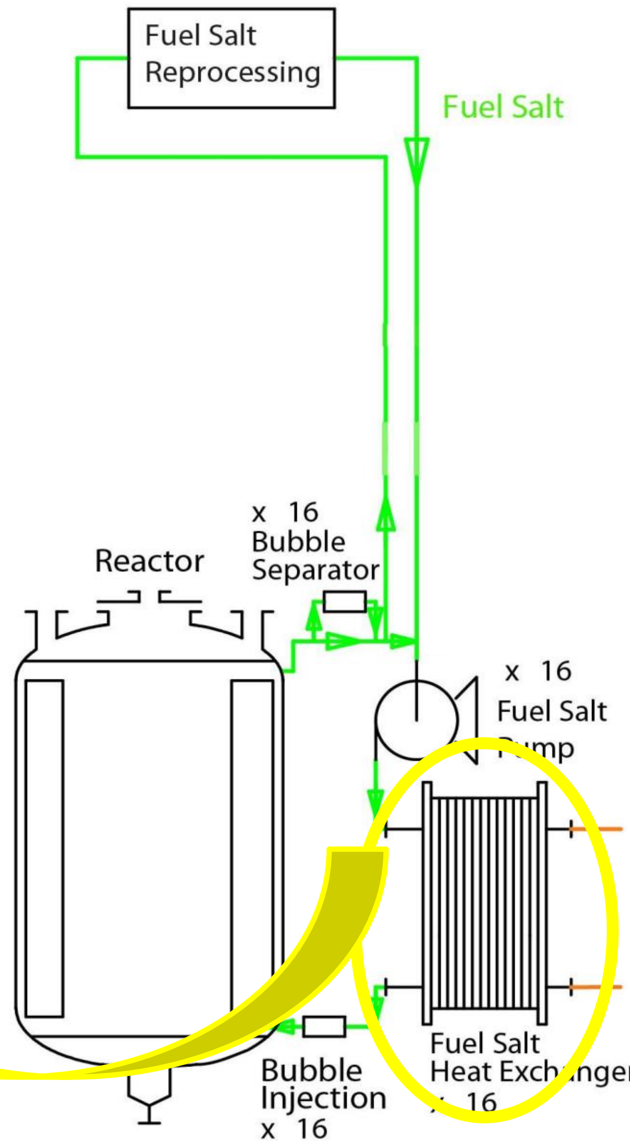
Outside structure: Upper and lower Reflectors, Fertile Blanket Wall

+ 16 external modules:

- Pipes (cold and hot region)
- Bubble Separator
- Pump
- Heat Exchanger
- Bubble Injection

MSFR: conceptual design of the salt heat exchangers





Two kind of intermediate fluid considered in this study: liquid metal or fluoride salt

MSFR: conceptual design of the salt heat exchangers

Constrained Parameter	Limiting value (P_{0i})	Acceptable deviation (σ_i)
Minimum thickness of the fuel salt channel	2.5 mm	0.05 mm
Minimum thickness of the plate	1.75 mm	0.035 mm
Maximum speed of the fuel salt	3.5 m/s	0.07 m/s
Maximum speed of the intermediate fluid (liquid lead)	1.75 m/s	0.035 m/s
Maximum speed of the intermediate fluid (salt)	5.5 m/s	0.11 m/s
Maximum temperature of the materials	700 °C	1 °C
Minimum margin to solidification of the fuel salt	50 °C	1 °C
Minimum margin to solidification of the intermediate fluid	40 °C	1 °C

Each set of values of the variable parameters evaluated with the quality function:

$$\prod_i exp\left(\frac{P_i - P_{0i}}{\sigma_i}\right)$$

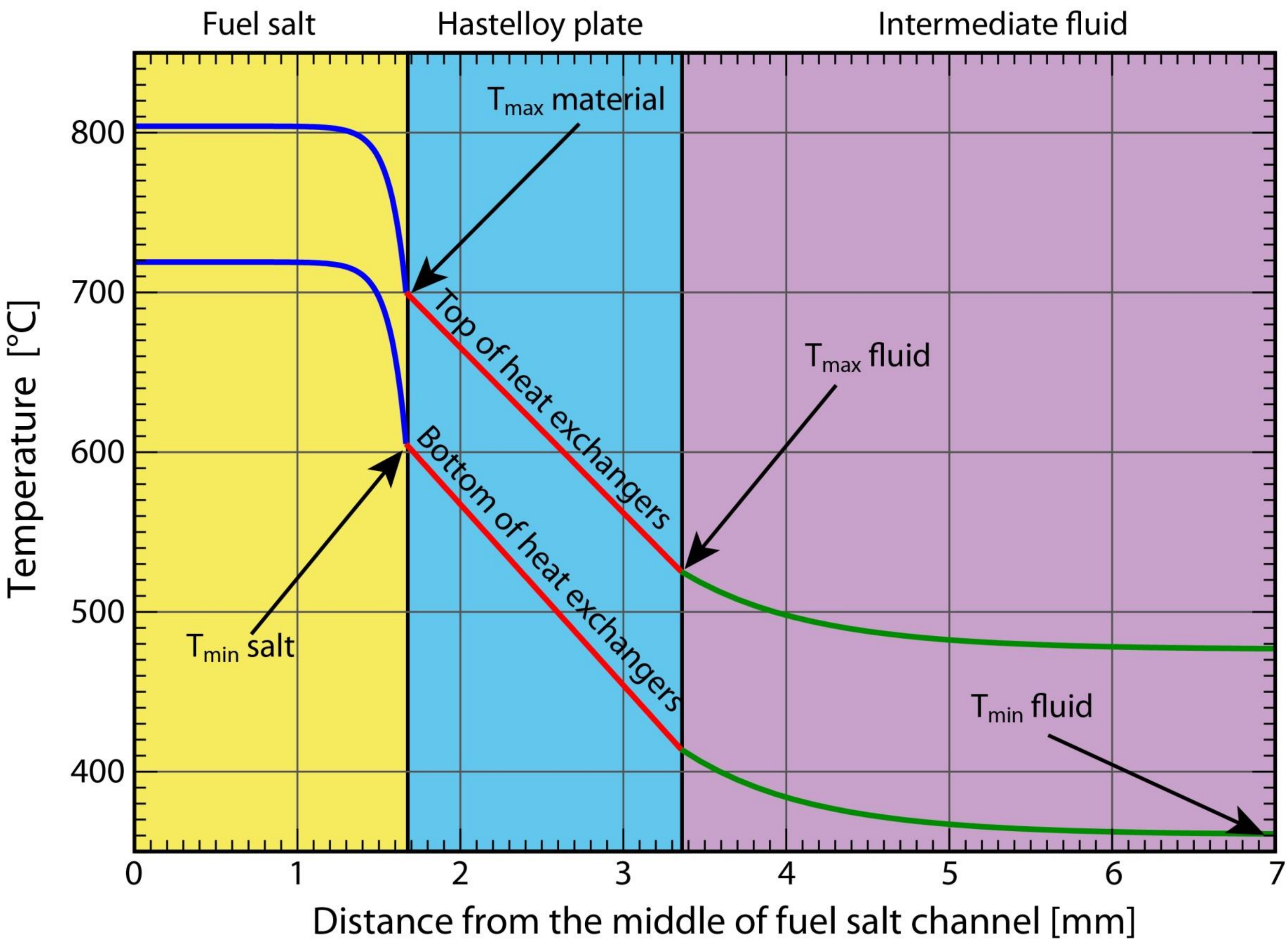
Variables of the study:

- ✓ the diameter of the pipes
- ✓ the thickness of the plates
- ✓ the gap between the plates on the intermediate fluid side (or "thickness of the intermediate fluid channel")
- ✓ the fuel salt temperature at core entrance
- ✓ the fuel salt temperature increase within the core
- ✓ the temperature increase of the intermediate fluid in the heat exchangers
- ✓ the mean temperature difference between the two fluids within the heat exchangers

MSFR: conceptual design of the salt heat exchangers

Evaluated parameter	Pb	FLiNaK	NaF-NaBF$_4$
Diameter of the fuel salt pipes [mm]	301	283	303
Diameter of the intermediate fluid pipes [mm]	897	507	470
Thickness of the plates [mm]	1.61	1.51	1.65
Fuel salt temperature at core entrance [°C]	754	698	704
Fuel salt temperature increase in the core [°C]	89	106	98
Intermediate fluid temperature increase within the heat exchangers [°C]	99	41	66
Mean temperature difference between the two fluids in the heat exchangers [°C]	382	242	280
Intermediate fluid temperature at the heat exch. outlet [°C]	466	530	506
Thickness of the fuel salt channel [mm]	3.38	2.17	2.37
Thickness of the intermediate fluid channel [mm]	29.8	4.49	4.38
Fuel salt speed in the pipes [m/s]	3.92	3.97	3.73
Fuel salt speed in the heat exchangers [m/s]	3.85	2.36	2.91
Intermediate fluid speed in the pipes [m/s]	1.94	6.00	5.67
Intermediate fluid speed in the heat exchangers [m/s]	1.92	5.54	5.75
Maximum temperature of the intermediate fluid [°C]	523	622	595
Maximum temperature of the materials [°C]	701	701	699
Margin to the solidification of the fuel salt [°C]	43.7	54.7	46.7
Margin to the solidification of the intermediate fluid [°C]	39.6	34.5	56.2
Pressure loss of the fuel salt in the heat exchangers [bar]	2.56	2.03	2.56
Pressure loss of the fuel salt in the pipes [bar]	0.99	1.02	0.90
Pressure loss of the intermediate fluid in the heat exch. [bar]	0.09	2.09	1.66
Pressure loss of the intermediate fluid in the pipes [bar]	0.32	0.71	0.57

MSFR: conceptual design of the salt heat exchangers



The concept of Molten Salt Fast Reactor

Design of the reference MSFR

- Initial Salt: 77.5%LiF – 2.5% $^{233}UF_3$ - ThF_4
- ^{233}U initial inventory per $GW_{él}$: 3260 kg
- ^{233}U production (breeder reactor): 95 kg/year
- Feedback Coefficient: -5 pcm/K

- Fuel Salt Temperature: 750 °C
- Produced power: 3 GW_{th} (~1.5 GW_{el})

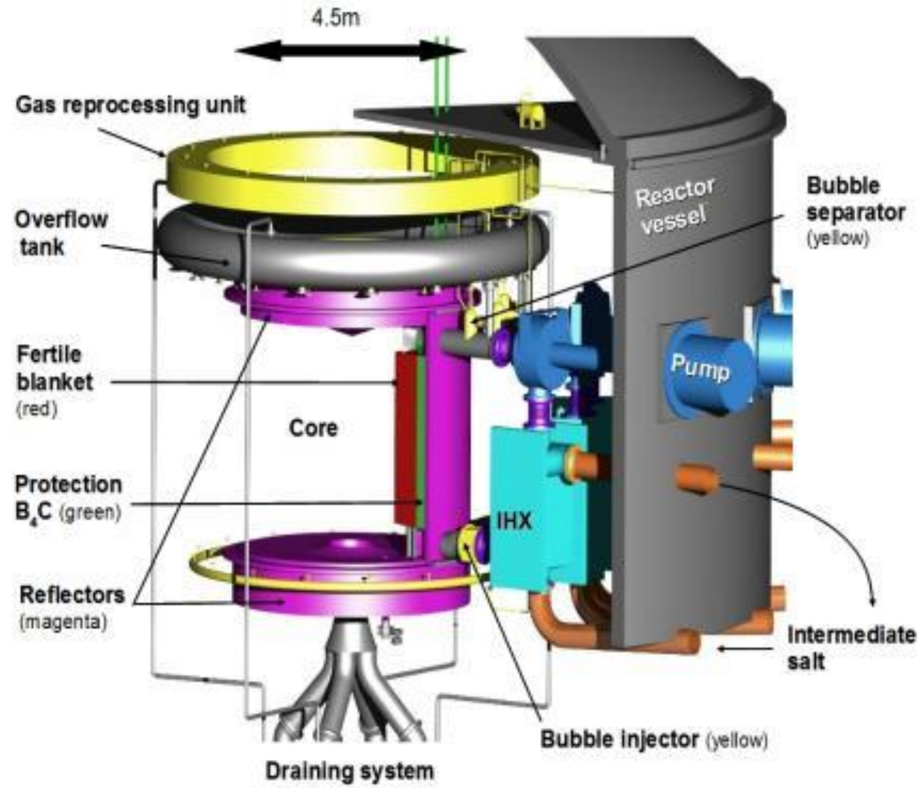
- Core Internal Diameter = Core Height = 2.3 m

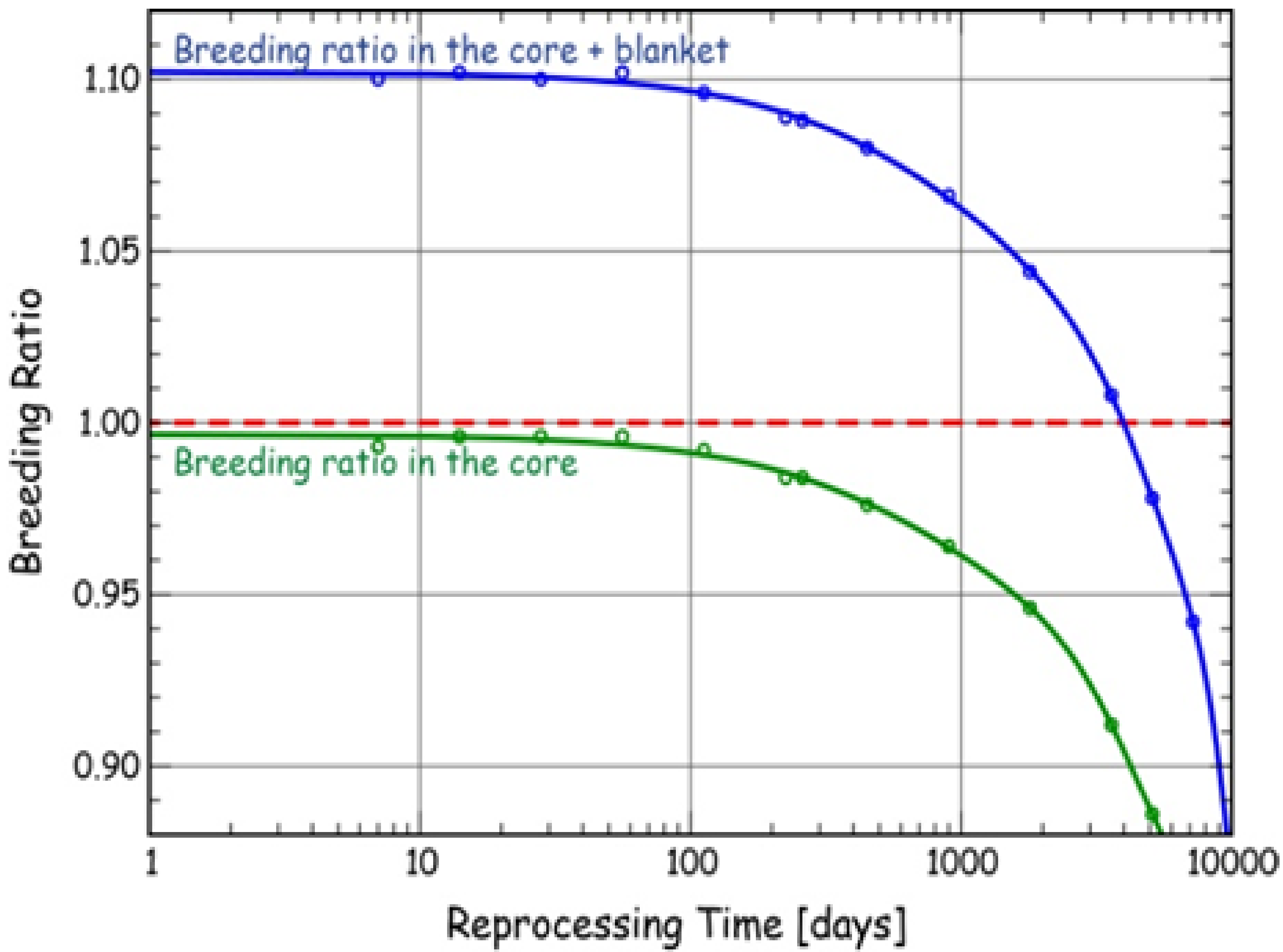
- Fuel Salt Volume: 18 m^3
    - 1/2 in the active zone (core + plenums)
    - 1/2 in the external circuit (heat exchangers, pipes, pumps)

- Thickness of Fertile Blanket: 50 cm
- Volume of Fertile Blanket: 7.7 m^3
- Initial Fertile Salt: 77.5%LiF - 22.5%ThF_4

- Core reprocessing: 10 to 40 l of fuel salt cleaned per day (on-site batch reprocessing for lanthanides extraction) + on-line He bubbling in the core

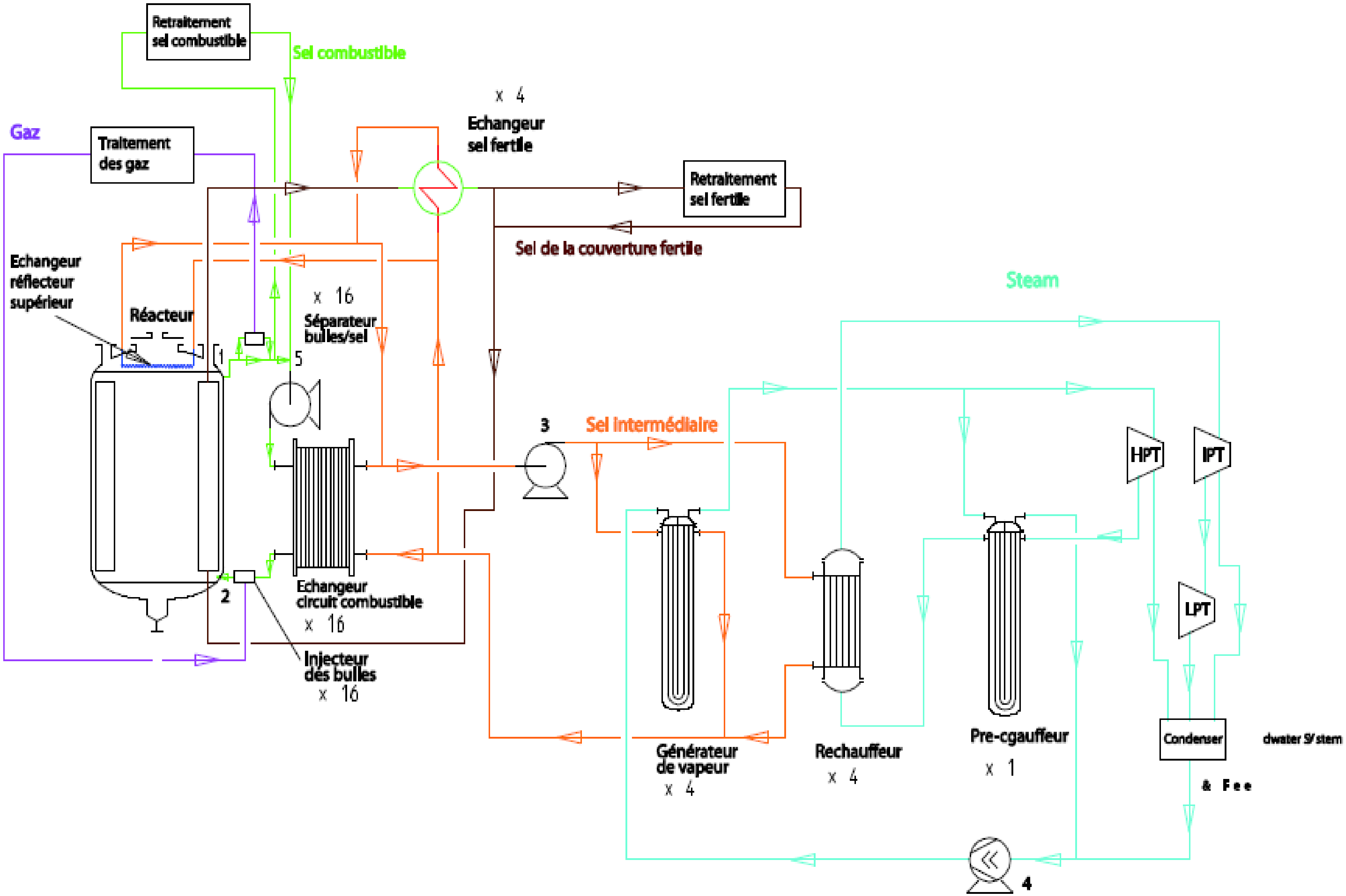
MSFR concept selected for further studies by the GIF "MSR Steering Committee" – Choice approved by the Policy Group (since 2008)



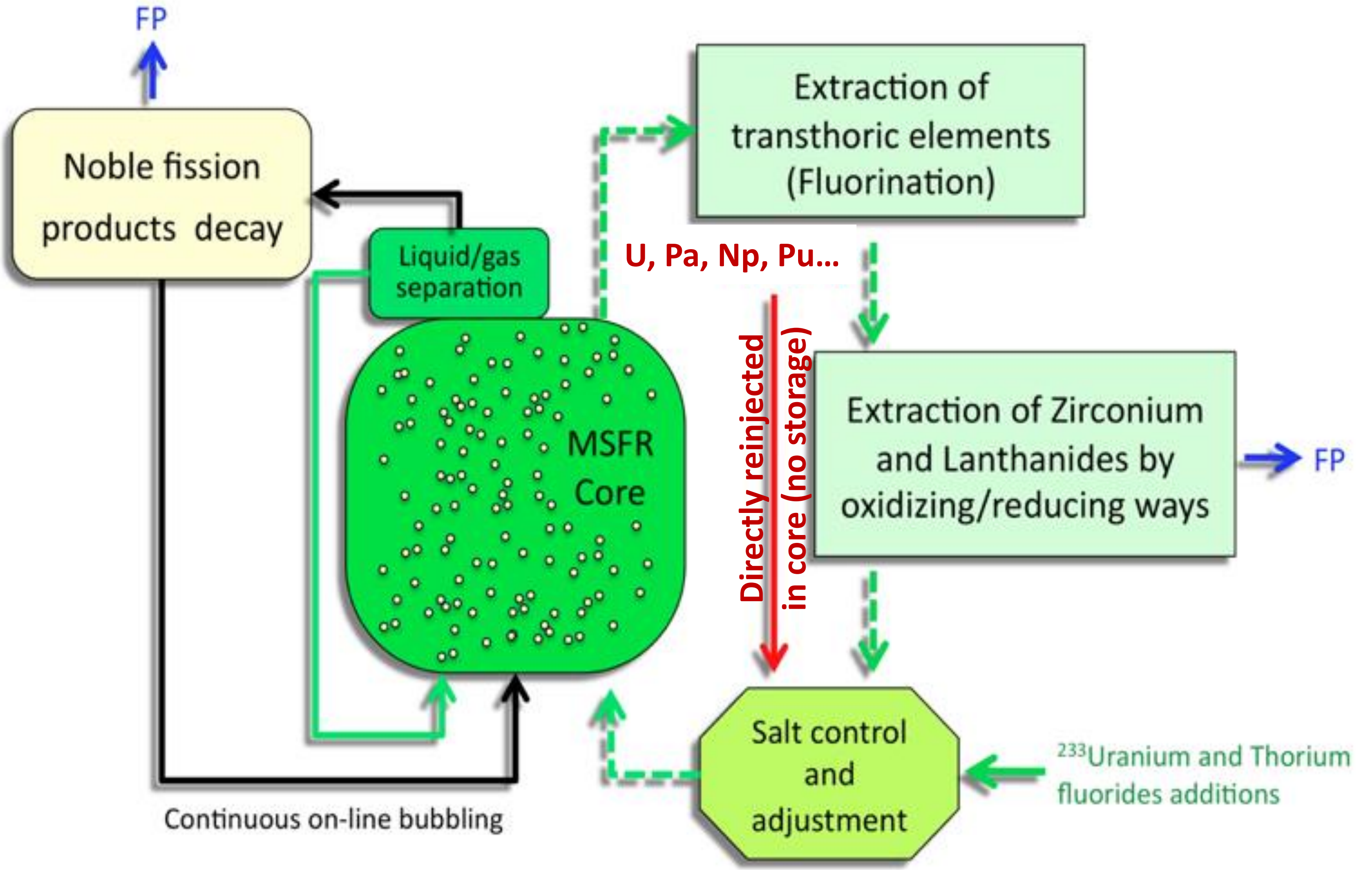


Breeding ratio in the core + blanket
Breeding ratio in the core
Breeding Ratio
1.10
1.05
1.00
0.95
0.90
1
10
100
1000
10000
Reprocessing Time [days]

Initial Fuel Salt Composition – EVOL Benchmark				
^{233}U-started MSFR		TRU-started MSFR		
Th	^{233}U	Th	Actinides	
38 281 kg 19.985 %mol	4 838 kg 2.515 %mol	30 619 kg 16.068 %mol	Pu	11 079 kg 5.628 %mol
			Np	789 kg 0.405 %mol
			Am	677 kg 0.341 %mol
			Cm	116 kg 0.058 %mol



Retraitement sel combustible
Sel combustible
x 4
Echangeur sel fertile
Gaz
Traitement des gaz
Retraitement sel fertile
Sel de la couverture fertile
Echangeur réflecteur supérieur
Réacteur
x 16
Séparateur bulles/sel
5
1
Steam
3
Sel Intermédiaire
HPT
IPT
LPT
2
Echangeur circuit combustible
x 16
Injecteur des bulles
x 16
Générateur de vapeur
x 4
Rechauffeur
x 4
Pre-cgauffeur
x 1
Condenser
dwater Sy stem
& Fee
4


FP
Noble fission products decay
Liquid/gas separation
MSFR Core
Continuous on-line bubbling
Extraction of transthoric elements (Fluorination)
U, Pa, Np, Pu…
Directly reinjected in core (no storage)
Extraction of Zirconium and Lanthanides by oxidizing/reducing ways
FP
Salt control and adjustment
233Uranium and Thorium fluorides additions

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

Which constraints for a liquid fuel?

- Melting temperature not too high
- High boiling temperature
- Low vapor pressure
- 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

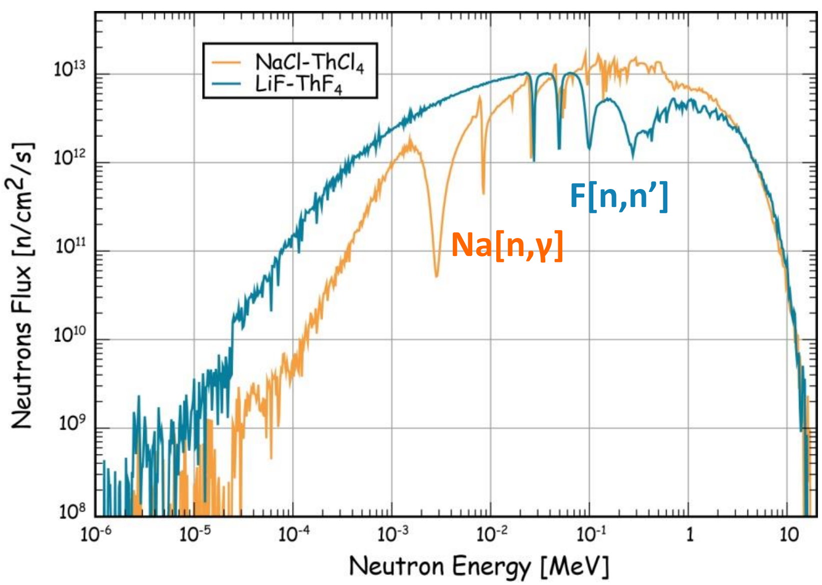
Lithium fluorides fulfill all constraints

Molten Salt Reactors

+

Neutronic cross-sections of fluorine versus neutron economy in the fuel cycle

Thorium / ^{233}U Fuel Cycle



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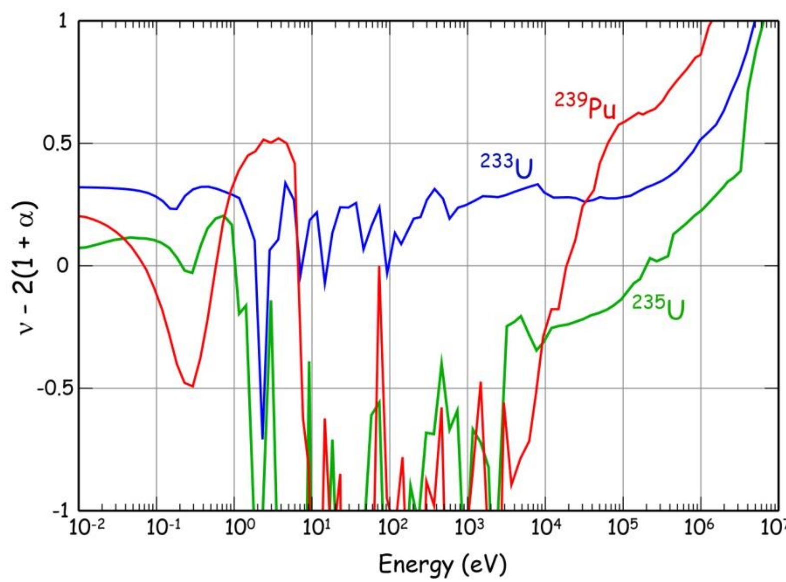
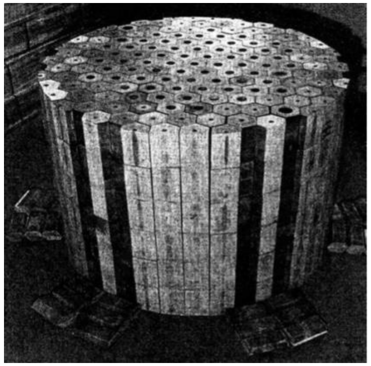


Molten Salt Reactor (MSR): Historical studies

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

Future of nuclear reactors: 4th Generation Systems

Generation 4 International Forum: Criteria for Future Nuclear Reactors

⇒ Development of innovative MSR concepts to fulfill these criteria

Sustainable development

- Availability
  - Long term availability of the system
  - Resources availability → Reactors at least breeder
- Minimization of the waste production
  - Recycling of Actinides + Minimizing the MA production
  - Minimizing the Industrial Wastes (structural elements and processes)
- Deployment capacities
  - Minimizing the Initial Fissile Inventory versus breeding
  - Availability of the Initial Fissile Matter

Optimal Safety and Reliability

- Reduction of major accident/incident's initiators
- Risks and consequences of core damages limited
  - No inflammable matters in the core, no high pressure
  - Minimized reactivity margins
  - All negative safety coefficients

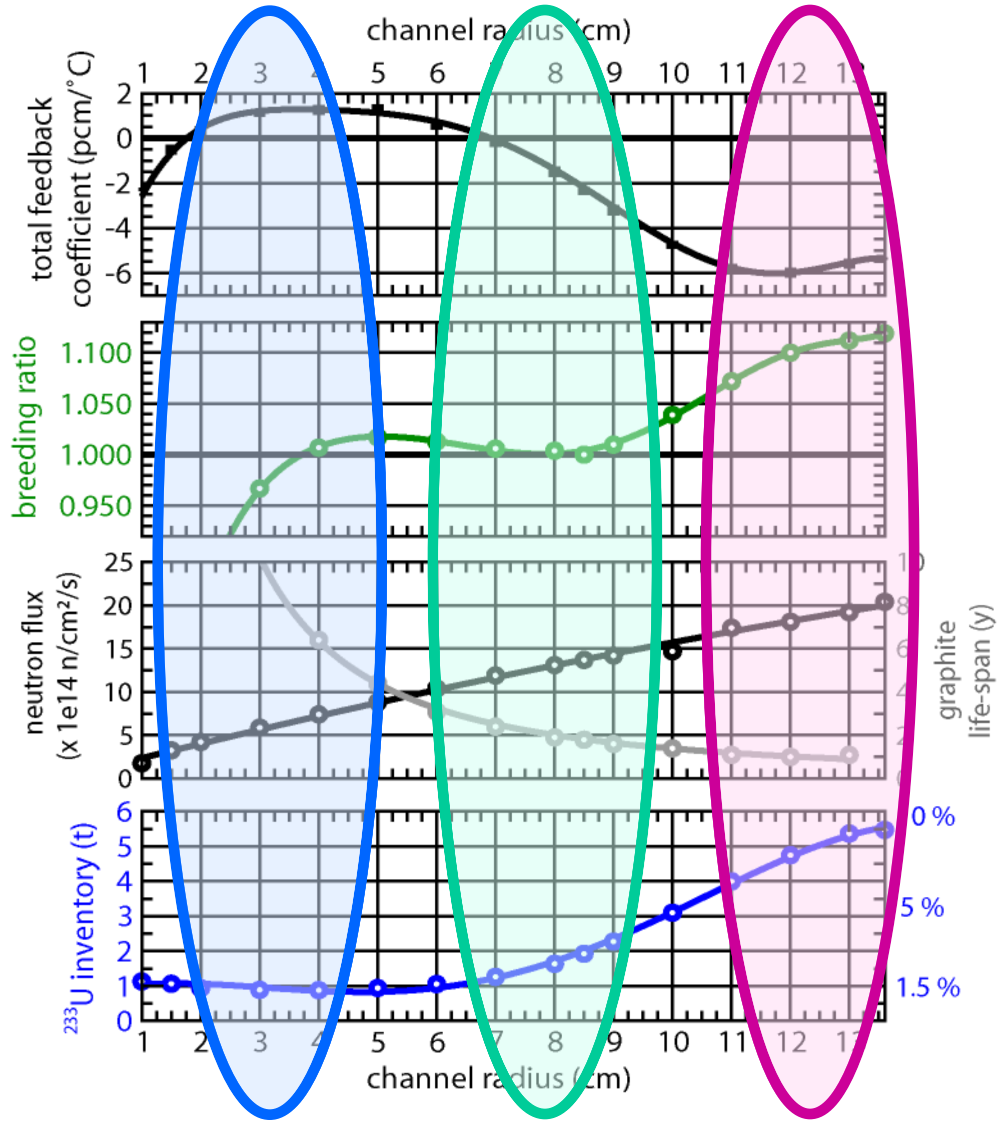
Proliferation Resistance and Physical Protection

Economic Competitiveness

Historical MSR Studies at CNRS
Influence of the channel radius on the core behavior

<u>Three types of configuration:</u>

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



Historical MSR Studies at CNRS

Influence of the channel radius on the core behavior

PhD thesis of Ludovic MATHIEU

Thermal spectrum configurations

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

Epithermal spectrum configurations

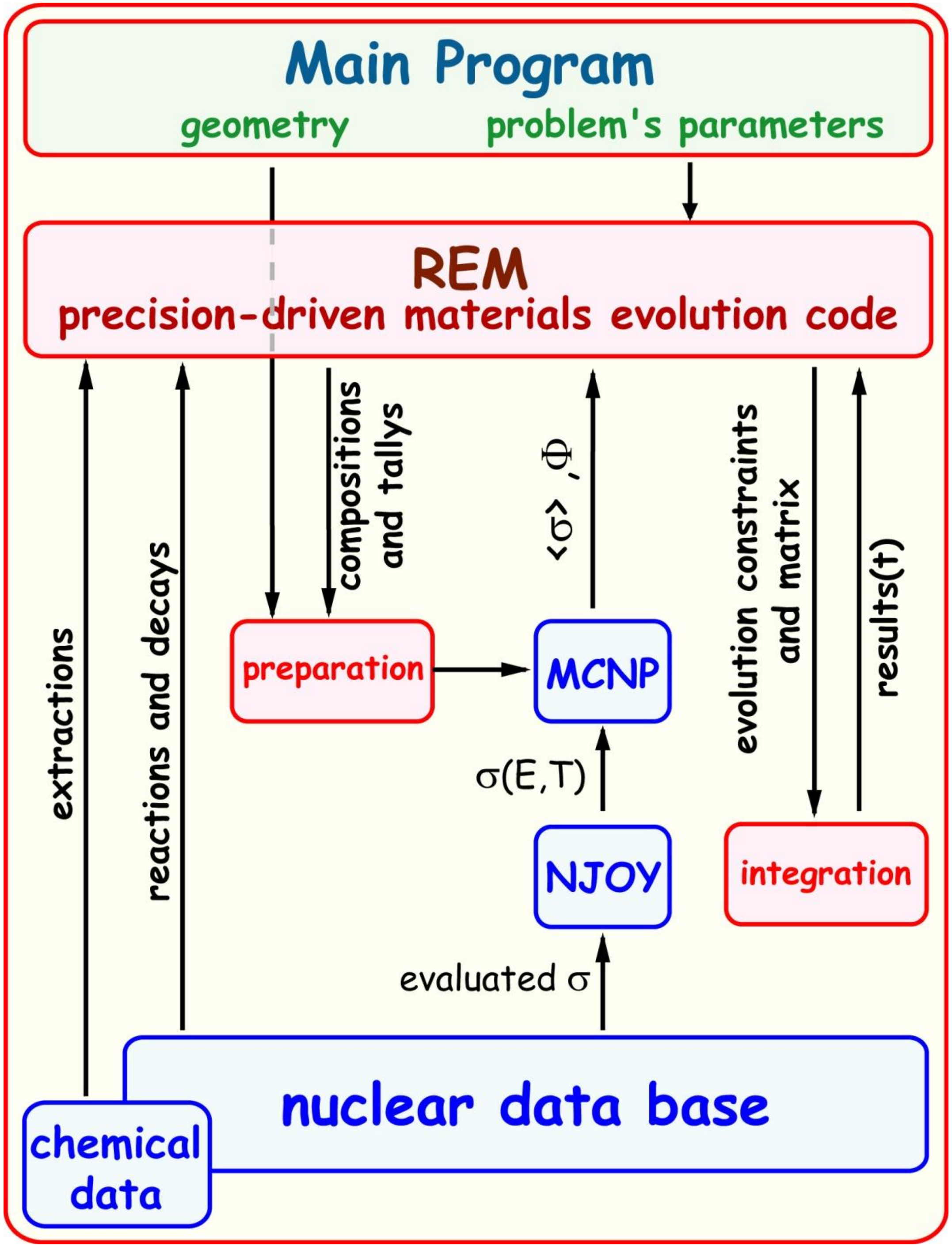
- quite negative feedback coefficient
- iso-breeder
- **very short graphite life-span**
- **quite low ^{233}U initial inventory**

Fast spectrum configurations (no moderator)

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

Tools for the Simulation of Reactor Evolution:
Details of the program

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:
Integration Module: Bateman Equation for nucleus i

$$\frac{\partial N_i}{\partial t} = \sum_j \Big(\underbrace{\langle \sigma_j \phi(t) \rangle N_j(t) b_{j\to i} + \lambda_j N_j(t) b'_{j\to i}}_{\text{production from nucleus } j} \Big) - \Big(\underbrace{\langle \sigma_i \phi(t) \rangle N_i(t) + \lambda_i N_i(t)}_{\text{disappearance}} \Big)$$

sum over all nuclei j

production by nuclear reaction

production by radioactive decay

disappearance by nuclear reaction

disappearance by radioactive decay

Molten Salt Reactors: addition of a feeding term, equal to the number of nuclei added per time unit for each element (flow)

Reprocessing: new terms $-\lambda_i^{extr.} N_i$ with $\lambda_i^{extr.} = \dfrac{1}{T_i^{reprocess.}}$

Efficiency linked to the nucleus extraction probability

The concept of MSFR: Fuel Reprocessing

Fission Products Extraction: Motivations

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

✓Keep good neutronic properties

Physical Separation (in the core)

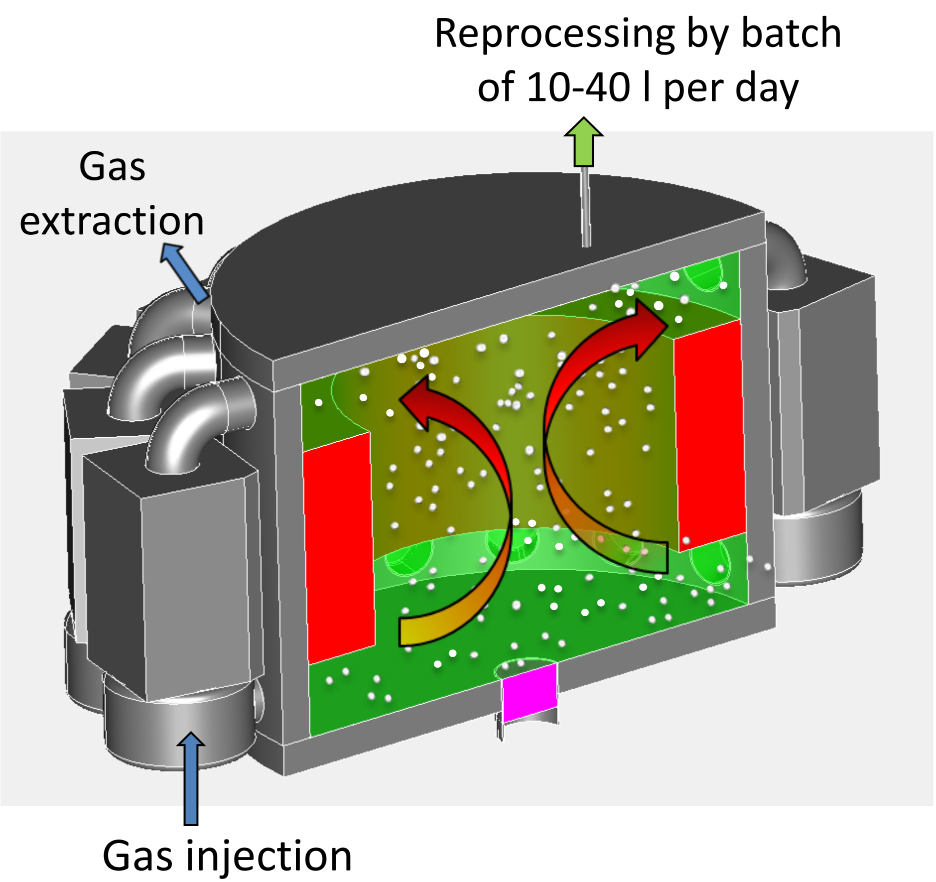
➢Gas Reprocessing Unit through bubbling extraction

➢Extract Kr, Xe, He and particles in suspension

Chemical Separation (by batch)

➢Pyrochemical Reprocessing Unit

➢Located on-site, but outside the reactor vessel



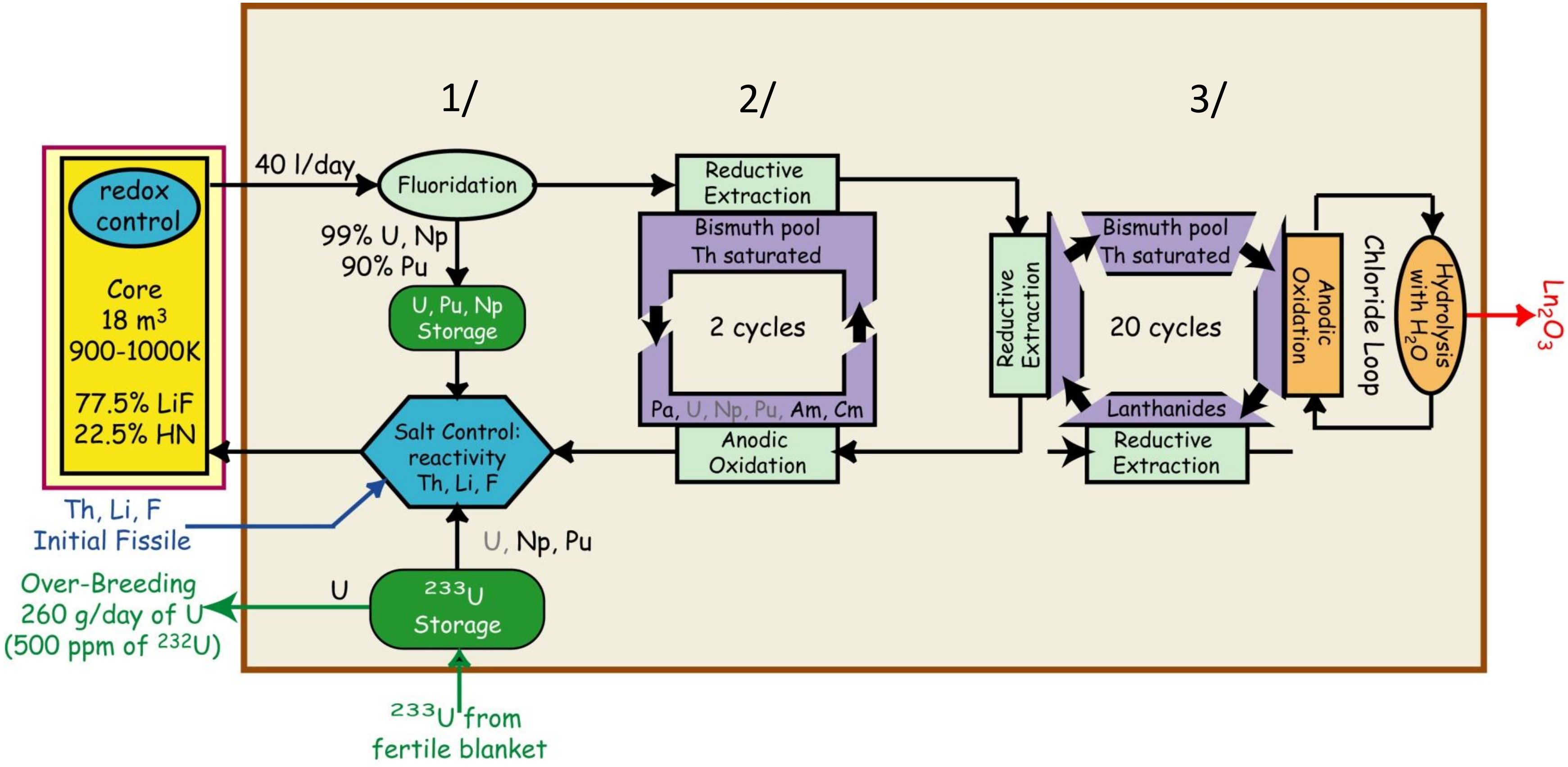
The concept of MSFR: Fuel Reprocessing

On-site Chemical Reprocessing Unit

1/ Salt Control + Fluorination to extract U, Np, Pu + few FPs - Expected efficiency of 99% for U/Np and 90% for Pu – Extracted elements re-injected in core

2/ Reductive extraction to remove actinides (except Th) from the salt – MA re-injected by anodic oxidation in the salt at the core entrance

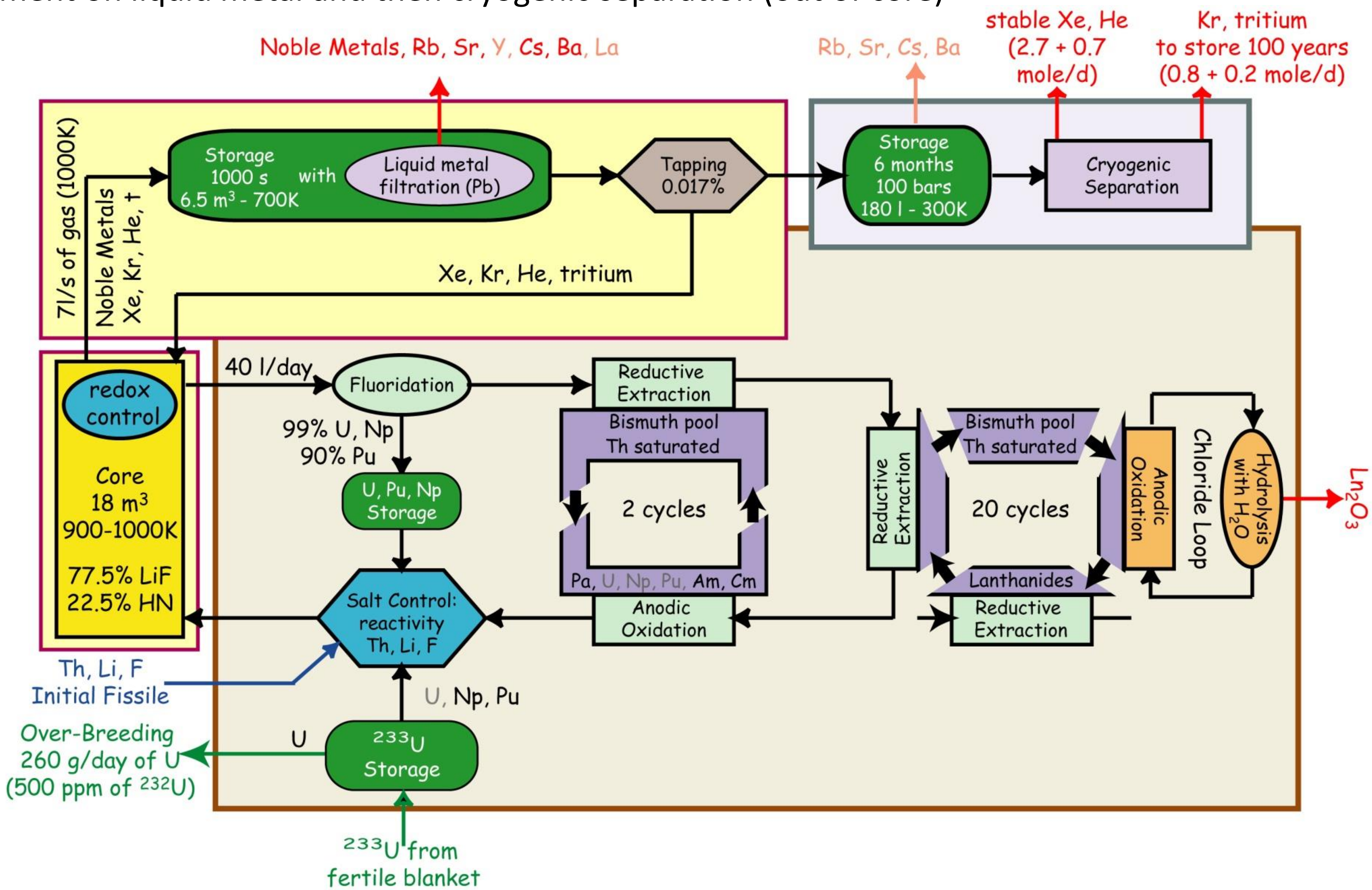
3/ Second reductive extraction to remove all the elements other than the solvent - lanthanides transferred to a chloride salt before being precipitated



The concept of MSFR: Fuel Reprocessing

Noble gazes bubbling in the core (within the fuel salt loop)

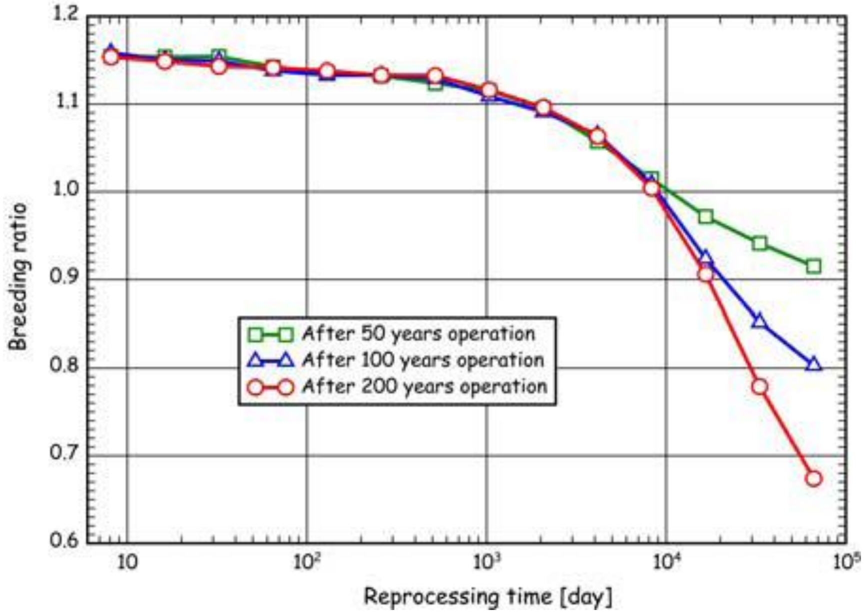
To remove all insoluble fission products (mostly noble metals) and rare gases, helium bubbles are voluntary injected in the flowing liquid salt (bottom of the core) → Separation salt / bubbles → Treatment on liquid metal and then cryogenic separation (out of core)



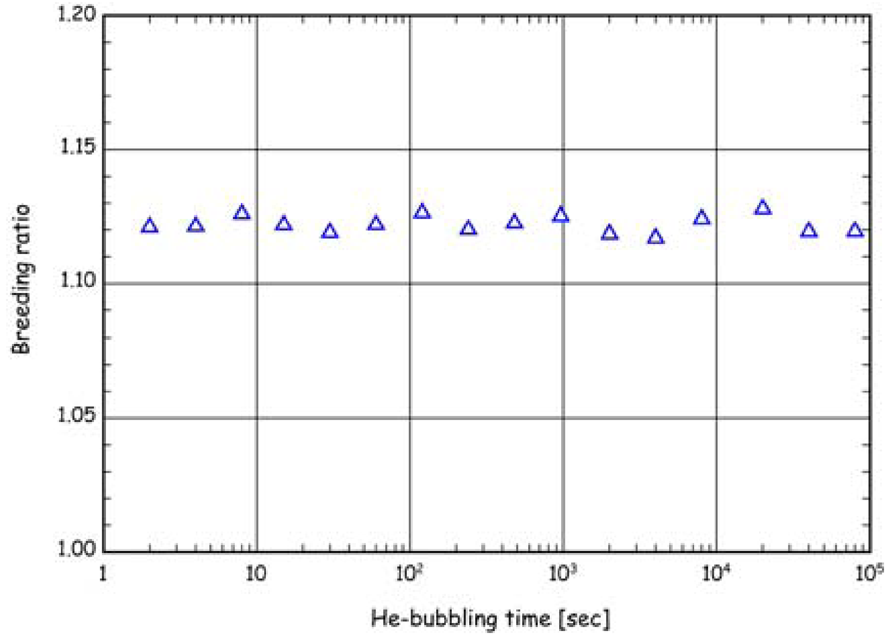
The concept of MSFR: Fuel Reprocessing

Batch reprocessing:

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) reprocessing:



Fast neutron spectrum

$\Rightarrow$ very low capture cross-sections

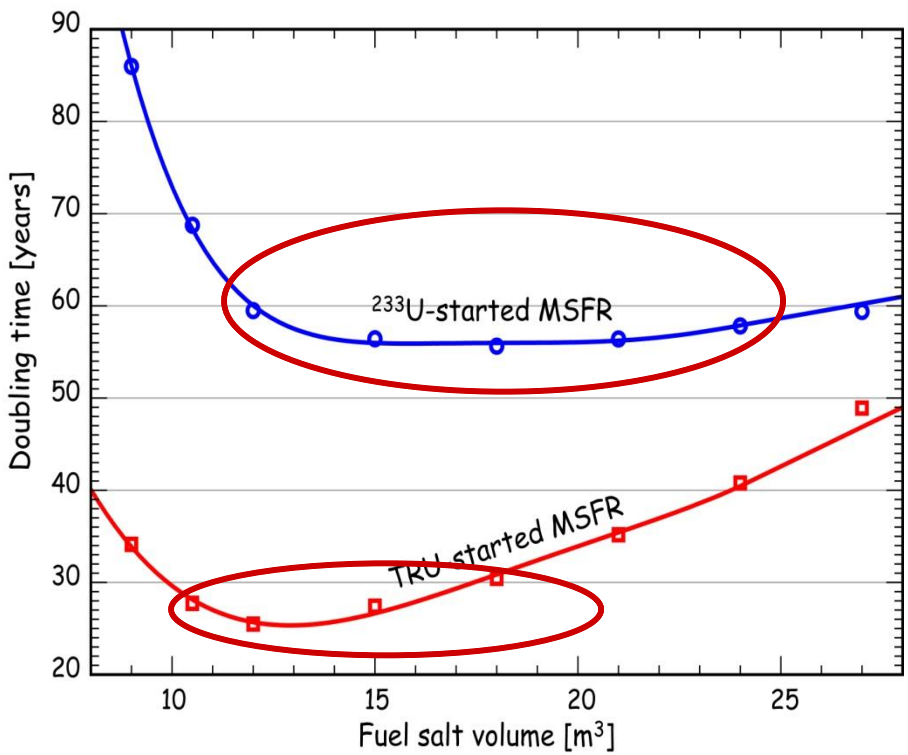
$\Rightarrow$ low impact of the FP extraction on neutronics

$\Rightarrow$ Parallel studies of chemical and neutronic issues possible

MSFR: Design and Fissile Inventory Optimization

Fuel salt volume / specific power	t(100 dpa)	t(100 ppm He)	t(-1 at% of W)
12 m^3 - 500 W/cm^3	85 years	2.2 years	4.7 years
18 m^3 – 330 W/cm^3	133 years	3.2 years	7.3 years
27 m^3 - 220 W/cm^3	211 years	5.5 years	10.9 years

Optimization = Medium Fuel Salt Volumes



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 $\Rightarrow$ 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) $\rightarrow$ *large fuel salt volume and small specific power*

- The neutronic irradiation damages to the structural materials which modify their physicochemical properties. Three effects: displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium $\rightarrow$ *large fuel salt volume and small specific power*

- The neutronic characteristics of the reactor in terms of burning efficiencies $\rightarrow$ *small fuel salt volume and large specific power* and of deployment capacities, i.e. breeding ratio (= ^{233}U production) versus fissile inventory $\rightarrow$ *optimum near 15m³ and 400W/cm³*

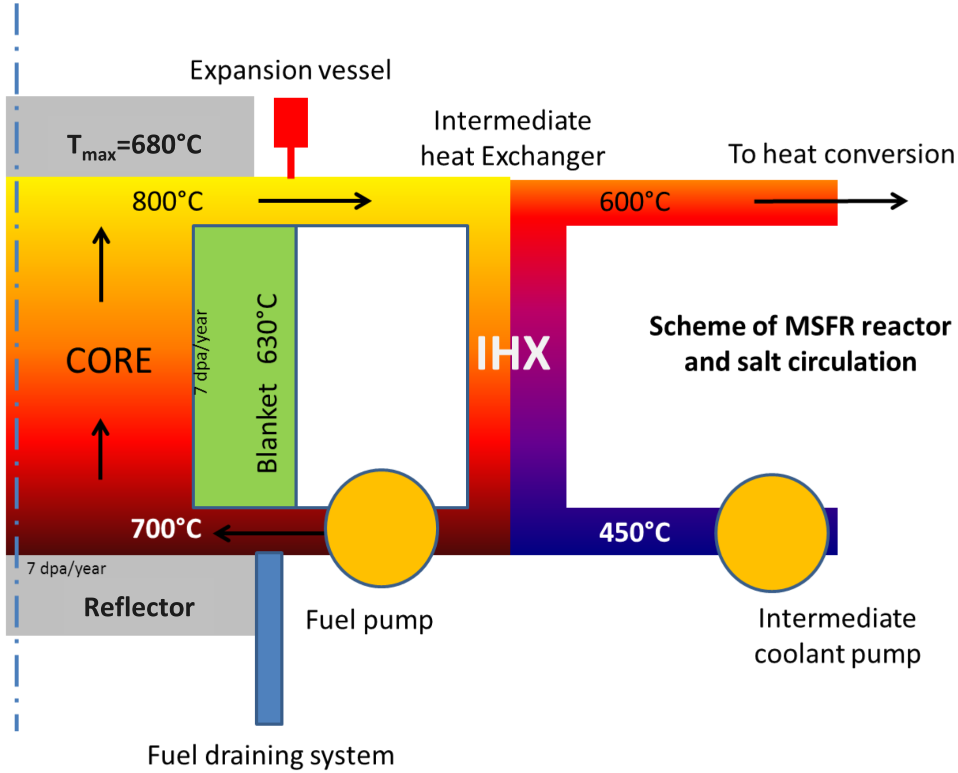
$\Rightarrow$ Reference MSFR configuration with 18m³ et 330 W/cm³ corresponding to an initial fissile inventory of 3.5 tons per GWe

4th Generation International Forum and MSFR: Availability

MSFR Availability: structural materials (Ni-based alloys) resistance

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

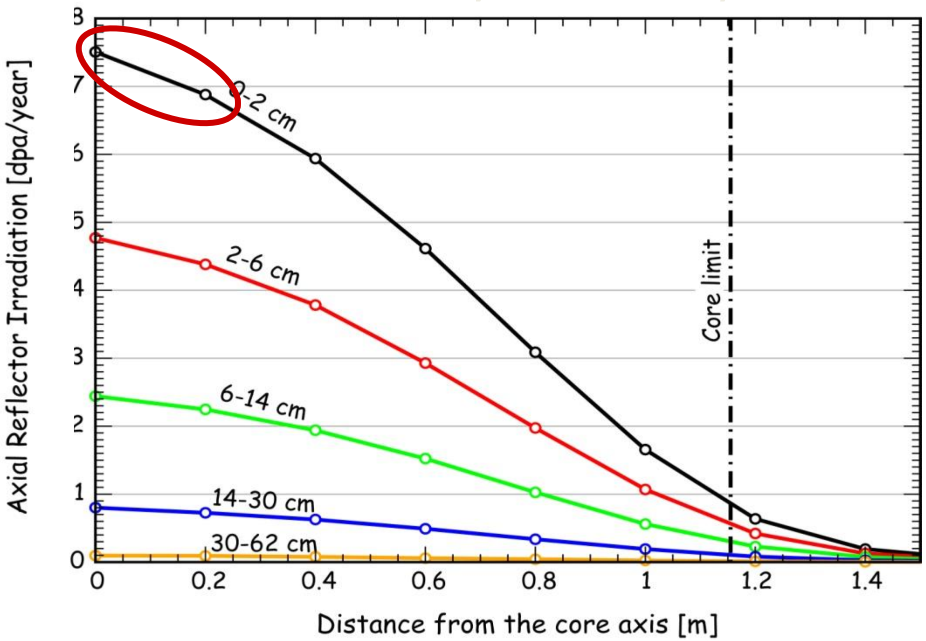
Neutronic irradiation damages to the structural materials (modify their physicochemical properties) = displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium, activation – At high temperatures



4^{th} Generation International Forum and MSFR: Availability

Displacements per atom: represent the number of times one atom is displaced for a given neutron flux

Most irradiated area (central part of axial reflector – radius 20 cm/thickness 2 cm)



+ Effects due to fissions occurring near the material wall - damages on the first tens µm

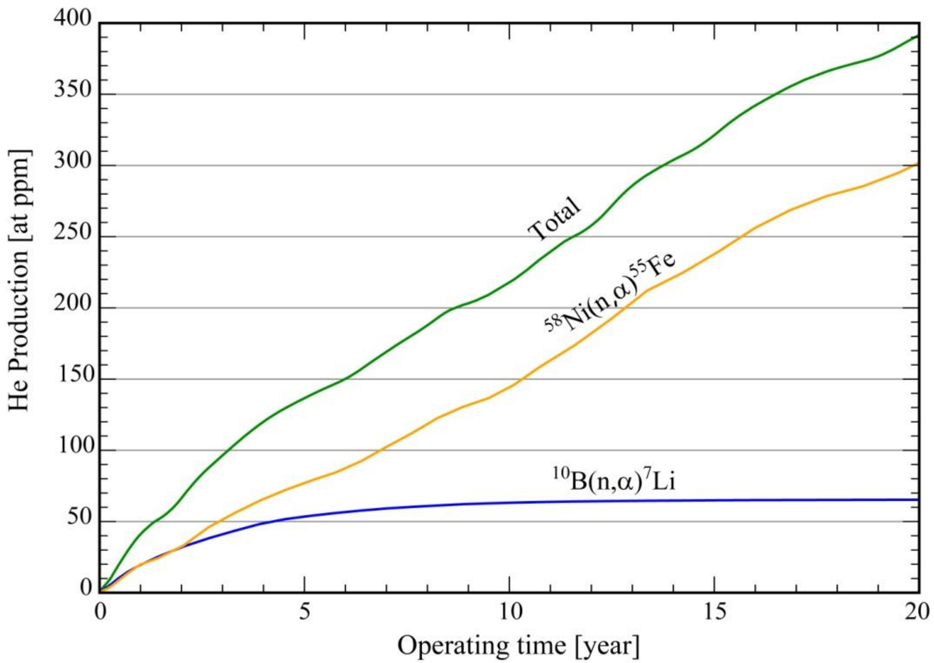
Main activated elements in structural materials

	$T_{1/2}$ [years]	[At/cm³]	Decay mode
^{59}Ni	76000	$2.97\ 10^{20}$	EC
^{63}Ni	99	$3.56\ 10^{19}$	β^- 67 keV
^{99}Tc	211300	$1.26\ 10^{19}$	β^- 294 keV
^{93}Mo	3012	$2.85\ 10^{18}$	EC +88% 31 keV
^{93}Nb	16	$1.75\ 10^{15}$	IT 31 keV
^{3}H	12	$1.23\ 10^{15}$	β^- 19 keV

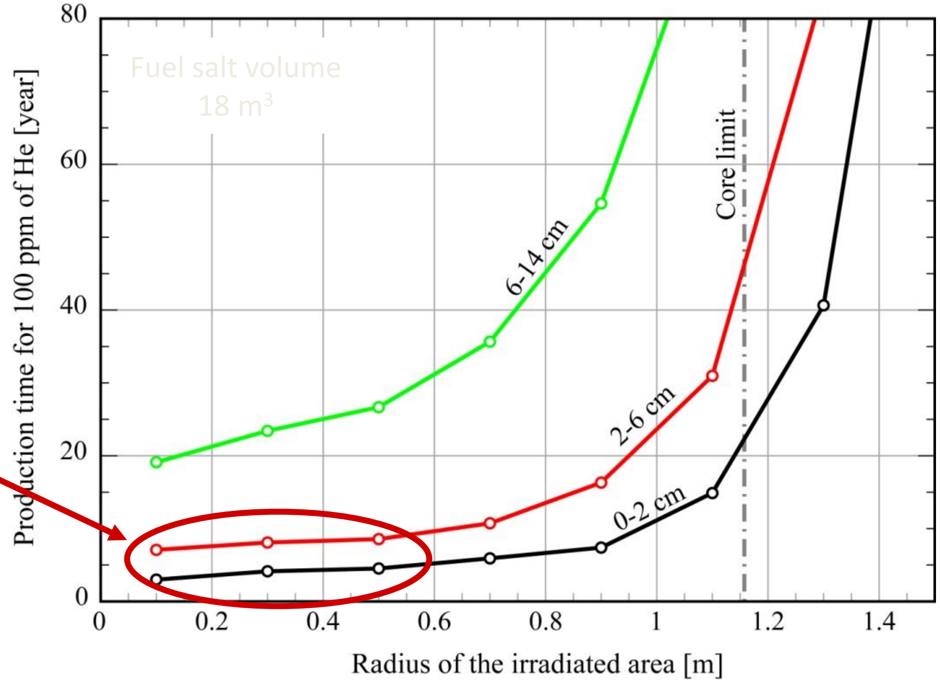
4^{th} Generation International Forum and MSFR: Availability

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



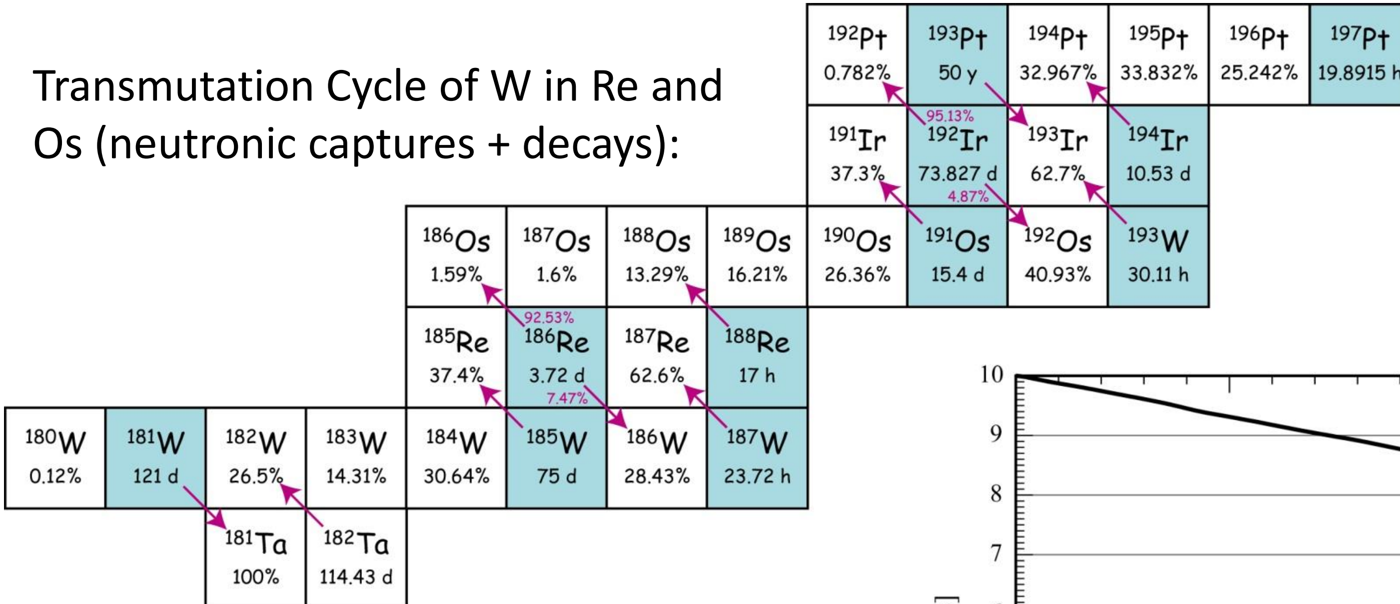
⇒ Regular replacements of these area to be planned (first 10cm only) <u>or</u> enriched Ni (lower ^{58}Ni content) <u>or</u> addition of a thin layer of another material (SiC?) to protect the surface of these reflectors

4th Generation International Forum and MSFR: Availability

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

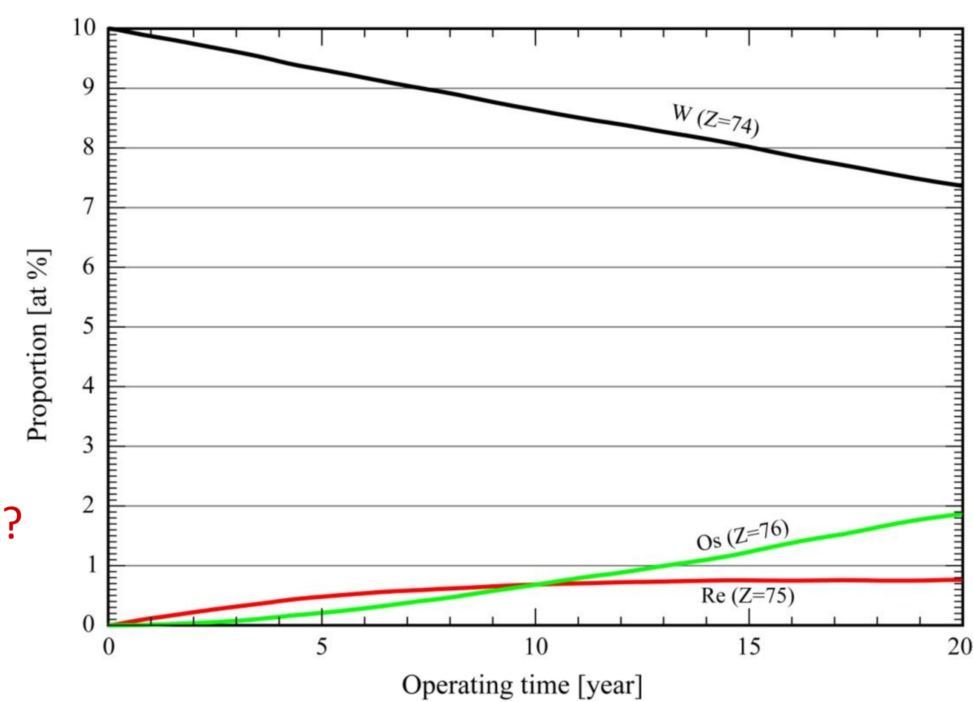
Ni	W	Cr	Mo	Fe	Ti	C	Mn	Si	Al	B	P	S
79.432	9.976	8.014	0.736	0.632	0.295	0.294	0.257	0.252	0.052	0.033	0.023	0.004

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 18 m^3:

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



4th Generation International Forum and MSFR: Availability

MSFR Availability: structural materials (Ni-based alloys) resistance

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

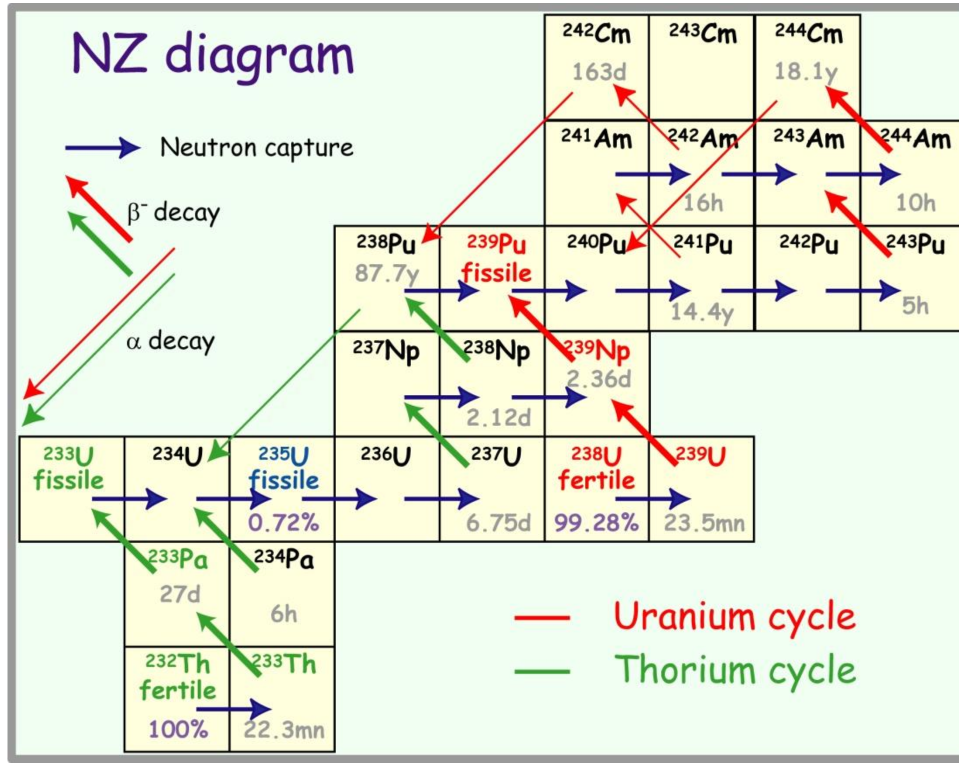
Generation4 and Th fuel cycle

Uranium cycle partially used in currently operating reactors (cf MOX fuel)

+ used in Phenix / Superphenix

+ studied in mainly Gen4 reactors

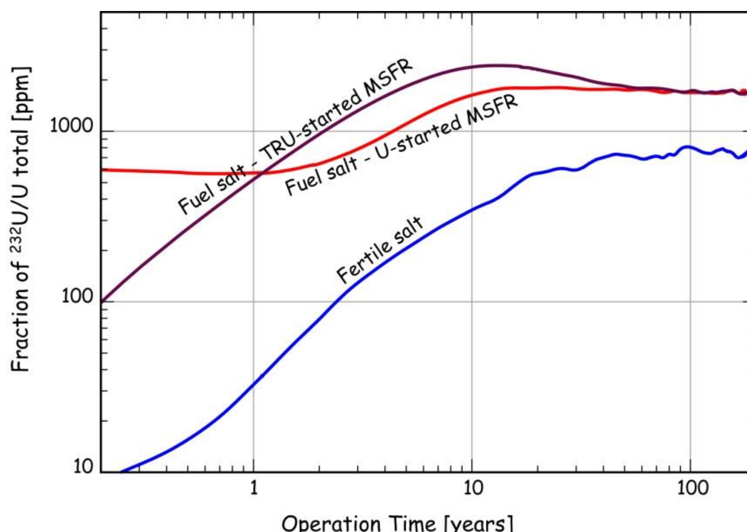
Thorium/^{233}U fuel cycle = only alternative to U/Pu fuel cycle

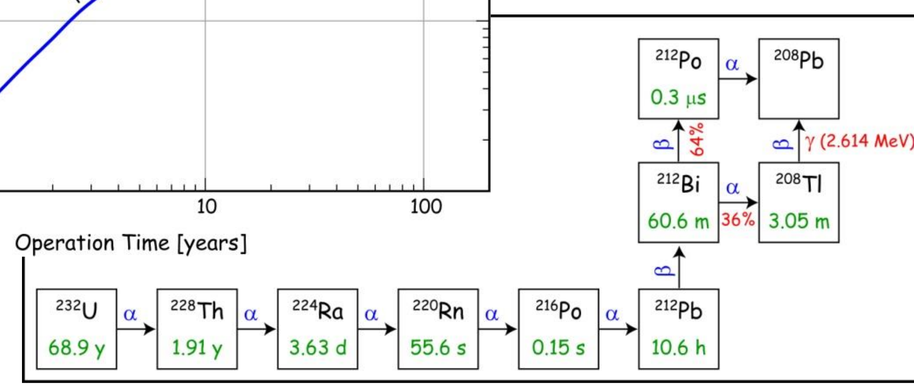


Thorium fuel cycle presents 2 essential advantages :

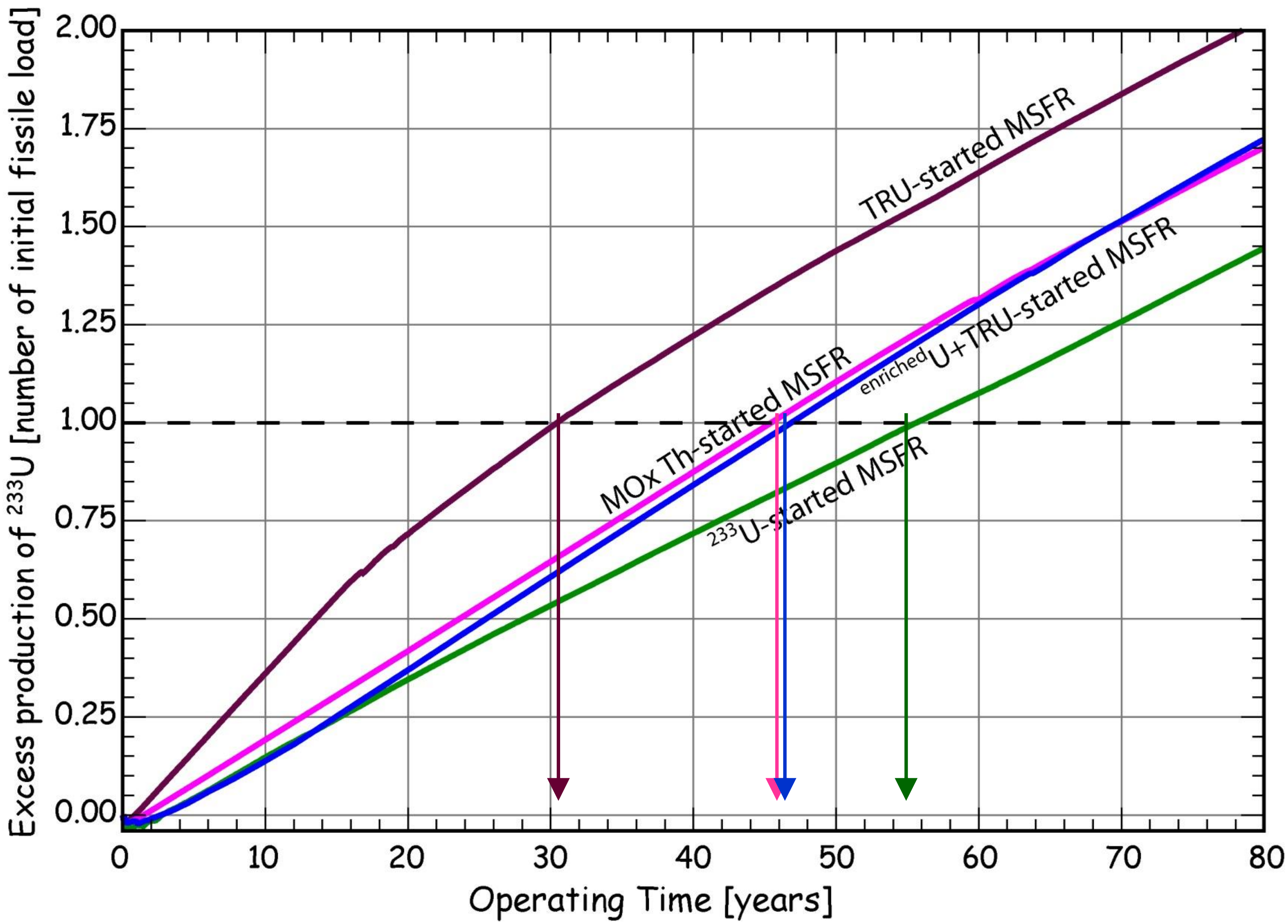
- Lower production of transuranic elements (TRU)

- High proliferation resistance thanks to the decay of ^{232}U (2.6 MeV gamma - activity of 1g of ^{232}U at equilibrium = 270 GBq) mixed with ^{233}U in the core + blanket





4^{th} Generation International Forum and MSFR: Starting Modes and Deployment Capacities



4^{th} Generation International Forum and MSFR: Starting Modes and Deployment Capacities

Deployment scenario at a French scale with a linear doubling of the installed nuclear power between 2020 and 2100 with these assumptions:

- Current PWRs stopped after 45 years of operation
  - 10% using MOX fuel (corresponding Minor Actinides vitrified)
- EPR fleet: deployed from 2014
  - From 2040: some of these EPRs loaded with MOX fuel and Thorium
- MSFR fleet deployed in 2070, using the output of MoX-Th irradiated in these EPRs
- As soon as possible (when ^{233}U available): MSFRs started with a mix of ^{233}U-PuUoX or ^{233}U-PuMoX (cf [^{233}U+TRU]-started MSFR configurations)
- First half of the XXIIth century: decision to stop the fission based electricity production (replaced by a novel technology)
  - Introduction of “incinerator MSFRs” to further reduce the heavy nuclei inventories discharged after the final shutdown of the MSFR fleet

Deployment scenarios: reduction of the final HN inventory

"Incinerator MSR" identical to MSFR except for the fuel salt composition + suppression of the fertile blanket

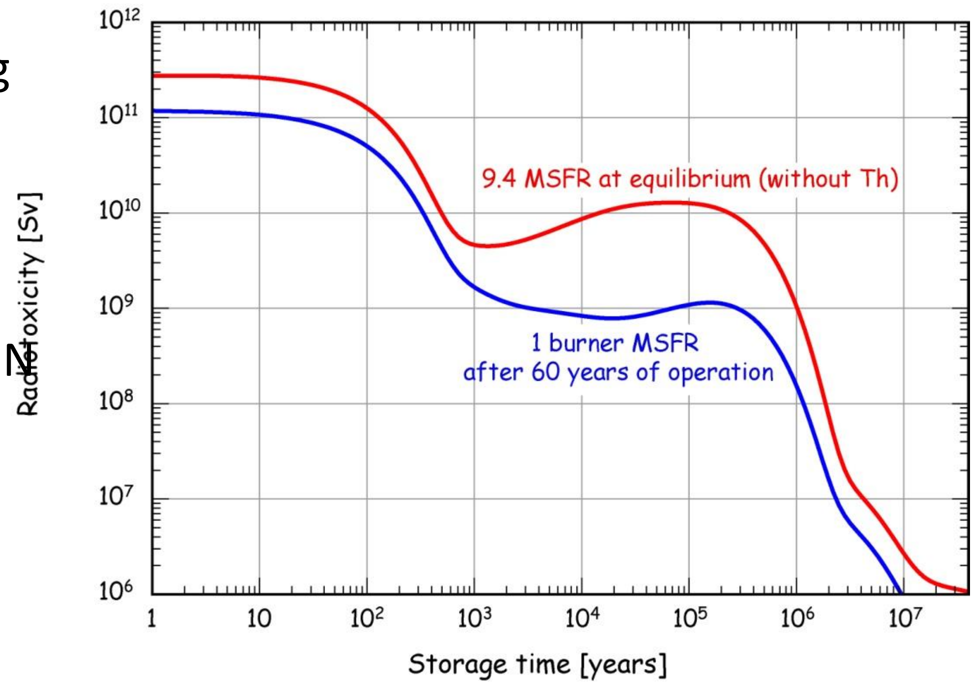
Fuel salt: FLiNaK with 46.5% ^{7}LiF, 11.5% NaF, 41.7% KF, $(HN)F_4$

- Melting point correctly low even with small HN proportion (no Th) in the salt
- Neutron spectrum not too thermalized

Incinerator operation:

- Initial HN load to reach criticality: 685 kg of transTh from MSFR
- Fueled with transTh from MSFR to maintain reactivity
- Shutdown after 60 years of operation: HN burning equivalent to 9.4 MSFR inventories

[kg]	9.4 MSFR (input)	Inventory at 60 yrs	Burning efficiency
U	72 751	6 407	11.5
Np	1 381	506	2.8
Pu	2 768	1 530	1.8
Am	72	39	1.8
Cm	33	64	0.5
HN	77 005	8 550	9.1



4th Generation International Forum and MSFR:
Deployment scenarios of the Th fuel cycle with MSFRs

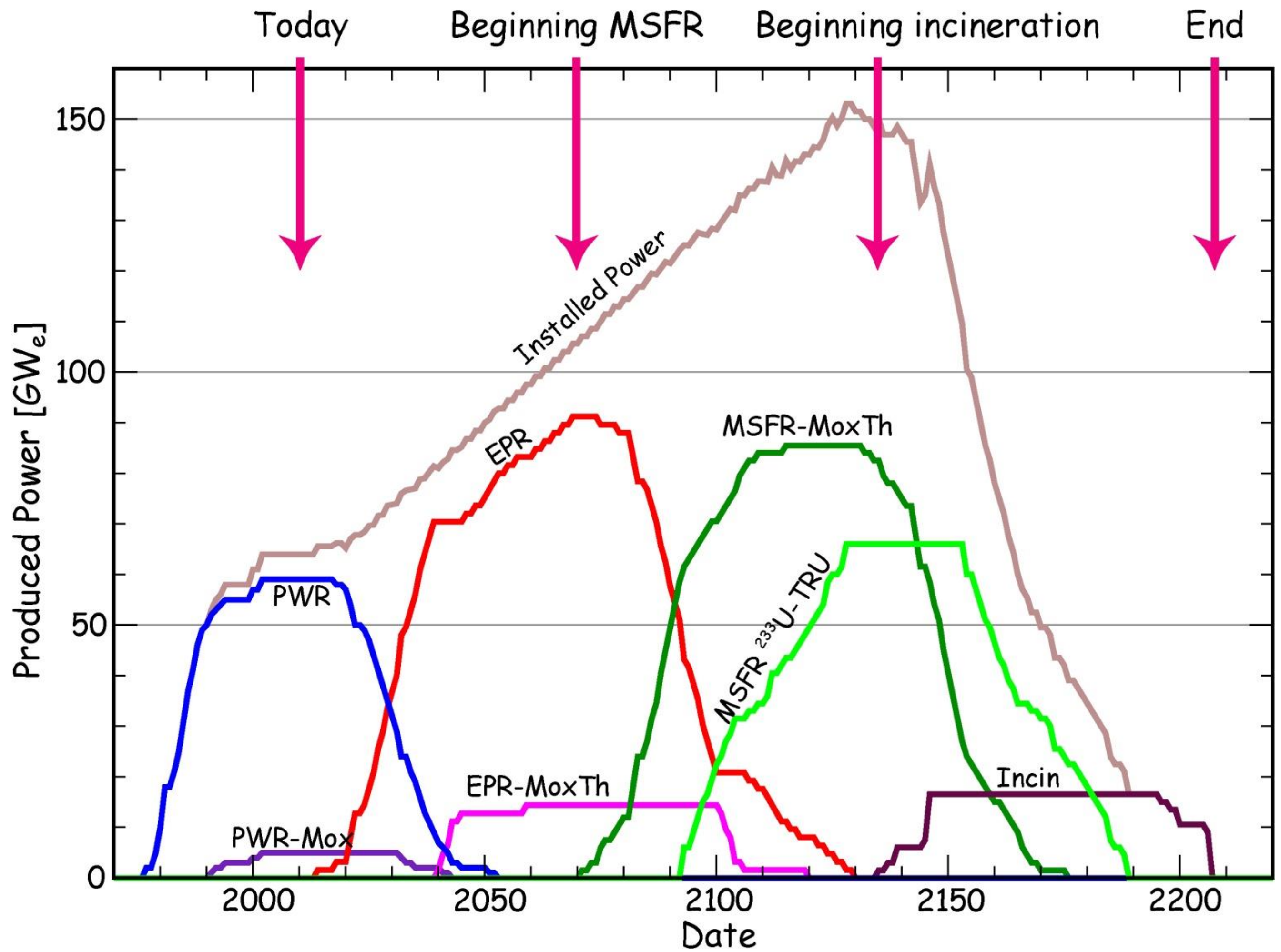
Total power produced = 138 000 TWh

among which 72 300 TWh by the MSFR fleet

Very good deployment capacities -

Transition to the Thorium fuel cycle achieved

+ Close the current fuel cycle (reduce the stockpiles of produced transuranic elements)



4th Generation International Forum and MSFR:
Deployment scenarios of the Th fuel cycle with MSFRs

- Stockpiles of uranium from reprocessing largely reduced

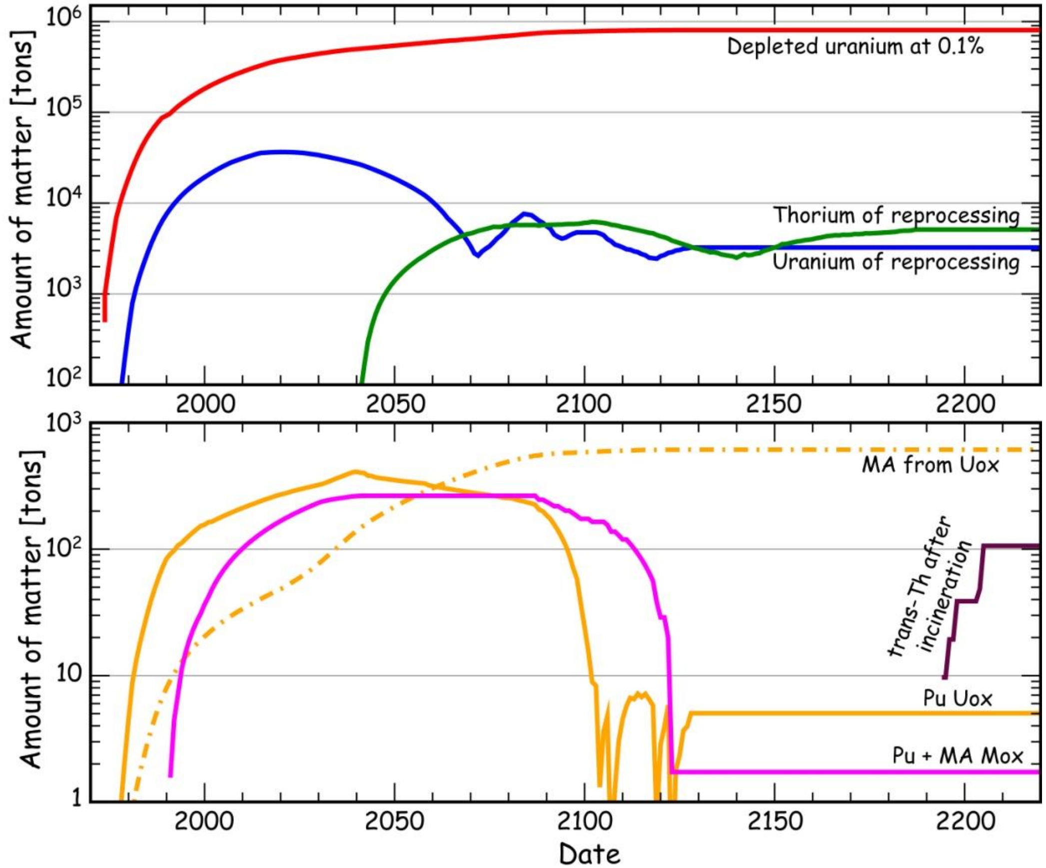
- Stockpiles of Pu-Uox, Pu-Mox and AM-Mox totally burned in MSFR $\Rightarrow$ remains only MA extracted from Uox fuel when using Pu-Mox in PWRs and EPRs

- After incinerator MSFRs: only 100 tons of transthorian elements remaining

-Around 18 000 t of actinides used for fission (138 000 TWh

- 11 700 t from natural U
- 6 300 t from Th

- Natural resources needed for this nuclear deployment:
  - 821 400 t of natural U
  - 11 600 t of Th



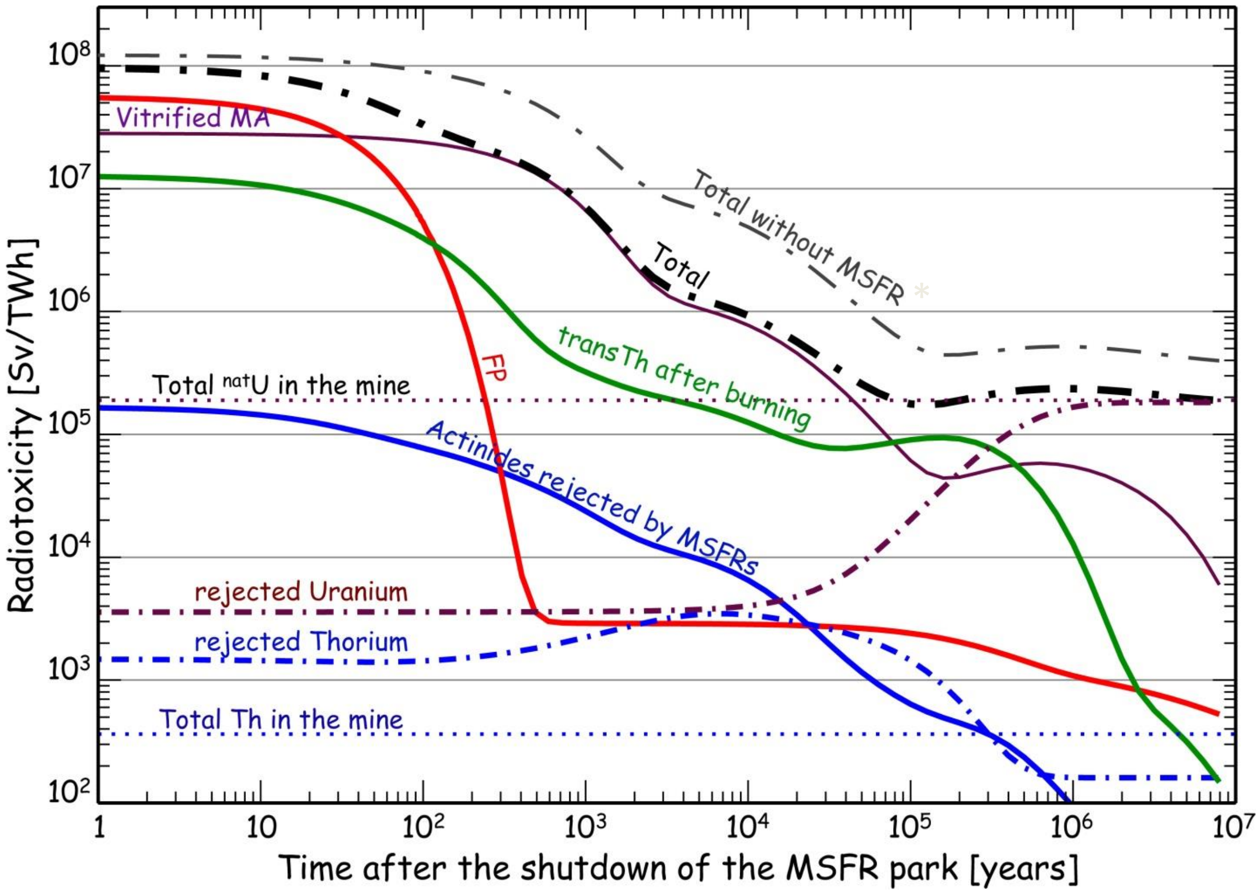
4th Generation International Forum and MSFR:
Deployment scenarios of the Th fuel cycle with MSFRs

⇒ Scenario optimized but without MSFR and the Th fuel cycle: radiotoxicity 3 to 5 times higher between 1000 and 100 000 years

- Long term radiotoxicity dominated by the vitrified MA from Uox fuel mixed with the FPs (Gen2 and Gen3 reactors)

- Very long term radio-toxicity (after 300 000 years) dominated by the rejected uranium (depleted + reproc.) – see long life decay products of ^{238}U (as ^{230}Th and ^{234}U)

- Radiotoxicity of the transthorian elements from the MSFR fleet (final inventories) lower than the extracted natural U after 3 000 years



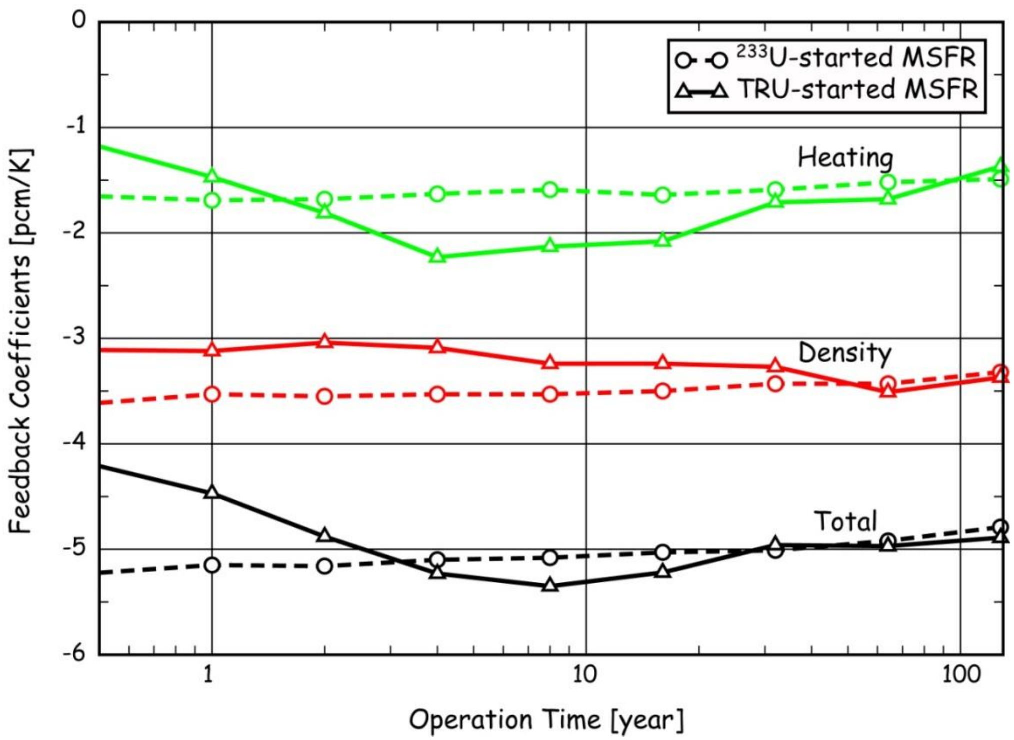
* Based on a production with PWRs and EPRs of 65 700 TWh minimizing the actinides stockpiles

3- Safety parameters: Feedback coefficients

dk/dT = Variation of the multiplication factor (dk) with the core temperature (dT)

Reactor intrinsically safe if dk/dT < 0 (if T ↗ then k ↘)

$$\left.\frac{dk}{dT}\right)_{\text{Total}} = \left[\left.\frac{dk}{dT}\right)_{\text{Salt Heating}} + \left.\frac{dk}{dT}\right)_{\text{Salt Density}} + \text{correlations} \right] < 0$$



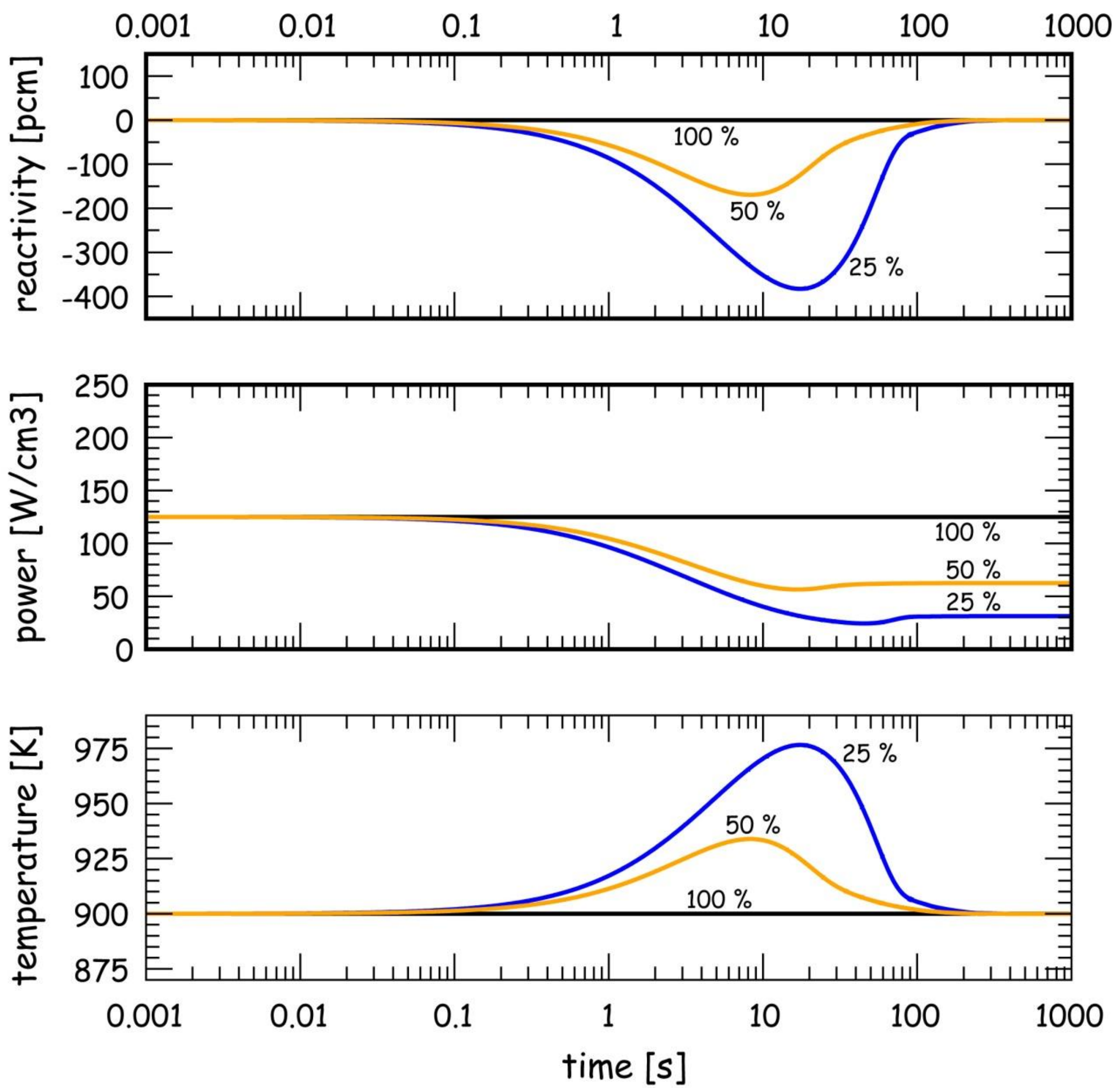
⇒ dk/dT largely < 0 for all MSFR configurations and equal to -5 pcm/K for the reference configuration

+ Salt density coefficient (equivalent to void coefficient) < 0 for all configurations too

⇒ MSFR: Only Gen4 system being both breeder and with all negative safety coefficients

4th Generation International Forum and MSFR

Control of the chain reaction & Power demand regulation



Power demand decrease

⇩

Rise of temperature
+
Drop of reactivity

⇩

Return to equilibrium at the nominal temperature
+
Decrease of the produced power

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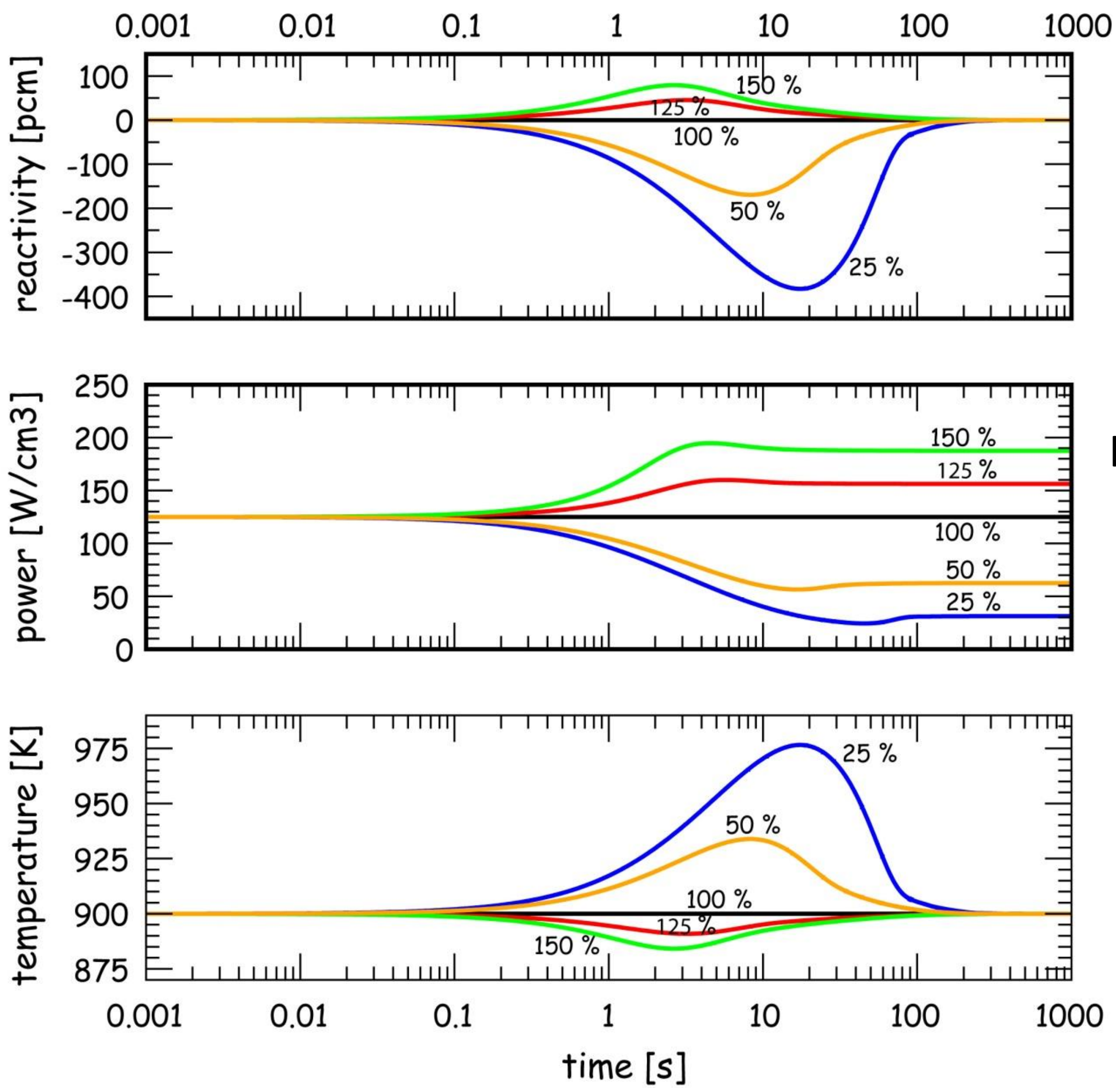
4th Generation International Forum and MSFR
Control of the chain reaction & Power demand regulation



Increase of the power demand

⇩

Drop of temperature
+ Increase of reactivity

⇩

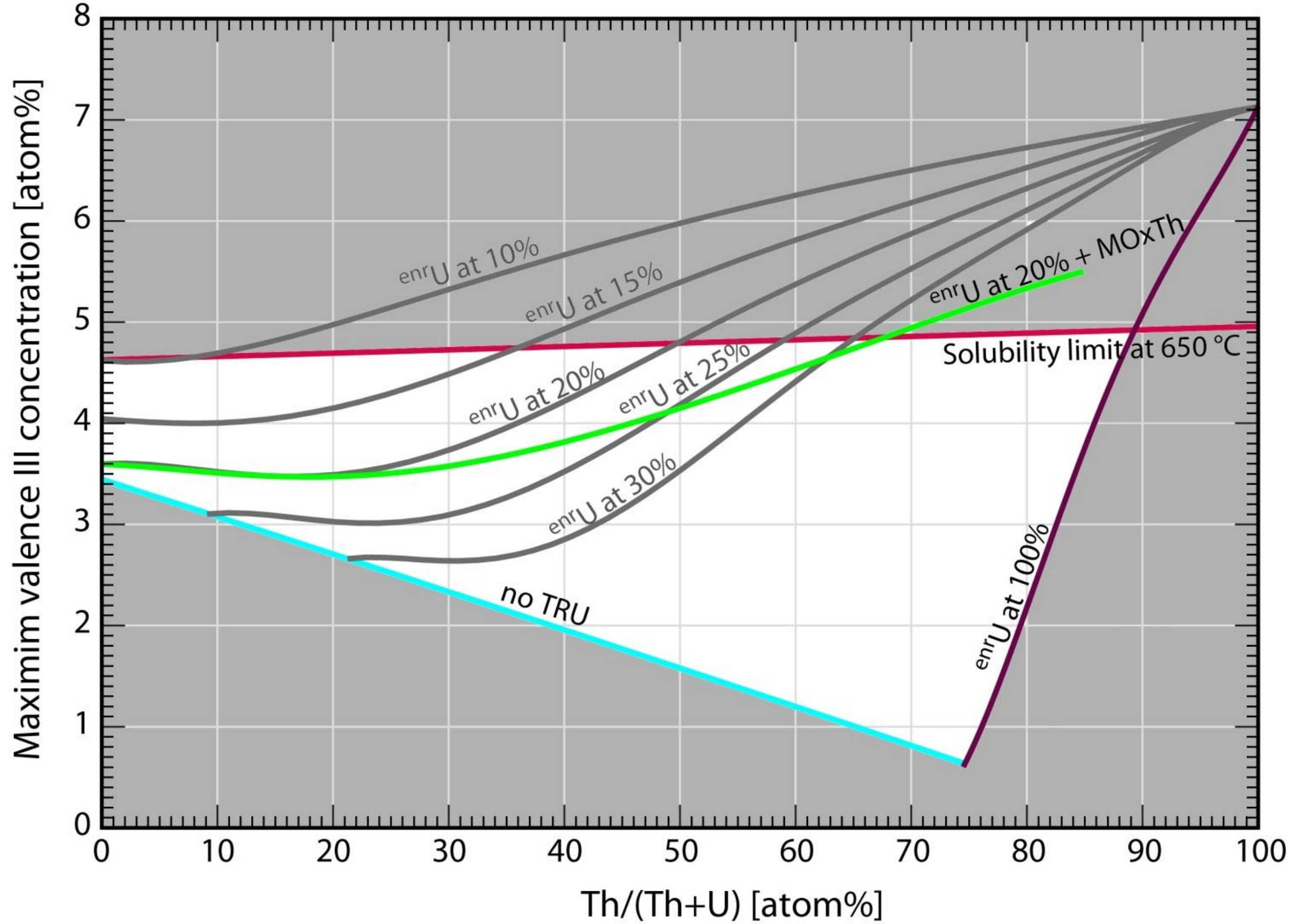
Increase of the produced power

⇩

MSFR driven by the extracted power, and thus by the energy demand through the secondary circuit

Control rods not mandatory
(Cf. reactivity insertion source)

Power Demonstrator of the MSFR: initial fissile load



- ✓ enrichedU 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%

From Power Demonstrator of the MSFR to SMR

	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)

From Power Demonstrator of the MSFR to SMR

	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

From Power Demonstrator of the MSFR to SMR

	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	Radial blanket and H/D=1.5	Radial blanket and H/D=1.5
Power [MW_{th}]	**100**	**200**	**100**	**200**	**100**	**200**
Initial ^{233}U load [kg]	654	654	667	667	677	677
Fuel reprocessing of 1l/day						
Feeding in ^{233}U [kg/an]	11.38	23.38	1.72	4.70	-0.07	0.98
Breeding ratio	**-29.83%**	**-30.64%**	**-4.52%**	**-6.16%**	***0.18%***	**-1.29%**
Total ^{233}U needed [kg]	1013.87	1388.37	738.83	835.16	715.05	754.25
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	-0.38	-0.26
Breeding ratio	**-29.37%**	**-29.59%**	**-3.88%**	**-4.69%**	***1.00%***	***0.34%***
Total ^{233}U needed [kg]	1001.86	1353.13	722.50	794.21	709.74	723.03
Breeding ratio (radial + axial fertile blankets)			***2.49%***	***1.54%***		

Addition of a radial fertile blanket + Elongated core ⇨ small modular breeder MSFR

Thermo-hydraulic model

The control equations for the liquid-fuel in the COUPLE code are written as following:

Mass conversation equation: $$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho \vec{u}) = 0$$

Momentum conversation equation: $$\frac{\partial \rho u_i}{\partial t} + \nabla \cdot (\rho \vec{u} u_i) = -\nabla \cdot \tau_i$$

Energy conversation equation: $$\frac{\partial \rho T}{\partial t} + \nabla \cdot (\rho \vec{u} T) = \nabla \cdot \left(\frac{\lambda}{c_p} \nabla T\right) + \frac{S}{c_p}$$

<u>See the previous presentation :</u>
ZHANG D., ZHAI Z.-G., CHEN X.-N., WANG S., RINEISKI A., “COUPLE, a coupled neutronics and thermal-hydraulics code for transient analyses of molten salt reactors”

Neutronics model

- based on the multi-group (here 2) diffusion theory while considering flow effects of the liquid-fuel

Diffusion equation for the neutron flux of group g:

[illegible]

The balance equation for the delayed neutron precursor of family i:

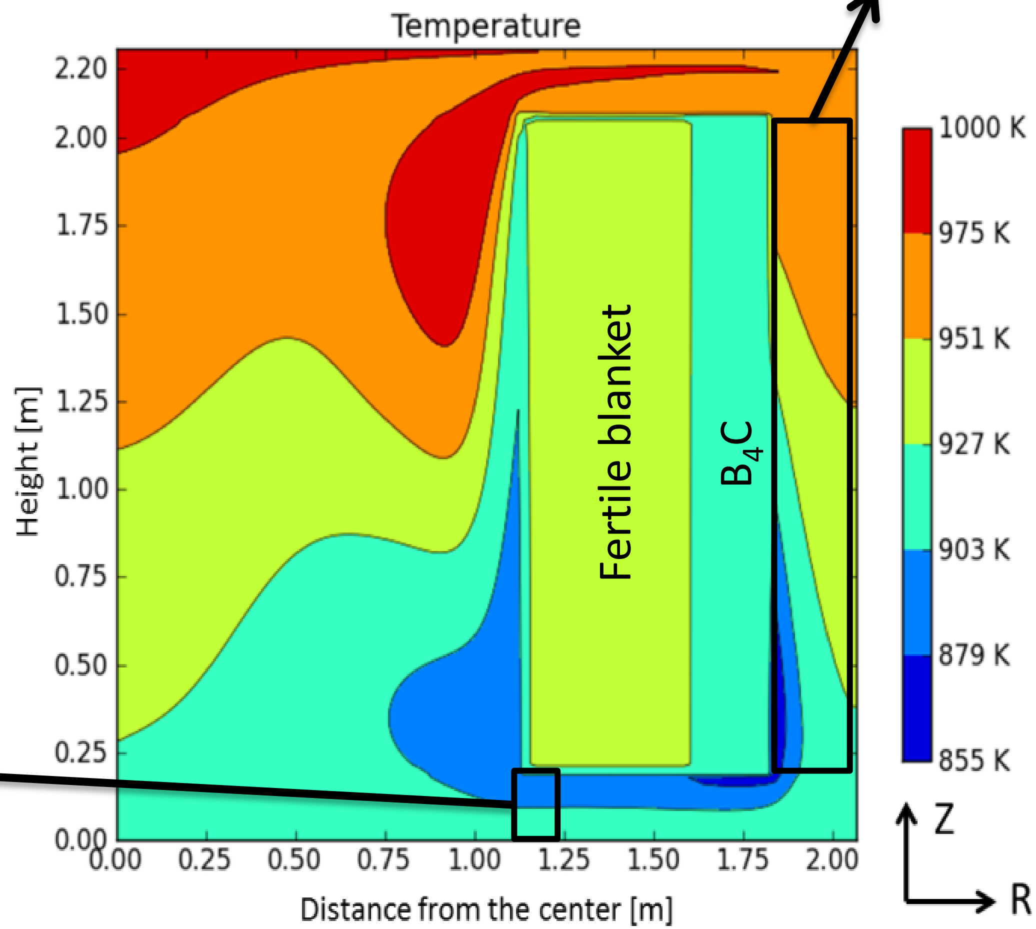
[illegible]

MSFR model

Steady state calculation

- Half of the core model
- with 112/130 cells in the R/Z directions

Heat exchanger model:
Negative heat source

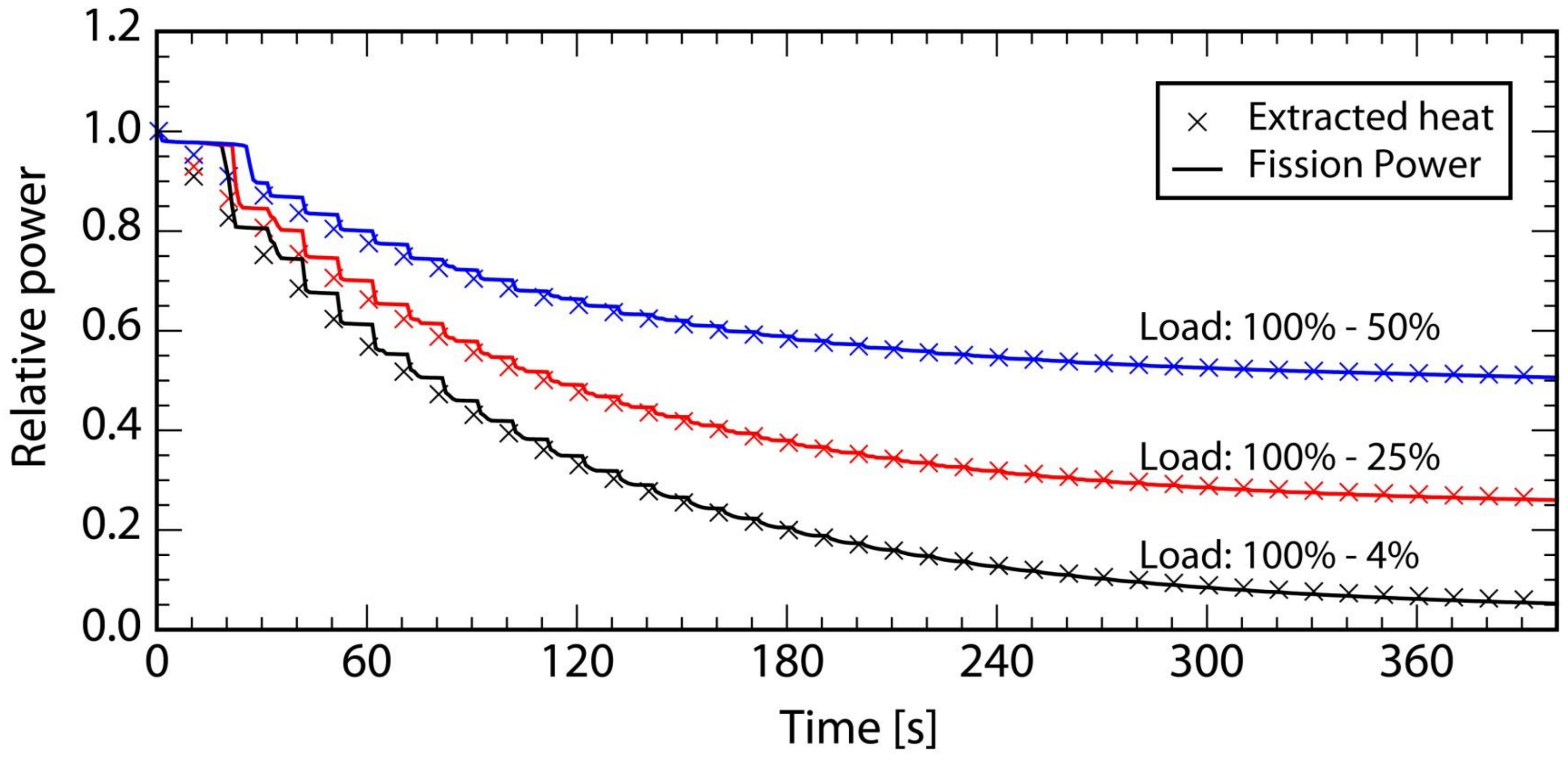


Pump model:
injection velocity
profile adjusted to
avoid recirculation

Operational transients

Load following

- negative heat source in the heat exchanger decreases from 100% to 50/25/4% exponentially (stepwise) with τ=100s

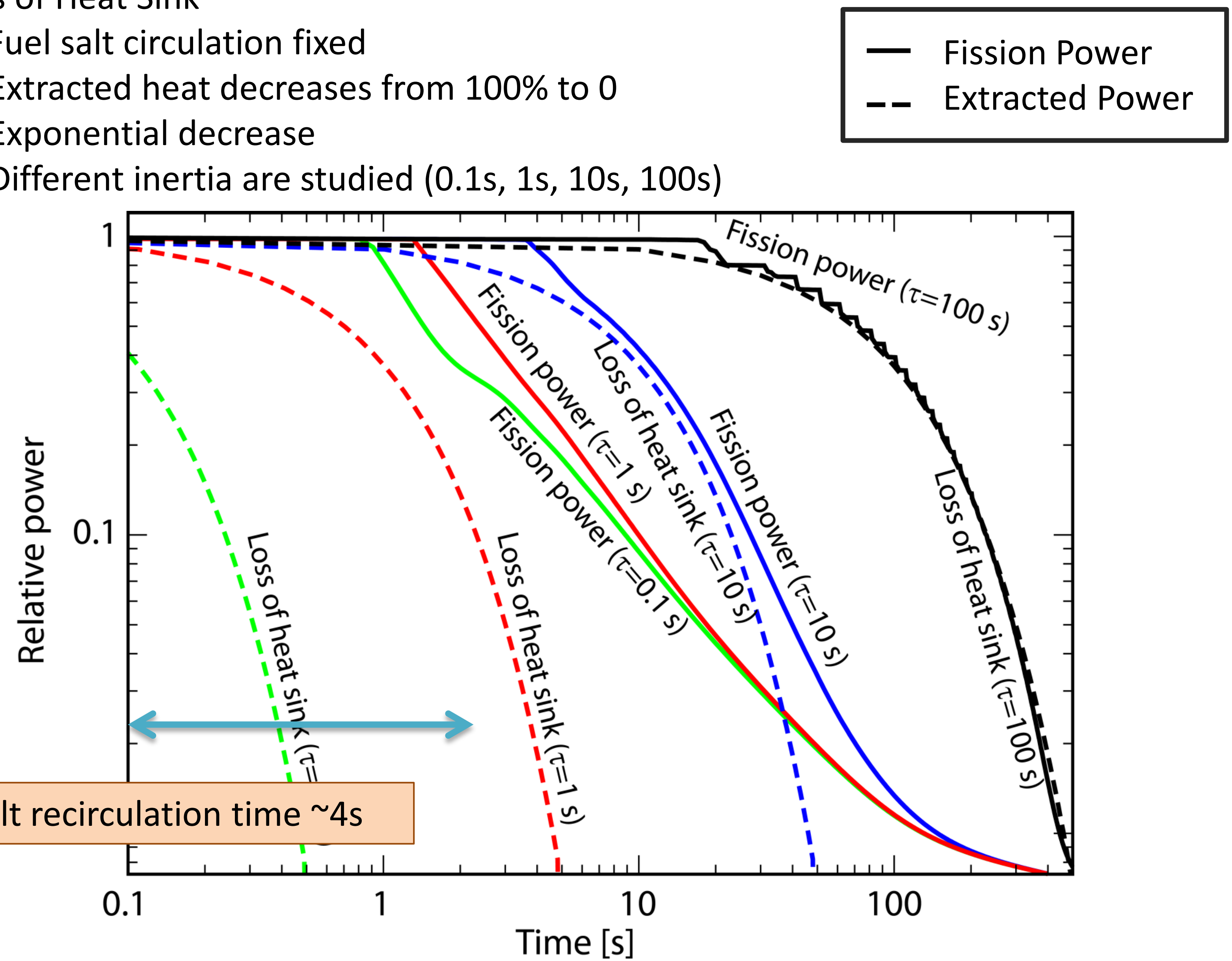


⇨ Fission power follows rapidly the extracted power

Accidental transients

Loss of Heat Sink

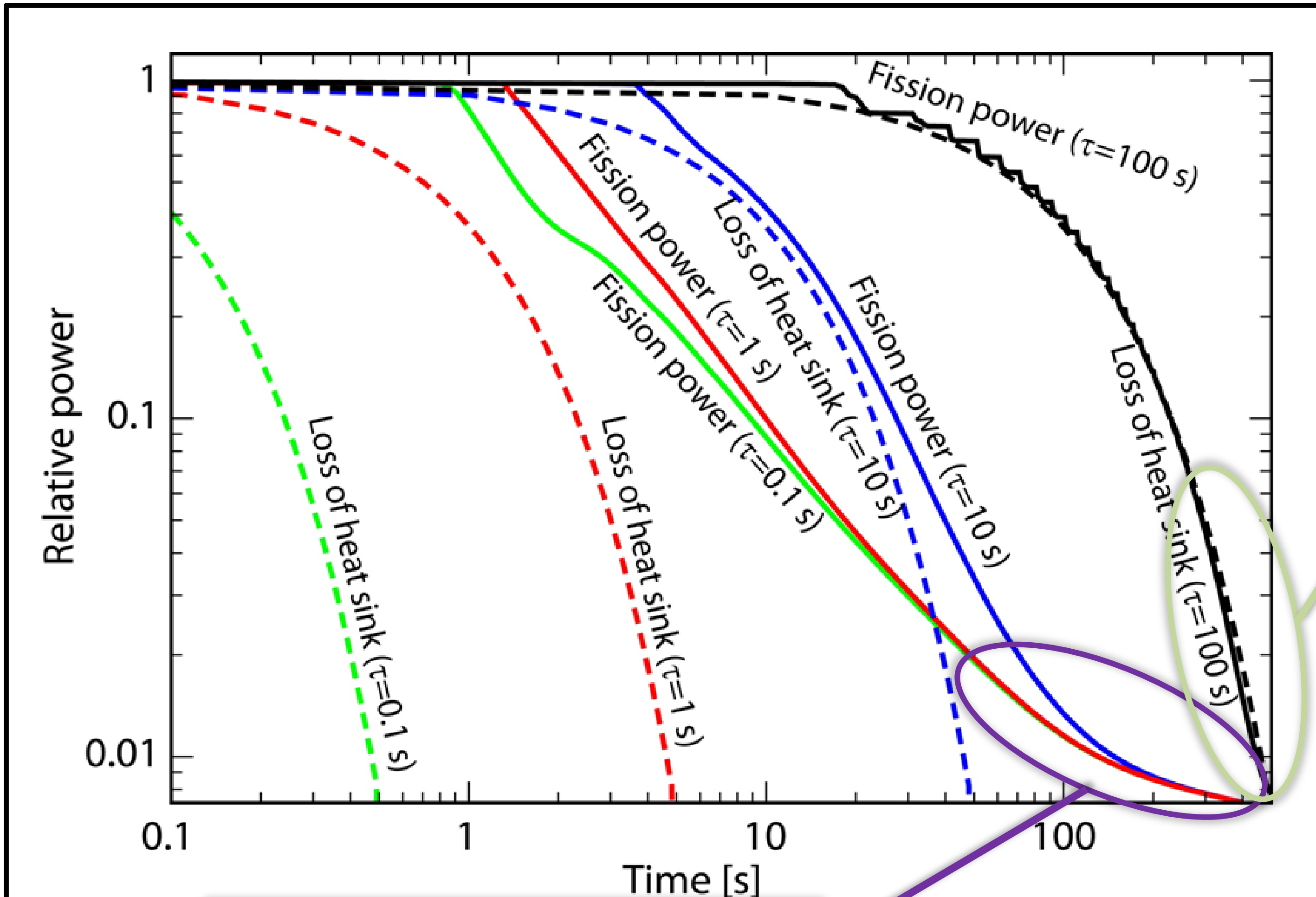
- Fuel salt circulation fixed
- Extracted heat decreases from 100% to 0
- Exponential decrease
- Different inertia are studied (0.1s, 1s, 10s, 100s)



Salt recirculation time ~4s

Accidental transients

Loss of Heat Sink



$P_{fission} < P_{ext}$ because $P_{core} = P_{fission} +$ Decay Heat

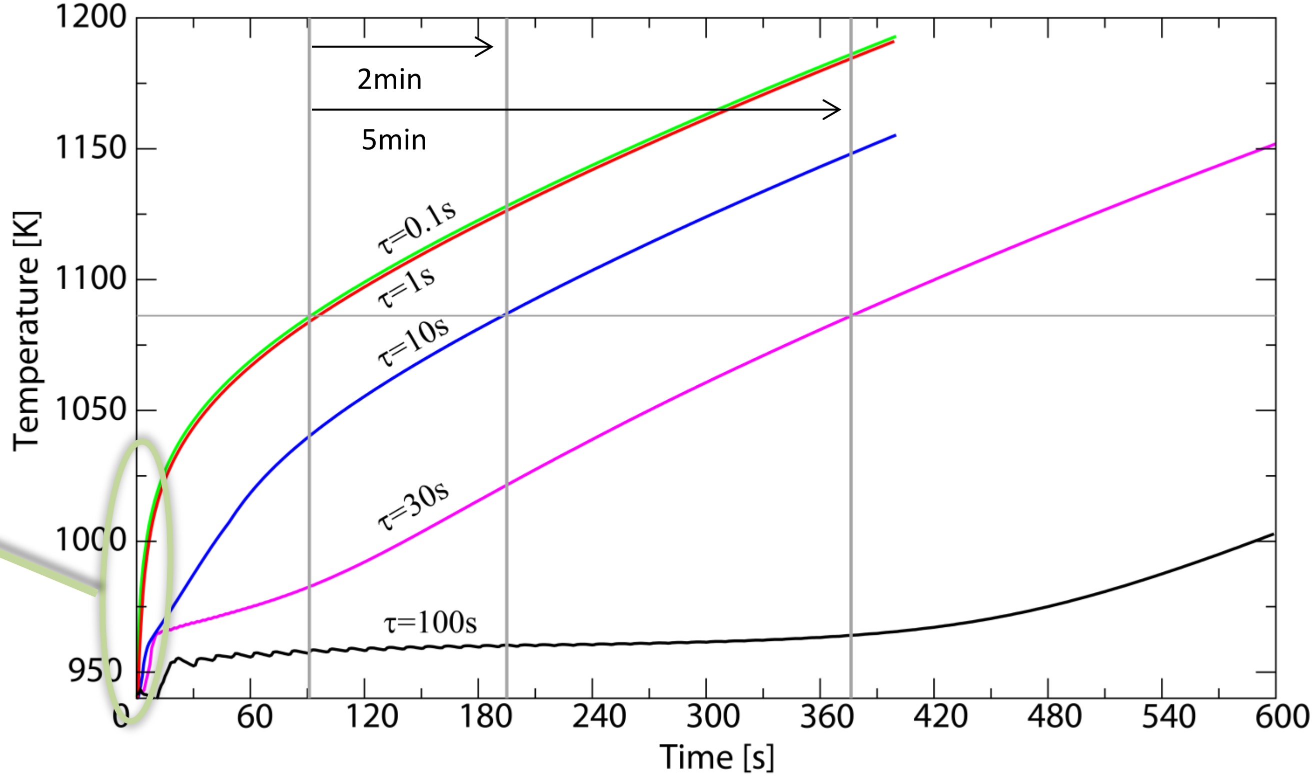
Minimal fission power due to delayed neutrons precursors

Accidental transients

Loss of Heat Sink

- Temperature increase caused by not extracted fission power + decay heat
- **Very pessimistic hypothesis** of extracted heat = 0 (heat losses through structure material, natural circulation of the fuel salt and intermediate fluid …)

Fast temperature increase



- Inertia of τ = 10s delays the global temperature increase of 2min and avoids the fast temperature increase
- Inertia of τ > 10s should be implemented on the pumps of the fuel circuit and the intermediate circuit

Accidental transients

How to manage this temperature increase?

⇨ Protection systems in the fuel salt circuit studied (redondant safety cooling system or natural convection)

⇨ Main safety system = **draining of the fuel salt**

⇨ Active and passive systems will be implemented on the bottom of the fuel circuit to allow draining by gravity