Concept of Molten Salt Fast Reactor

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
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

**DPA**

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

**He production**

Most irradiated area (central part of axial reflector – radius 20 cm/thickness 2 cm)
## Liquid-fueled reactors: fluoride or chloride?

### Elements produced

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

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**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?

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Lithium fluorides fulfill all constraints

Molten Salt Reactors

Thorium / $^{233}\text{U}$ Fuel Cycle

Molten Salt Reactors

Molten Salt Reactors

## Choice of the fluoride salt:
- Waste considerations (production of $^{36}\text{Cl}$ 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

![Neutron Energy vs Flux](image)

![Neutron Energy vs Neutron Cross-Section](image)
Journées MSR de Massy, 2018

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
  - $^{235}$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

- core volume adjusted to keep the same salt volume -

Présentation MSFR
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
• 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
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- Possibility to reprocess the fuel without stopping the reactor:

Neutronic Optimization of MSR (Gen4 criteria):

- Safety: negative feedback coefficients
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- Deployment: good breeding of the fuel + reduced initial fissile inventory

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 provision of SSC to shift the R&D orientations and objectives to the original Generation IV Roadmap issued in 2002, in order to encompass 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.

Journées MSR de Massy, 2018
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

![Diagram of MSFR system with labels for pump, heat exchangers, and fuel.]
Description of the Molten Salt Fast Reactor (MSFR) system

General characteristics:
- Liquid circulating fuel
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- Fast neutron spectrum
- Thorium fuel cycle

Three circuits:
- Fuel salt circuit
- Intermediate circuit
- Thermal conversion circuit
- Draining / storage tanks
- Processing units

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)

MSFR and the European project EVOL


**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), IHTHE (Ekaterburg), VNIKHT (Moscow) et MUCATEX (Moscow)
Concept of MSFR: Fuel processing

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

### Concept of MSFR: Fuel processing

#### Batch chemical processing:

<table>
<thead>
<tr>
<th>Element</th>
<th>Absorption (per fission neutron)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heavy Nuclei</td>
<td>0.9</td>
</tr>
<tr>
<td>Alkalines</td>
<td>&lt; $10^{-4}$</td>
</tr>
<tr>
<td>Metals</td>
<td>0.0014</td>
</tr>
<tr>
<td>Lanthanides</td>
<td>0.006</td>
</tr>
<tr>
<td><strong>Total FPs</strong></td>
<td><strong>0.0075</strong></td>
</tr>
</tbody>
</table>

#### On-line (bubbling) processing:

- **Fast neutron spectrum**
  - $\Rightarrow$ very low capture cross-sections

- $\Rightarrow$ low impact of the processing (chemical and bubbling) on neutronics
- $\Rightarrow$ Parallel studies of chemical and neutronic issues possible
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.

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

**233U-started MSFR**

<table>
<thead>
<tr>
<th>Operation time [years]</th>
<th>Feedback coefficient [pcm/K]</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.05</td>
<td>-0.5</td>
</tr>
<tr>
<td>0.5</td>
<td>-1.5</td>
</tr>
<tr>
<td>5</td>
<td>-2.5</td>
</tr>
<tr>
<td>50</td>
<td>-3.5</td>
</tr>
</tbody>
</table>

**TRU-started MSFR**

- **database**: ENDF-B6

**PhD thesis of Mariya Brovchenko**

**Operation time [years]**

- **233U**
- **POLIMI**
- **KI Density**
- **KI Doppler**
- **LPSC Density**
- **LPSC Doppler**
- **POLITO**
- **Density TU Delft**

**Excellent (largely negative) feedback coefficients**,

Database: ENDF-B6

**Fuel Salt Inventory [kg]**

- **TRU**
- **Pu**
- **Am**
- **Cm**
- **U**

**Operation time [years]**

- **233U**

**Density**

- **POLIMI**
- **POLITO**
- **Density TU Delft**
"Reference MSFR" neutronic characteristics: from EVOL to SAMOFAR

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

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal/electric power</td>
<td>3000 MWth / ~1300 MWe</td>
</tr>
<tr>
<td>Fuel salt temperature rise in the core (°C)</td>
<td>100</td>
</tr>
<tr>
<td>Fuel molten salt - Initial composition</td>
<td>LiF-ThF$_4$-$^{233}$UF$_4$ or LiF-ThF$_4$-$^{enr}$UF$_4$-(Pu-MA)F$_3$ with 77.5 mol% LiF</td>
</tr>
<tr>
<td>Fuel salt melting point (°C)</td>
<td>585</td>
</tr>
<tr>
<td>Mean fuel salt temperature (°C)</td>
<td>725</td>
</tr>
<tr>
<td>Fuel salt density (g/cm$^3$)</td>
<td>4.1</td>
</tr>
<tr>
<td>Fuel salt dilation coefficient (g.cm$^{-3}$/°C)</td>
<td>8.82 $10^{-4}$</td>
</tr>
<tr>
<td>Fertile blanket salt - Initial composition (mol%)</td>
<td>LiF-ThF$_4$ (77.5%-22.5%)</td>
</tr>
<tr>
<td>Breeding ratio (steady-state)</td>
<td>1.1</td>
</tr>
<tr>
<td>Total feedback coefficient (pcm/°C)</td>
<td>-8</td>
</tr>
<tr>
<td>Toroidal core dimensions (m)</td>
<td>Radius: 1.06 to 1.41 Height: 1.6 to 2.26</td>
</tr>
<tr>
<td>Fuel salt volume (m$^3$)</td>
<td>18 (1/2 in the core)</td>
</tr>
<tr>
<td>Total fuel salt cycle in the fuel circuit</td>
<td>3.9 s</td>
</tr>
<tr>
<td>Intermediate fluid</td>
<td>fluoroborate (8NaF-92NaBF$_4$), FLiNaK, LiF-ZrF$_4$, FLiBe</td>
</tr>
<tr>
<td>Structural materials</td>
<td>Ni-based alloy (Hastelloy-N)</td>
</tr>
</tbody>
</table>
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, CIR TEN (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"

<table>
<thead>
<tr>
<th>Del. n°</th>
<th>Deliverable title</th>
<th>Lead benef.</th>
<th>Delivery date</th>
</tr>
</thead>
<tbody>
<tr>
<td>D1.1</td>
<td>Description of initial reference design and identification of safety aspects</td>
<td>CNRS</td>
<td>Month 6</td>
</tr>
<tr>
<td>D1.2</td>
<td>Identifying safety related physico-chemical and material data</td>
<td>JRC</td>
<td>Month 6</td>
</tr>
<tr>
<td>D1.3</td>
<td>Development of a power plant simulator</td>
<td>CNRS</td>
<td>Month 24</td>
</tr>
<tr>
<td>D1.4</td>
<td>Safety issues of normal operation conditions, including start, shut-down and load-following</td>
<td>CIRTEN</td>
<td>Month 30</td>
</tr>
<tr>
<td>D1.5</td>
<td>Development on an integral safety assessment methodology for MSR</td>
<td>IRSN</td>
<td>Month 36</td>
</tr>
<tr>
<td>D1.6</td>
<td>Identification of risks and phenomena involved, identification of accident initiators and accident scenarios</td>
<td>CIRTEN</td>
<td>Month 36</td>
</tr>
<tr>
<td>D1.7</td>
<td>Improved Integral power plant design (reactor core and chemical plant) to maximize safety and proposal for safety demonstrator</td>
<td>CNRS</td>
<td>Month 48</td>
</tr>
</tbody>
</table>
Specificities of the MSFR impacting the 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 active area where fissions occur (but in the core vessel)
  - Fuel reprocessing and loading during reactor operation

- **No control rods**
  - 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**
  - 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)

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**PhD theses of Mariya Brovchenko, Axel Laureau and Delphine Gérardin**

- 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?

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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 ‘Integrated MSFR design’ to suppress pipes/leakage
Specificities of the MSFR impacting the safety analysis

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Voir présentation de S. Beils et D. Gérardin

Demonstration steps and Demonstrator of MSFR

**Sizing of the facilities:**

- **Small size:** ~1 liter - 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 m³ 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, ²³³U, Pu, MA) ⇒ Nuclear facility
  - Fuel salt processing: Pyrochemistry, Actinides recycling
Some PhD Thesis in France on MSR


Some documents mentioning the MSRF


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


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
Thank you for your attention!
**Chemical characteristics**

- **Chloride salts** less studied than 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 $\text{H}_2\text{O}$ contamination and thus more corrosion risk.
- More processes of separative chemistry for chloride salts.
**Liquid-fueled reactors: fluoride or chloride? Breeding**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Fluoride Salt</th>
<th>Chloride Salt</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thorium capture cross-section in core (barn)</td>
<td>0.61</td>
<td>0.315</td>
</tr>
<tr>
<td>Thorium amount in core (kg)</td>
<td>42,340</td>
<td>47,160</td>
</tr>
<tr>
<td>Thorium capture rate in core (mole/day)</td>
<td>11.03</td>
<td>8.48</td>
</tr>
<tr>
<td>Thorium capture cross-section in blanket (barn)</td>
<td>0.91</td>
<td>0.48</td>
</tr>
<tr>
<td>Thorium amount in the blanket (kg)</td>
<td>25,930</td>
<td>36,400</td>
</tr>
<tr>
<td>Thorium capture rate in the blanket (mole/day)</td>
<td>1.37</td>
<td>2.86</td>
</tr>
<tr>
<td>$^{233}$U initial inventory (kg)</td>
<td>5720</td>
<td>6867</td>
</tr>
<tr>
<td>Neutrons per fission $\nu$ in core</td>
<td>2.50</td>
<td>2.51</td>
</tr>
<tr>
<td>$^{233}$U capture cross-section in core (barn)</td>
<td>0.495</td>
<td>0.273</td>
</tr>
<tr>
<td>$^{233}$U fission cross-section in core (barn)</td>
<td>4.17</td>
<td>2.76</td>
</tr>
<tr>
<td>Capture/fission ratio $\alpha$ (spectrum-dependent)</td>
<td>0.119</td>
<td>0.099</td>
</tr>
<tr>
<td>Total breeding ratio</td>
<td>1.126</td>
<td>1.040</td>
</tr>
</tbody>
</table>
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
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’s

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 sustainability.

1. In-house research: neutronic / processing / corrosion
2. Support to CNRS and CEA research
3. Contribution to European projects

Since 2005 non moderated Thorium breeder as most promising 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
• Participation to the project TIER I of C. Bowman (1998)

• Re-evaluation of the MSBR from 1999 to 2002
  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:

$^{232}\text{Th}/^{233}\text{U}$ 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

\[ L_{\text{graphite}} \propto \frac{1}{\phi} \]

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)
**Breeding and deployment capacities**

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

**Burning efficiency**

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

Initial fissile inventory of 3 to 4.5 tons / GWe reachable
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

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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

<table>
<thead>
<tr>
<th><strong>Power</strong></th>
<th>100 MW to 300 MW</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Mean fuel salt temperature</strong></td>
<td>675 °C</td>
</tr>
<tr>
<td><strong>Fuel temperature increase in the core</strong></td>
<td>30 °C</td>
</tr>
<tr>
<td><strong>Initial fuel salt composition</strong></td>
<td>75% LiF-(Heavy Nuclei)F₄ – in Th²³³U and U/Pu fuel cycle</td>
</tr>
<tr>
<td><strong>Core size</strong></td>
<td>Internal diameter ~1.3 m External diameter ~2.3 m</td>
</tr>
<tr>
<td><strong>Fuel salt volume</strong></td>
<td>2 to 4 m³ 1.1 to 2.3 in core 0.9 to 1.7 in external circuit</td>
</tr>
<tr>
<td><strong>Total fuel cycle in the fuel circuit</strong></td>
<td>3 - 5 s</td>
</tr>
</tbody>
</table>

**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, …)

PhD thesis of Johann Martinet
## Small Modular Reactor – S-MSFR

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

<table>
<thead>
<tr>
<th></th>
<th>No radial blanket and H/D=1</th>
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</tr>
</thead>
<tbody>
<tr>
<td><strong>Power [MW&lt;sub&gt;th&lt;/sub&gt;]</strong></td>
<td>100</td>
<td>200</td>
</tr>
<tr>
<td><strong>Initial $^{233}$U load [kg]</strong></td>
<td>654</td>
<td>654</td>
</tr>
<tr>
<td><strong>Fuel reprocessing of 1l/day</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feeding in $^{233}$U [kg/an]</td>
<td>11.38</td>
<td>23.38</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>-29.83%</td>
<td>-30.64%</td>
</tr>
<tr>
<td>Total $^{233}$U needed [kg]</td>
<td>1013.87</td>
<td>1388.37</td>
</tr>
</tbody>
</table>

*Around 650kg of $^{233}$U to start*

*Under-breeder reactor*

<p>| | |</p>
<table>
<thead>
<tr>
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</tr>
</thead>
<tbody>
<tr>
<td><strong>Fuel reprocessing of 4l/day</strong></td>
<td></td>
</tr>
<tr>
<td>Feeding in $^{233}$U [kg/an]</td>
<td>11.20</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>-29.37%</td>
</tr>
<tr>
<td>Total $^{233}$U needed [kg]</td>
<td>1001.86</td>
</tr>
</tbody>
</table>

*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

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</thead>
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<tr>
<td><strong>Power [MW&lt;sub&gt;th&lt;/sub&gt;]</strong></td>
<td>100</td>
<td>200</td>
<td>100</td>
<td>200</td>
</tr>
<tr>
<td><strong>Initial $^{233}$U load [kg]</strong></td>
<td>654</td>
<td>654</td>
<td>667</td>
<td>667</td>
</tr>
<tr>
<td><strong>Fuel reprocessing of 1l/day</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feeding in $^{233}$U [kg/an]</td>
<td>11.38</td>
<td>23.38</td>
<td>1.72</td>
<td>4.70</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>-29.83%</td>
<td>-30.64%</td>
<td>-4.52%</td>
<td>-6.16%</td>
</tr>
<tr>
<td>Total $^{233}$U needed [kg]</td>
<td>1013.87</td>
<td>1388.37</td>
<td>738.83</td>
<td>835.16</td>
</tr>
<tr>
<td><strong>Breeding ratio (radial + axial fertile blankets)</strong></td>
<td>1.81%</td>
<td>-0.04%</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Fuel reprocessing of 4l/day</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Feeding in $^{233}$U [kg/an]</td>
<td>11.20</td>
<td>22.58</td>
<td>1.48</td>
<td>3.58</td>
</tr>
<tr>
<td>Breeding ratio</td>
<td>-29.37%</td>
<td>-29.59%</td>
<td>-3.88%</td>
<td>-4.69%</td>
</tr>
<tr>
<td>Total $^{233}$U needed [kg]</td>
<td>1001.86</td>
<td>1353.13</td>
<td>722.50</td>
<td>794.21</td>
</tr>
<tr>
<td><strong>Breeding ratio (radial + axial fertile blankets)</strong></td>
<td>2.49%</td>
<td>1.54%</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Addition of axial + radial fertile blankets $\Rightarrow$ small modular breeder MSFR
Small Modular Reactor – S-MSFR

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

![Graph showing the relationship between maximum valence III concentration and Th/(Th+U) ratio for different uranium enrichment levels.](image)
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%
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%

Solubility limit at 650 °C
R&D studies in the coming years: Study of a S-MSFR using U/Pu fuel cycle

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, residual heat, no fertile blanket nor draining tank, ...)

PhD thesis of Johann Martinet
R&D Activities in Reactor Physics, Design and Safety at CNRS

Concepts considered:

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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

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

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

With

- ‘B’ = location of nucleus i in the sub-system B
- ‘B \rightarrow C’ = transfer from sub-system B to sub-system C
- production by radioactive decay
- production by nuclear reaction
- 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
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

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

<table>
<thead>
<tr>
<th>Element</th>
<th>Mass Fraction (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ni</td>
<td>79.432</td>
</tr>
<tr>
<td>W</td>
<td>9.976</td>
</tr>
<tr>
<td>Cr</td>
<td>8.014</td>
</tr>
<tr>
<td>Mo</td>
<td>0.736</td>
</tr>
<tr>
<td>Fe</td>
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</tr>
<tr>
<td>Ti</td>
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</tr>
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<td>C</td>
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</tr>
<tr>
<td>Mn</td>
<td>0.257</td>
</tr>
<tr>
<td>Si</td>
<td>0.252</td>
</tr>
<tr>
<td>Al</td>
<td>0.052</td>
</tr>
<tr>
<td>B</td>
<td>0.033</td>
</tr>
<tr>
<td>P</td>
<td>0.023</td>
</tr>
<tr>
<td>S</td>
<td>0.004</td>
</tr>
</tbody>
</table>

**Reflector**

**Concept of MSFR: choice of the structural materials (Ni-based alloys)**
Radiation damages concentrated on the core axis and concern a limited material depth.
Main contribution to Helium production in the most irradiated area (radius 20 cm /thickness 2 cm) for a fuel salt volume of 18 m³ due to $^{58}$Ni

<table>
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<tr>
<th>Element</th>
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</tr>
<tr>
<td>S</td>
<td>0.004</td>
</tr>
</tbody>
</table>

⇒ Regular replacements of these area to be planned (first 10cm only) or addition of another material to protect the surface of these reflectors?
Transmutation of the Tungsten contained in the alloy into Rhenium and Osmium

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

<table>
<thead>
<tr>
<th>Ni</th>
<th>W</th>
<th>Cr</th>
<th>Mo</th>
<th>Fe</th>
<th>Ti</th>
<th>C</th>
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<td>0.033</td>
<td>0.023</td>
<td>0.004</td>
</tr>
</tbody>
</table>

W, Re and Os contents of the most irradiated area for a fuel salt volume of 18 m³:

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

→ Alloy without Tungsten
Neutronic irradiation damages to the structural materials (modify their physicochemical properties) = displacements per atom, production of Helium gas, transmutation of Tungsten in Osmium, activation

<table>
<thead>
<tr>
<th>Structural elements: layers</th>
<th>Displacements per atom</th>
<th>He production</th>
<th>Tungsten transmutation</th>
</tr>
</thead>
<tbody>
<tr>
<td>0-2.5 cm</td>
<td>6.8 dpa/year</td>
<td>12 ppm / year</td>
<td>0.11 at% /year</td>
</tr>
<tr>
<td>2.5-7.5 cm</td>
<td>3.5 dpa/year</td>
<td>6 ppm / year</td>
<td>0.07 at% /year</td>
</tr>
</tbody>
</table>

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