Overview and Perspectives of the Molten Salt Fast Reactor (MSFR) Concept

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With the support of the EVOL European Collaborative Project (FP7) and of the PACEN and NEEDS programs of CNRS
1. Background

2. The Molten Salt Fast Reactor Concept (MSFR)

3. Numerical Tools for the Simulation of the MSFR

4. Reactor studies

5. Concluding remarks
1- Why “molten salt reactors”?

Advantages of a Liquid Fuel

- Homogeneity of the fuel (no loading plan)
- Heat is produced directly in the heat transfer fluid
- Possibility to reconfigure passively the geometry of the fuel:
  - One configuration optimize the electricity production managing the criticality
  - An other configuration allow 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 deteriorate the neutronic and physicochemical performances
  - No reactivity reserve

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
Molten Salt Reactor (MSR) Concept

**MSR**

Molten Salt (fission fuel + coolant) flowing through the core and the heat exchangers

associated to

Reprocessing Units of the fuel located on-site

CMSNT-2013, January 9-11 2013, Mumbai
Future of nuclear reactors: 4th Generation Systems

Generation 4 International Forum: Criteria for Future Nuclear Reactors

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

⇒ Development of innovative MSR concepts to fulfill these criteria
Why a fast MSR?

Molten Salt Reactor operated in the Thorium Fuel Cycle

TMSR general parameters:
- total power: 2500 MWth (1000 MWe)
- salt composition:
  78% LiF – 22% (HN)F₄ (21.4% ThF₄ – 0.6% UF₄)
- mean temperature: 630 °C (900 K)

TMSR geometrical parameters:
- core radius: 1.6 m
- core shape: cylindrical (H=D)
- salt volume: 20 m³
- fertile blanket: Thorium
- hexagon size (moderator): 15 cm
- channel radius (fuel salt): varying
Why a fast MSR?
(Reactor channel radius parametric studies)

- core volume adjusted to keep the same salt volume -

3 different moderation ratios:
- thermal
- fast

CMSNT-2013, January 9-11 2013, Mumbai
Three types of configuration:

**Thermal spectrum (r = 3-6 cm)**
- Positive feedback coefficient
- Iso-breeder
- Quite long graphite life-span
- Low $^{233}$U initial inventory

**Epithermal spectrum (r = 6-10 cm)**
- Quite negative feedback coefficient
- Iso-breeder
- Very short graphite life-span
- Quite low $^{233}$U initial inventory

**Fast spectrum (r > 10 cm)**
- Very negative feedback coefficients
- Very good breeding ratio
- No problem of graphite life-span
- Larger $^{233}$U initial inventory
Why a fast MSR?

The Molten Salt Fast Reactor - MSFR
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 ⇒ Fast neutron spectrum
Selection of the fuel salt

Which constraints for a liquid fuel?

- Melting temperature not too high
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- Low vapor pressure
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Lithium fluorides fulfill all constraints

Molten Salt Reactors

Thorium /\(^{233}\)U Fuel Cycle

Neutronic cross-sections of fluorine versus neutron economy in the fuel cycle
• Initial Salt: 77.5%LiF – (fissile matter)F₃ - ThF₄
  
  \((fissile \ matter = {}^{233}\text{U}, \ TRU (Pu), \ enriched\text{U})\)

• \(^{233}\text{U}\) initial inventory per \(GW_e\): 3600 kg

• Th feeding per \(GW_e\): 1100 kg/year

• Mean fuel temperature: 750°C

• Produced power: 3 GW\(_{th}\) (~1.5 GW\(_e\))

• Specific power: 330 W/cm³

• Fuel Salt Volume: 18 m³
  
  1/2 in the active zone (core + plenums)
  
  1/2 in the external circuit (heat exchangers, pipes, pumps)

• Fuel salt circulation: 3.9 seconds

• Core Internal Diameter = Core Height = 2.3 m

• Core reprocessing: 10-40 l of fuel salt per day (batch reprocessing) + on-line Bubbling

• \(^{233}\text{U}\) production (breeder reactor): 52 to 90 kg/year

• Reactor doubling time: 98 to 56 years
3- Tools for the Simulation of Reactor Coupling of neutronics and reprocessing simulation codes

- nuclear reaction database
- chemical database
- radioactive database

REM
- new compositions
- reaction rates

MCNP

evolution code

Results
REM solves the Bateman equation

- for all nuclei
- in all components of the system (core, reprocessing unit, bubbling system, draining tank, etc.)
- Including production/destruction terms from nuclear reaction and radiative decay, and also inlet/outlet flows between components due to feeding and/or reprocessing operations
Results of a reactor evolution simulation:

- compositions
- reaction rates
- mean cross sections
- neutron spectrum
- breeding ratio
- material irradiation

... in all cells at any time for all nuclei
Coupling of neutronics and thermalhydraulics codes

• Adequate design of the MSFR requires coupled neutronics and T&H numerical simulations to take into account the phenomena caused by the liquid fuel circulation
• Some of these phenomena includes:
  o Circulation of delayed neutron precursors
  o Effect of the salt temperature distribution of the feedback coefficient
  o Fuel irradiation
  o Fluctuations of the reactor core wall temperature distribution
• Due to the complexity of the flow distribution, a Computational Fluid Dynamics (CFD) code is used to determine the velocity and temperature distribution in the fuel salt
• The CFD code is coupled to the MCNP code which provides the power distribution
4- Reactor studies: 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 15m$^3$ and 400W/cm$^3$

$\Rightarrow$ Reference MSFR configuration with 18m$^3$ et 330 W/cm$^3$ corresponding to an initial fissile inventory of 3.5 tons per GWe
## Optimization = Medium Fuel Salt Volumes

<table>
<thead>
<tr>
<th>Fuel salt volume / specific power</th>
<th>t(100 dpa)</th>
<th>t(100 ppm He)</th>
<th>t(-1 at% of W)</th>
</tr>
</thead>
<tbody>
<tr>
<td>12 m³ - 500 W/cm³</td>
<td>85 years</td>
<td>2.2 years</td>
<td>4.7 years</td>
</tr>
<tr>
<td>18 m³ - 330 W/cm³</td>
<td>133 years</td>
<td>3.2 years</td>
<td>7.3 years</td>
</tr>
<tr>
<td>27 m³ - 220 W/cm³</td>
<td>211 years</td>
<td>5.5 years</td>
<td>10.9 years</td>
</tr>
</tbody>
</table>
Starting modes of the MSFR

Nuclear reactors need fissile matter to operate – Only $^{235}$U naturally available

4th generation breeder reactors: fissile load required only once at the beginning but their initial fissile matter ($^{233}$U for Th/$^{233}$U cycle or $^{239}$Pu for U/Pu cycle) does not exist on earth ⇒ Necessity to produce it from $^{235}$U

How to launch the Thorium fuel cycle in MSFR?

- Start directly with enriched U as initial fissile load
- Start with the Pu of current PWRs mixed with other TRU elements
- Use Th (or MOx-Th) in Gen3+ or other Gen4 reactors to produce $^{233}$U
- Mix these solutions

Simulations of MSFR identical to the $^{233}$U-started version, Thorium fueled but started with different fissile matters

⇒ Fuel compositions identical at equilibrium, characteristics very similar
Starting modes of the MSFR

Initially TRU (~7 tons of Pu/GWe) and $^{232}$Th (24 tons / GWe)
Breeder reactor (fed with 1t of Th / GWe per year) - All negative safety coefficients

Initial Pu proportion of the TRU-started MSFR reaches the solubility limit $\Rightarrow$ either operate at higher temperature or mix initial Pu load with another fissile matter
Which initial fissile load for the MSFR?

- Start directly with enriched U: enrichment required > 25%
- Start with the Pu of current LWRs mixed with other TRU elements: solubility limit in LiF
- Put Th in Gen3+ or other Gen4 reactors to produce $^{233}$U

- Mix of these solutions:
  - $^{233}$U + TRU produced in LWRs
  - MOx-Th in Gen3+ / other Gen4
  - Uranium enriched at 13% + TRU currently produced + Th
Starting modes of the MSFR

![Graph showing starting modes of the MSFR](image)
**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

Reprocessing by batch of 10-40 l per day
The concept of MSFR: Fuel Reprocessing

**Batch reprocessing:**

<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) reprocessing:**

Fast neutron spectrum
⇒ very low capture cross-sections
⇒ low impact of the FP extraction on neutronics
⇒ Parallel studies of chemical and neutronic issues possible
### MSFR Material Studies

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

<table>
<thead>
<tr>
<th>Ni</th>
<th>W</th>
<th>Cr</th>
<th>Mo</th>
<th>Fe</th>
<th>Ti</th>
<th>C</th>
<th>Mn</th>
<th>Si</th>
<th>Al</th>
<th>B</th>
<th>P</th>
<th>S</th>
</tr>
</thead>
<tbody>
<tr>
<td>79.432</td>
<td>9.976</td>
<td>8.014</td>
<td>0.736</td>
<td>0.632</td>
<td>0.295</td>
<td>0.294</td>
<td>0.257</td>
<td>0.252</td>
<td>0.052</td>
<td>0.033</td>
<td>0.023</td>
<td>0.004</td>
</tr>
</tbody>
</table>

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
MSFR Safety Characteristics and Risk Analysis

Liquid fuel

+ Reprocessing during reactor operation
  - No reactivity reserve
  - Decay heat reduction in the core
- Solubility

+ Fuel element is already molten
- Salt dissociation/ebullition
+ No high pressure
+ Possibility of a passive draining
  - Transfer of the fuel element from the core to heat sinks

Fuel salt is
"the coolant"

+/- Fuel element is bidden to the heat transfer fluid
- Pumps and heat exchangers are under radiation (neutrons)

High °T

(+ High efficiency)
- Structure material behavior at high T°
For a liquid fuel reactor another safety approach is needed:

→ Core melt accident is not the reference accident
→ Loss of the coolant = loss of the fuel element

Guidelines available (IAEA, INPRO, GIF...) have to be adapted to liquid-fuelled reactors

⇒ Studies of the system by applying global risk analyses:
- To identify the major accidents for the MSFR
- To define and design the safety systems

MSFR concept developed with a Design-by-safety approach:
Iterative integration of the results of the safety studies in the early stages of the reactor design
Specificities of a nuclear reactor:

- Huge energy reserve concentrated in the fuel
- Accumulation of radioactive elements (dangerous + produce heat)
- Large release of energy even after the reactor shutdown

Bases of the nuclear safety = control the reactor

- Confinement of the radioactive elements (= 3 barriers in LWRs)
- Control of the chain reaction at any time = drive the reactor
- Heat evacuation even after the chain reaction stops (residual heat management)
1. Total reactivity reserves in the core: very low reserves to be inserted accidentally

<table>
<thead>
<tr>
<th>Source</th>
<th>Accidental Reactivity Insertion</th>
<th>Loss of Salt Circulation</th>
<th>Core Draining</th>
<th>Blanket Loss</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactivity Margin</td>
<td>95 pcm</td>
<td>180 pcm</td>
<td>-</td>
<td>-</td>
<td>&lt; 300 pcm</td>
</tr>
</tbody>
</table>

2- Safety parameters: Fraction of delayed neutrons

\[ \beta_{\text{tot}} = 360 \pm 7 \text{ pcm} \quad \text{with } 1/2 \text{ of the fuel salt out of flux} \]

\[ \beta_{\text{core}} \sim 180 \text{ pcm} \]

\[ T_{\text{circul}} (s) \quad \beta (\text{pcm}) \]

<table>
<thead>
<tr>
<th>( T_{\text{circul}} ) (s)</th>
<th>( \beta ) (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>4</td>
<td>191</td>
</tr>
<tr>
<td>3</td>
<td>187</td>
</tr>
<tr>
<td>2</td>
<td>185</td>
</tr>
</tbody>
</table>

Precursor \((i)\) | \(^{87}\text{Br}\) | \(^{137}\text{I}\) | \(^{88}\text{Br}\) | \(^{93}\text{Rb}\) | \(^{139}\text{I}\) | \(^{91}\text{Br}\) | \(^{96}\text{Rb}\) |
<table>
<thead>
<tr>
<th></th>
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</tr>
</thead>
<tbody>
<tr>
<td>Lifetime ((\ln 2/l_{i}))</td>
<td>55.9 s</td>
<td>24.5 s</td>
<td>16.4 s</td>
<td>5.85 s</td>
<td>2.3 s</td>
<td>0.54 s</td>
<td>0.20 s</td>
</tr>
<tr>
<td>Abundance ((C_{i}))</td>
<td>0.074</td>
<td>0.168</td>
<td>0.121</td>
<td>0.192</td>
<td>0.353</td>
<td>0.068</td>
<td>0.024</td>
</tr>
</tbody>
</table>

See Fuel salt cycle in “core+pipes+IHX” = 3 to 4 seconds
3- Safety parameters: Feedback coefficients

\[ \frac{dk}{dT} = \text{Variation of the multiplication factor (dk) with the core temperature (dT)} \]

**Reactor intrinsically safe if** \( \frac{dk}{dT} < 0 \) (if \( T_k \) then \( k_m \))

\( \Rightarrow \) \( \frac{dk}{dT} \) 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

\[ \Rightarrow \text{MSFR: Only Gen4 system being both breeder and with all negative safety coefficients} \]
Transient response following a reactivity insertion

Reactor response was found to be acceptable under most credible scenarios.
Transient response following a variation of the power demand

Example: 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)
Heat evacuation: decay heat calculations

- Lower decay heat 3.5 % compared to solid-fuelled reactors (PWR 6%)
- Important part of the FP decay heat transferred in the reprocessing units
- Actinides dominant some hours after the reactor shut down (Pa effect)
Hypothesis: no heat losses

Temperature Limits:

Fuel salt \( T_{\text{lim}} \approx 1600^\circ \text{C} \)

Materials \( T_{\text{lim}} \approx 1200^\circ \text{C} \)

\[ T_{\text{mat}} \neq T_{\text{salt}}^{\text{mean}} \]

Salt is a bad heat conductor:

\[ \lambda_{\text{mat}} \gg \lambda_{\text{salt}} \]

- Drainage must occur in **5-10 min** in order not to damage the system
- **Fuel drainage** is an important safety issue (see cold plug tests)
- The impact of the **inertia of the cooling system** is very important

Technological issue
First steps toward a demonstration of MSFR: the FFFER loop

The FFFER experiment
in progress in Grenoble (LPSC)

Forced Fluoride Flow for Experimental Research:
a 80liters loop of circulating FLiNaK at 550°C
to test the on-line processing

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- salt manipulation & confinement
- salt-bubbles exchange of rare gases
- metallic particles flotation by bubbles (future)
- measurements (velocity, bubbles, particles)
The Forced Fluoride Flow Experiment
Reproduces the gases and particles extractions at 1/10th flow scale in simulant salt
First steps toward a demonstration of MSFR: the FFFER loop

loop final design

Tank pressurization is used for loop filling. Draining is done by gravity.

The “cold plug” is a system where some quantity of salt is solidified to form a plug which prevents the salt from going back to the tank.

It is foreseen as a passive security system: without cooling, the plug melts before solidification of the salt in the loop.
5- Concluding remarks

- Preliminary design and safety studies confirmed the MSFR attractive features which make this concept very suitable for breeding and for minor actinide burning.

- No design stopper have been identify in the current design and safety studies.

- However, there are a number of technical areas that require further progress to prove the industrial feasibility of this concept.