MSR provisional System Steering committee

J. Serp, France

GIF Symposium
Chiba
May 19, 2015
Studied Concepts

Two reactor concepts using molten salt are discussed in GIF MSR meetings

- **Molten salt reactors, in which the salt is at the same time the fuel and the cooling liquid**
  - MSR MOU Signatories France and EU work on **MSFR (Molten Salt Fast Reactor)**
  - Russian Federation works on **MOSART (Molten Salt Actinide Recycler & Transmuter)**. **Russian Federation joined the Memorandum of Understanding (11/2013)**

- **Solid fueled Reactors cooled by molten salt**
  - USA and China work on **FHR (fluoride-salt-cooled high-temperature reactor)** concepts and are **Observers** to the **PSSC**

High temperature (750 °C)
Low pressure (1 bar)
1000 MWe

**MSFR**

**MOSART**

**FHR** 3,400 MWth
Reference concept:

From thermal to fast neutron spectrum

The first Molten Salt Reactors (MSR) developed in the USA (1960s and 1970s) were thermal-neutron-spectrum graphite-moderated concepts.

Since 2005, European R&D interest has focused on fast neutron MSR (MSFR) as a long term alternative to solid-fueled fast neutrons reactors.

General characteristics of MSR

- Molten fluorides as fuel fluid (no loading pattern)
- Low-pressure and high boiling-point coolant
- Possibility to drain fuel passively towards non-critical volumes
- On-site fuel reprocessing unit

Specific features of MSFR

- Strongly negative reactivity feedback coefficients (thermal and void)
- Reprocessing needs decreased (from several m³ to 40 liters/day)
- No graphite elements in the core (maintenance)
GIF MSR Project

- A Provisional Project Management Board has been set up
  - Two meetings per year where members and observers report on their activities and recent progresses

- The project is devoted to Molten Salt Reactors
  - Information is also exchanged on solid fueled reactors cooled by molten salt

- The various molten salt reactor projects like FHR, MOSART, MSFR, and TMSR have common themes in basic R&D areas, of which the most prominent are:
  - liquid salt technology,
  - materials behavior,
  - the fuel and fuel cycle chemistry and modeling,
  - the numerical simulation and safety design aspects of the reactor
Collaborations (1/2)

SAMOFAR Project – Safety Assessment of a MOlten salt FAst Reactor

4 years (2015-2019), 3,5 M€

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

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

5 technical work-packages:

WP1 Integral safety approach and system integration
WP2 Physical and chemical properties required for safety analysis
WP3 Proof of concept of key safety features
WP4 Numerical assessment of accidents and transients
WP5 Safety evaluation of the chemical processes and plant
Collaborations (2/2)

US and China Are Initiating a Cooperative Research and Development Agreement (CRADA) on FHRs

- Collaboration supports the US-China memorandum of understanding on cooperation in civilian nuclear energy science and technology
- ORNL and the Shanghai Institute of Applied Physics (SINAP) are the lead organizations
- Project is intended to benefit both countries through more efficiently and rapidly advancing a reactor class of common interest
- FHR remain at a pre-commercial level of maturity
  - All of the results are intended to be openly available
  - Project is scheduled to end after SINAP’s higher-power test reactor has completed its operational testing program
- Collaboration includes research and development to support the evaluation, design, and licensing of a new reactor class
  - Does not include fissile material separation technology
Liquid fueled-reactors

Which constraints for a liquid fuel?

• Melting temperature not too high
• High boiling temperature
• Low vapor pressure
• Good thermal and hydraulic properties
• Stability under irradiation

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

There are some challenges for MSR that must be factored into design

• Must keep system at high temperature to avoid salt freezing
• Lifetime of components (graphite)
• Chemical interactions with structural materials
• The salt of choice (LiF based salt) produces tritium during operation and requires Li enrichment
• Complexity of a combined reactor and fuel processing system
In the last decade ITU (European Commission) has developed an expertise in determination of High temperature properties of An fluorides and mixtures.

- Phase diagrams
- Melting points
- Heat capacity

Vapour pressure
Knudsen cell with MS up to 2800 K

Drop and DSC calorimeters up to 1800 K
LiF-ThF$_4$ system

- The vapour pressure of the end members LiF and ThF$_4$ were measured, obtaining a very good agreement with the literature data. The appearance potential of the different species has been also measured.

<table>
<thead>
<tr>
<th>Ion</th>
<th>AP (Lit.)</th>
<th>AP (Meas.)</th>
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<tbody>
<tr>
<td>Th$^+$</td>
<td>39 eV</td>
<td>39.9 eV</td>
</tr>
<tr>
<td>ThF$^+$</td>
<td>30 eV</td>
<td>31.9 eV</td>
</tr>
<tr>
<td>ThF$_2^+$</td>
<td>23 eV</td>
<td>23.9 eV</td>
</tr>
<tr>
<td>ThF$_3^+$</td>
<td>14.5 eV</td>
<td>16.5 eV</td>
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</table>
LiF-ThF$_4$-UF$_4$-PuF$_3$ ternary system assessed

$T_{\text{min}} = 818.14$ K

LiF-ThF$_4$-PuF$_3$ (69.6-28.6-1.8)
Actinides and lanthanides solubility measurements

Individual and joint solubility of PuF$_3$ and UF$_4$ in LiF-NaF-KF eutectic, mol. %

<table>
<thead>
<tr>
<th>Temperature, K</th>
<th>Individual Solubility, mol.%</th>
<th>Joint Solubility, mol. %</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>PuF$_3$</td>
<td>UF$_4$</td>
</tr>
<tr>
<td>823</td>
<td>6.1±0.6</td>
<td>15.3±0.8</td>
</tr>
<tr>
<td>873</td>
<td>11.1±1.1</td>
<td>24.6±1.2</td>
</tr>
<tr>
<td>923</td>
<td>21.3±2.1</td>
<td>34.8±1.7</td>
</tr>
<tr>
<td>973</td>
<td>32.8±3.3</td>
<td>44.7±2.2</td>
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<tr>
<td>1023</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>1073</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Temperature, K</th>
<th>72,5LiF-7ThF$_4$-20,5UF$_4$</th>
<th>78LiF-7ThF$_4$-15UF$_4$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>PuF$_3$ CeF$_3$</td>
<td>PuF$_3$ CeF$_3$</td>
</tr>
<tr>
<td>873</td>
<td>0.35±0.02 1.5±0.1</td>
<td>1.45±0.7 2.6±0.1</td>
</tr>
<tr>
<td>923</td>
<td>4.5±0.2 2.5±0.1</td>
<td>5.6±0.3 3.6±0.2</td>
</tr>
<tr>
<td>973</td>
<td>8.4±0.4 3.7±0.2</td>
<td>9.5±0.5 4.8±0.3</td>
</tr>
<tr>
<td>1023</td>
<td>9.4±0.5 3.9±0.2</td>
<td>10.5±0.6 5.0±0.3</td>
</tr>
</tbody>
</table>

Near the liquidus temperature for 78LiF-7ThF$_4$-15UF$_4$ and 72,5LiF-7ThF$_4$-20,5UF$_4$ salts, the CeF$_3$ significantly displace plutonium trifluoride.
Material corrosion

- The corrosion facility allows to test the alloy specimens in the nonisothermal dynamic conditions

- For the fuel salt with \([U(IV)/U(III)]\) ratio = 90 at 800°C, the tellurium intergranular corrosion for the HN80MTY alloy (Russian Federation) is by about ten times lower as compared to original Hastelloy N
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

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
Safety analysis: accident types

Classified by the initiators of the transient:

TOP - Transient Over Power (or RAA - Reactivity Anomalies Accident)

LOF - Loss Of Flow (in the fuel circuit)
  ✗ Pumps of the fuel salt & ✔ Pumps of the intermediate fluid

LOH - Loss Of Heat sink (in the fuel circuit)
  ✔ Pumps of the fuel salt & ✗ cooling of the fuel salt

TLOP - Total Loss Of Power

OVC – Fuel salt OVer-Cooling

LOLF - Loss Of Liquid Fuel
  ✗ Confinement of the fuel salt in the fuel circuit
MSFR and Safety Evaluation: example of accidental scenario

Scenario = passive decrease of the chain reaction (thermal feedback coefficients) + increase of the fuel salt temperature due to residual heat
MSFR TRANSIENT CALCULATIONS: THE TFM (TRANSIENT FISSION MATRIX) APPROACH

- Liquid fuel (precursor motion)
- Fuel = coolant
- Fast neutron spectrum
- Circulation time ~ 3 s
- Reynolds in core: ~ 500000
- Power: 3GWth
- Molten Salt: LiF - (Th\(^{233}\)U)\(_4\)
  - density: 4 x water
  - viscosity: 2 x water (oil ~ 1000x water)
  - low pressure
  - mean fuel temperature ~ 900 K

Objective: multiphysics simulations of liquid-fuelled reactors – here optimized coupling of neutronics + thermal-hydraulics:

- high precision of the T&H modeling (flow distribution)
  - CFD code: OpenFOAM
- high precision of the neutronics modeling ...
  - Monte Carlo code - MCNP or SERPENT codes
- ... with a low computational cost (many cases to perform)
  - Diffusion? Improved point kinetics? ...
  - innovative method: TFM approach

1/16 of the reactor modeled

MSFR scheme
MSFR TRANSIENT CALCULATIONS: THE TFM (TRANSIENT FISSION MATRIX) APPROACH

General Coupling Strategy

Numerical scheme

outer iteration

\[ G_{\text{TFM}}(T_i) \text{ and } P_i \text{ interpolation} \]

neutronics time step

T&H time step

TFM (developed model)

neutronics

thermal-hydraulics

CFD

k-ε realizable turbulence model

delayed neutron source

Doppler feedback effect

density feedback effect

TFM mesh

CFD mesh

temperature field

precursor decay position

performed by

power

buoyancy effect

precursor production

matrices calculation

directly implemented and performed in

Serpent (Monte Carlo calculation code)
Good behavior of the MSFR for load-following transients
- Coupling code (OpenFoam – TFM) operational
- Calculations with high precision & low computational cost
FHR : No Technology Breakthroughs Required

Significant Technical Development and Demonstration Remains

- Tritium release prevention is the most significant technical issue
  - Tritium stripping membranes are promising new technology
  - Double walled heat exchangers acceptable
- Replacement industrial scale lithium enrichment
- Salt chemistry control system requires design for large scale
- Qualified fuel must be developed
- Structural ceramics must become safety grade nuclear engineering materials
- Safety and licensing approach must be developed and demonstrated
- Instrumentation has substantial technical differences from LWR technology
- More complete reactor conceptual design required

FHRs are emerging from viability assessment and entering into technology development and engineering concepts
UC Berkeley built the Compact Integral Effects Test (CIET) facility, to validate computer models for passive, natural circulation heat removal from FHRs under both steady-state and transient conditions.

CIET will provide integral effects test data to validate thermal hydraulics safety codes for application to FHRs.
Tritium Control is Necessary for FHR Acceptability

- At FHR temperatures tritium diffuses through structural alloys
  - Primary heat exchanger is a significant escape path
- Membrane reactor recently invented to strip tritium from fluoride salts
  - Turbulent salt flow overcomes slow diffusion limits of sparging and spraying techniques
  - Similar to systems used for gaseous hydrogen separation
- Double walled heat exchangers coupled with a sweep gas or yttrium chemical trap can block tritium escape
  - Tritium diffusion barrier layers have proven challenging in practice
  - Trapping tritium at the primary to intermediate heat exchanger preserves separation of nuclear and non-nuclear portions of plant
Chinese program

The near-term Goal of TMSRs project:
- 2MW Pebble-bed FHR (TMSR-SF1) (~2017)
- 2MW Molten Salt Reactor with liquid fuel (~2020)
- Build up R&D abilities (include research conditions, key technology and research team, Molten-Salt Test Loops, radiochemistry research platform etc.) for future TMSR development, including

Long-term Goal of TMSRs: ~100MW

NG-CT-10: nuclear graphite  SiC heat exchanger  GH3535 domestic alloy
MSR offer many options: thermal or fast neutron spectrum, breeder or burner, with or without thorium support.

Although MSR-pSSC partners interests are focused on different baseline concepts (MSFR, MOSART and FHR), large commonalities in basic R&D areas (liquid salt technology, materials, safety aspects) do exist and the Generation IV framework could be useful to optimize the R&D effort.
Thank you for your attention