MOLTEN SALT ACTINIDE RECYCLER & TRANSFORMING SYSTEM WITH AND WITHOUT TH-U SUPPORT: MOSART

Dr. Victor Ignatiev
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Meet the presenter

Dr. John E. Kelly is the Deputy Assistant Secretary for Nuclear Reactor Technologies in the U.S. Department of Energy's Office of Nuclear Energy. His office is responsible for the civilian nuclear reactor research and development portfolio, which includes programs on Small Modular Reactors, Light Water Reactor Sustainability, and Advanced (Generation IV) Reactors. His office also is responsible for the design, development, and production of radioisotope power systems, principally for missions of the U.S. National Aeronautics and Space Administration. In the international arena, Dr. Kelly is the immediate past chair of the Generation IV International Forum (GIF) and the former chair of the International Atomic Energy Agency's Standing Advisory Group on Nuclear Energy. Prior to joining the Department of Energy in 2010, Dr. Kelly spent 30 years at Sandia National Laboratories, where he was engaged in a broad spectrum of research programs in nuclear reactor safety, advanced nuclear energy technology, and national security. In the reactor safety field, he led efforts to establish the scientific basis for assessing the risks of nuclear power plant operation and specifically those risks associated with potential severe accident scenarios. His research focused on core melt progression phenomena and this led to an improved understanding of the Three Mile Island accident and, more recently, the Fukushima Daiichi accident. In the advanced nuclear energy technology field, he led efforts to develop advanced concepts for space nuclear power, Generation IV reactors, and proliferation-resistant and safe fuel cycles. These research activities explored new technologies aimed at improving the safety and affordability of nuclear power. Dr. Kelly received his B.S. degree in nuclear engineering from the University of Michigan in 1976 and his Ph.D. in nuclear engineering from the Massachusetts Institute of Technology in 1980.

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Dr. Victor Ignatiev works at the NRC-“Kurchatov Institute”, Moscow, Russia both as the Head of the Molten Salt Reactor Laboratory (since 2012) and as a Professor (since 2009). He graduated from the Nuclear Power Systems Moscow Physical Engineering Institute, USSR, in 1976, and earned his Ph.D. in 1986 from the Kurchatov Institute of Atomic Energy, Moscow, USSR. His Ph.D. research focused on molten salt reactors. Since 2014, he has been the co-chair of Generation IV International Forum MSR pSSC. In 1985, he received the Kurchatov Award on the Fundamental Studies of Molten Salt Reactors and in 2016, he received the Kurchatov Award on Engineering studies of Molten Salt Reactors. The main area of his research activities focus on Molten Salt Reactor: (1) Th - U fuel cycle and TRU burners, (2) Combined materials compatibility & salt chemistry control in selected molten salt environments at parameters simulating designs operation, (3) Physical & chemical properties for fuel and coolant salt compositions, and (4) Flow sheet optimization, including reactor physics, thermal hydraulics and safety related issues.

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Contents

- Introduction
- Core neutronics, thermal hydraulics and fuel cycle properties
- Safety aspects
- Key physical and chemical properties of fuel salt
- Materials compatibility and salt chemistry control
- Summary
In MSR devices solid fuel elements are replaced by liquids

Mechanical engineering device assumes that the fuel (solid) has to be used in a maximum condensed form that excludes reprocessing and has advantage of technical simplicity while reactor operating.

Chemical engineering device has not only possibilities of general benefits such as unlimited burn-up, easy and relatively low cost of purifying and reconstituting the fuel (fluid), but also there are some more specific potential gains.
This contribution proves the feasibility of Molten Salt Actinide Recycler & Transformer (MOSART) system fueled with TRU trifluorides from SNF in different scenarios without and with U-Th support

- The webinar has the main objective of presenting the fuel cycle flexibility and inherent safety features of the MOSART system while accounting technical constrains and experimental data received in our studies.
- MOSART design options with homogeneous core and fuel salt with high enough solubility for transuranic elements trifluorides were examined.
- A brief description is given of the experimental results on key physical and chemical properties of fuel salt as well as combined materials compatibility to satisfy MOSART system requirements.
Started with TRU Fluorides from LWR SNF MOSART can operate in different modes: Transmuter, Self-sustainable, Breeder
• Max temperature of the fuel salt in the primary circuit made of special Ni-alloy is mainly limited by Te IGC depending on salt Redox potential
• Min temperature of fuel salt is determining not only its melting point, but also the solubility for AnF$_3$ in the solvent for this temperature
MOSART – Transforming Reactor System

Heat source, W/m³

Velocity, m/s

$T_{\text{max wall}}$: 947 K; $T_{\text{max fluid}}$: 1192 K

$= 1.65 \text{ m}$
MOSART Fuel Cycles

<table>
<thead>
<tr>
<th>Solvent, mole %</th>
<th>Feed MA/TRU</th>
<th>Loading (EOL), t</th>
<th>TRU/MA, kg/yr</th>
</tr>
</thead>
<tbody>
<tr>
<td>15LiF-58NaF-27BeF₂</td>
<td>0.1</td>
<td>7.7</td>
<td>730/73</td>
</tr>
<tr>
<td>73LiF-27BeF₂</td>
<td>0.1</td>
<td>3.9</td>
<td>730/73</td>
</tr>
<tr>
<td>73LiF-27BeF₂</td>
<td>0.35</td>
<td>13.9</td>
<td>730/260</td>
</tr>
<tr>
<td>73LiF-27BeF₂</td>
<td>0.45</td>
<td>23.2</td>
<td>730/330</td>
</tr>
</tbody>
</table>

Contribution to Keff

<table>
<thead>
<tr>
<th>Isotope</th>
<th>BOL</th>
<th>EOL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu 239</td>
<td>0,75</td>
<td>0,25</td>
</tr>
<tr>
<td>Pu 238</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Pu 241</td>
<td>0,25</td>
<td>0,41</td>
</tr>
<tr>
<td>Cm 245</td>
<td>-</td>
<td>0,28</td>
</tr>
</tbody>
</table>

Figure: MOSART Fuel Cycles

- Neutron Flux per lethargy (cm⁻²s⁻¹)
- TRU Molar Concentration over time
- Contribution to Keff

Legend:
- MOSART with LiNaBe/F solvent
- MOSART with LiBe/F solvent

Solvent, feed, and loading values shown for different isotope combinations.
In MOSART core, the limitations on the radiation resistance of structural materials, along with the possibilities of heat removal, represent the main factors that inhibit the increase in the core specific power > 140 W / cm³.

The temperature in the fuel circuit due to the decay heat without heat sink should not reach the maximum temperature for the structural material.

He embrittlement for Ni-base alloy at T > 500°C:

\[
^{58}\text{Ni} + n \rightarrow ^{55}\text{Fe} + ^{4}\text{He}, ( >1\text{MeV});
\]

\[
^{60}\text{Ni} + n \rightarrow ^{57}\text{Fe} + ^{4}\text{He};
\]

\[
^{10}\text{B} + n \rightarrow ^{7}\text{Li} + ^{4}\text{He}.
\]

\[
^{58}\text{Ni} + n \rightarrow ^{59}\text{Ni} + \gamma, ^{59}\text{Ni} + n \rightarrow ^{56}\text{Fe} + ^{4}\text{He}.
\]

Basing on neutron fluence (3.8*10²¹n/(cm²yr)) and temperature (860-1000K) reflector should be changed in 5 yrs.
Nuclear fuel is a fluid. It circulates throughout the RCS, transfers heat to heat exchanger and becomes critical only in core.

Possible initiators of RCS breach accident: pipe failure missiles, and pressure or temperature transients in RCS, failure of the boundary between the 1st and 2nd salts in heat exchanger.

Problem of developing RCS which will be reliable, maintainable, inspectable over the plant's lifetime will probably be key factor in demonstrating ultimate safety and licenseability.

MSR must be designed so that decay heated fuel salt reaches the drain tank under any credible accident conditions.
The MOSART is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP. The system was shown to buffer reactivity insertion of up to +1.5$. System temperatures are expected to rise only ~300°C above nominal under this severe transient conditions. The mechanical and structural integrity of the system is not expected to be impaired.
Severe Accident with the Rupture of the Main Fuel Salt Pipe and Fuel Discharged on the Reactor Cell Bottom

- The model based on mass transfer theory describing main radionuclides distribution between the fuel salt, metallic surfaces of the primary circuit, graphite and the gas purging system was applied for calculation releases to the containment atmosphere.
- As a criteria characterizing an isotope yield from the fuel salt is accepted the ratio of this isotope activity changed into a gas phase of a containment \( (A_g) \) to its full activity built up in a reactor by the moment of the accident \( (A_o) \).
- After accident considered all noble gases and metals available should move to the gas phase \( (A_g/A_s = 1) \), where \( A_g/A_s \) - the ratio of isotope activity in the gas phase of the containment after an accident to its activity concentrated in the fuel salt by the moment of the accident.
- However, already as it noted before during the normal operation these nuclides are almost completely leave the fuel salt.
- For MSR the total release of radioactivity would be significantly lower (by 1 - 2 orders of magnitude compared to LWR), though for several particular nuclides such I\(^{131}\) and I\(^{133}\) the differences are smaller.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>( A_s/A_o )</th>
<th>( A_g/A_s )</th>
<th>( A_g/A_o )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Te129</td>
<td>0,25</td>
<td>1</td>
<td>0,25</td>
</tr>
<tr>
<td>Te132</td>
<td>0,005</td>
<td>1</td>
<td>0,005</td>
</tr>
<tr>
<td>Ru103</td>
<td>0,01</td>
<td>1</td>
<td>0,01</td>
</tr>
<tr>
<td>Ru106</td>
<td>0,001</td>
<td>1</td>
<td>0,001</td>
</tr>
<tr>
<td>Nb95</td>
<td>0,034</td>
<td>1</td>
<td>0,034</td>
</tr>
<tr>
<td>Zr95</td>
<td>0,99</td>
<td>0,0011</td>
<td>0,0011</td>
</tr>
<tr>
<td>Sr89</td>
<td>0,99</td>
<td>0,00046</td>
<td>0,00046</td>
</tr>
<tr>
<td>Sr90</td>
<td>0,98</td>
<td>0,00046</td>
<td>0,00046</td>
</tr>
<tr>
<td>La140</td>
<td>0,98</td>
<td>0,026</td>
<td>0,025</td>
</tr>
<tr>
<td>Ce141</td>
<td>0,99</td>
<td>0,0024</td>
<td>0,023</td>
</tr>
<tr>
<td>Ce144</td>
<td>0,96</td>
<td>0,0024</td>
<td>0,023</td>
</tr>
<tr>
<td>I131</td>
<td>0,62</td>
<td>0,43</td>
<td>0,27</td>
</tr>
<tr>
<td>I133</td>
<td>0,94</td>
<td>0,43</td>
<td>0,43</td>
</tr>
<tr>
<td>Cs137</td>
<td>0,7</td>
<td>0,016</td>
<td>0,011</td>
</tr>
</tbody>
</table>
Probability of an accident with a relatively low impact for MSR is higher than for LWR. This is due to the possibility of leakage of radioactive liquid fuel in case of accidents in the pump, piping, valves.

The consequences of severe accidents in particular, leading to the release of radioactive products into the environment for MSR significantly less than for LWRs.

Taube M., Fast and thermal molten salt reactors with improved inherent safety // TANS, 1981, Summer meeting, pp. 490-498
Gen IV MSR container materials

Combined environments

- Radiation
  - Fast neutrons
- High Temp
  - Creep, Creep-fatigue, Thermal fatigue, Aging, Welds...
- Corrosion
  - Redox, Heat up, Velocity ...

Corrosion effects

- Fission products
- Fuel addition
- Impurities
- Molten salt
- Gas
- Metallic alloy
- Graphite
In temperature range 500-800°C about 70 differently alloyed specimens of HN80MT were tested. Among alloying elements there were W, Nb, Re, V, Al and Cu.

Experiments results in polythermal loops with redox potential measurement demonstrated that operations with Li,Be/F salt, fuelled by UF₄ or PuF₃, are feasible using carefully purified molten salts and loop internals. Corrosion rate of HN80M-VI and HN80MTY in the LiF-BeF₂+UF₄, LiF-BeF₂-ThF₄+UF₄ и LiF-NaF-BeF₂+PuF₃ melts was <5μm/yr. No intergranular corrosion of alloys is observed. Alloys modified by Ti, Al and V have shown the best post irradiation properties.

### Table: Elemental Composition

<table>
<thead>
<tr>
<th>Element</th>
<th>Hasteloy N US</th>
<th>Hasteloy NM US</th>
<th>HN80M-VI Russia</th>
<th>HN80MTY Russia</th>
<th>MONICR Czech Rep</th>
<th>E-721 France</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ni</td>
<td>base</td>
<td>base</td>
<td>base</td>
<td>base</td>
<td>base</td>
<td>base</td>
</tr>
<tr>
<td>Cr</td>
<td>7,52</td>
<td>7,3</td>
<td>7,61</td>
<td>6,81</td>
<td>6,85</td>
<td>8</td>
</tr>
<tr>
<td>Mo</td>
<td>16,28</td>
<td>13,6</td>
<td>12,2</td>
<td>13,2</td>
<td>15,8</td>
<td>0,7</td>
</tr>
<tr>
<td>Ti</td>
<td>0,26</td>
<td>0,5–2,0</td>
<td>0,001</td>
<td>0,93</td>
<td>0,026</td>
<td>0,3</td>
</tr>
<tr>
<td>Fe</td>
<td>3,97</td>
<td>&lt;0,1</td>
<td>0,28</td>
<td>0,15</td>
<td>2,27</td>
<td>0,63</td>
</tr>
<tr>
<td>Mn</td>
<td>0,52</td>
<td>0,14</td>
<td>0,22</td>
<td>0,013</td>
<td>0,037</td>
<td>0,26</td>
</tr>
<tr>
<td>Nb</td>
<td>-</td>
<td>-</td>
<td>1,48</td>
<td>0,01</td>
<td>&lt;0,01</td>
<td>-</td>
</tr>
<tr>
<td>Si</td>
<td>0,5</td>
<td>&lt;0,01</td>
<td>0,040</td>
<td>0,040</td>
<td>0,13</td>
<td>0,25</td>
</tr>
<tr>
<td>Al</td>
<td>0,26</td>
<td>-</td>
<td>0,038</td>
<td>1,12</td>
<td>0,02</td>
<td>0,05</td>
</tr>
<tr>
<td>W</td>
<td>0,06</td>
<td>-</td>
<td>0,21</td>
<td>0,072</td>
<td>0,16</td>
<td>10</td>
</tr>
</tbody>
</table>

Experiments results in polythermal loops with redox potential measurement demonstrated that operations with Li,Be/F salt, fuelled by UF₄ or PuF₃, are feasible using carefully purified molten salts and loop internals. Corrosion rate of HN80M-VI and HN80MTY in the LiF-BeF₂+UF₄, LiF-BeF₂-ThF₄+UF₄ и LiF-NaF-BeF₂+PuF₃ melts was <5μm/yr. No intergranular corrosion of alloys is observed. Alloys modified by Ti, Al and V have shown the best post irradiation properties.
Te Corrosion in Li, Be, Th, U/F salt

The experimental facility was developed to study compatibility of Ni-based alloys under various mechanical loads to the materials specimens with fuel salts containing Cr$_3$Te$_4$ with redox potential measurement.

- LiF-BeF$_2$-ThF$_4$-UF$_4$: 5 tests of 250 hrs each at fuel salt temperature till to 750°C and $[\text{U(IV)}/[\text{U(III)}]$ ratio from 0.7 to 500
- LiF-BeF$_2$-UF$_4$: 3 tests of 250 hrs each at fuel salt temperature till to 800°C and $[\text{U(IV)}/[\text{U(III)}]$ ratio from 30 to 90

<table>
<thead>
<tr>
<th>Test</th>
<th>$[\text{UF}_3+\text{UF}_4]$ mole %</th>
<th>$[\text{U(IV)}/[\text{U(III)}]$</th>
<th>$T$ °C</th>
<th>Impurity content in the fuel salt after test, wt. %</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Ni</td>
</tr>
<tr>
<td>1</td>
<td>0.64</td>
<td>0.7</td>
<td>735</td>
<td>0.0034</td>
</tr>
<tr>
<td>2</td>
<td>2.1</td>
<td>4</td>
<td>735</td>
<td>0.0041</td>
</tr>
<tr>
<td>3</td>
<td>2.1</td>
<td>20</td>
<td>735</td>
<td>0.009</td>
</tr>
<tr>
<td>4</td>
<td>2.0</td>
<td>500</td>
<td>735</td>
<td>0.26</td>
</tr>
<tr>
<td>5</td>
<td>2.0</td>
<td>100</td>
<td>750</td>
<td>0.22</td>
</tr>
</tbody>
</table>
Te corrosion in LiF-BeF$_2$-UF$_4$

**U(IV)/(U(III))**

<table>
<thead>
<tr>
<th>Temperature</th>
<th>Alloy N</th>
<th>HN80MTY</th>
</tr>
</thead>
<tbody>
<tr>
<td>30°C, no loading</td>
<td>[Image]</td>
<td>[Image]</td>
</tr>
<tr>
<td>60°C, no loading</td>
<td>[Image]</td>
<td>[Image]</td>
</tr>
<tr>
<td>90°C, no loading</td>
<td>[Image]</td>
<td>[Image]</td>
</tr>
</tbody>
</table>

- **Alloy N**
  - U(IV)/(U(III)) = 30
  - Without loading at 760°C
  - $K = 3500 \text{pc} \times \mu\text{m/cm}; \ l = 69\mu\text{m}$

- **HN80MTY**
  - U(IV)/(U(III)) = 60
  - Without loading at 760°C
  - $K = 5300 \text{pc} \times \mu\text{m/cm}; \ l = 26\mu\text{m}$

- **HN80MTY**
  - U(IV)/(U(III)) = 90
  - Without loading at 750°C
  - $K = 4490 \text{pc} \times \mu\text{m/cm}; \ l = 148\mu\text{m}$

**Reaction**

$$2\text{UF}_3 + \text{CrF}_2 + \text{Te}^0 \rightarrow 2\text{UF}_4 + \text{CrTe} \quad (\text{U(IV)/(U(III)) < 100})$$

Reaction blocks transfer of free Te to the structural material and prevents such a corrosion of alloy.
In most cases the base-line fuel / coolant salt is lithium-beryllium fluoride salt as it has best properties.
An and Ln trifluoride solubility

Near the liquidus temperature for 78LiF-7ThF₄-15UF₄ and 72,5LiF-7ThF₄-20,5UF₄ salts, the CeF₃ significantly displace PuF₃.

<table>
<thead>
<tr>
<th>LiF</th>
<th>NaF</th>
<th>KF</th>
<th>BeF₂</th>
<th>ThF₄</th>
<th>T_melt K</th>
<th>S, mole%</th>
<th>Mode of operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>46.5</td>
<td>11.5</td>
<td>42</td>
<td>0</td>
<td>0</td>
<td>722</td>
<td>11.3</td>
<td>MA transmuter</td>
</tr>
<tr>
<td>73</td>
<td>0</td>
<td>27</td>
<td>0</td>
<td>0</td>
<td>853</td>
<td>2.1</td>
<td>TRU transmuter</td>
</tr>
<tr>
<td>15</td>
<td>58</td>
<td>0</td>
<td>27</td>
<td>0</td>
<td>752</td>
<td>1.9</td>
<td>TRU transmuter</td>
</tr>
<tr>
<td>78</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>22</td>
<td>843</td>
<td>4.0</td>
<td>Th-U breeder</td>
</tr>
<tr>
<td>77</td>
<td>0</td>
<td>17</td>
<td>6</td>
<td>870</td>
<td>3.4</td>
<td>4.0</td>
<td>Th-U breeder</td>
</tr>
</tbody>
</table>

Temperature, K | 72,5LiF-7ThF₄-20,5UF₄ | 78LiF-7ThF₄-15UF₄ |
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>PuF₃</td>
<td>CeF₃</td>
<td>PuF₃</td>
</tr>
<tr>
<td>873</td>
<td>0.35±0.02</td>
<td>1.5±0.1</td>
</tr>
<tr>
<td>923</td>
<td>4.5±0.2</td>
<td>2.5±0.1</td>
</tr>
<tr>
<td>973</td>
<td>8.4±0.4</td>
<td>3.7±0.2</td>
</tr>
<tr>
<td>1023</td>
<td>9.4±0.5</td>
<td>3.9±0.2</td>
</tr>
</tbody>
</table>

y = 0.0206x - 10.2
R² = 0.9957

y = 0.0264x - 13.25
R² = 0.9889
Fuel Salt Transport Properties

\[
\lambda = 0.928 + 8.397 \cdot 10^{-7} T + 0.054 \quad (0.78\text{LiF-0.22ThF}_4) \\
\lambda = 0.842 + 8.962 \cdot 10^{-7} T \pm 0.038 \quad (0.70\text{LiF-0.08CaF}_2-0.22\text{ThF}_4)
\]

--- 73LiF-27BeF\(_2\) --- (73LiF-27BeF\(_2\))+1CeF\(_3\)
ORNL and KI tests strongly suggested that the \( \text{F}_2 \) generation had not occurred at the high temperature (gas was generating mainly via reaction \( ^6\text{Li}(n,\alpha)^3\text{T} \)), but had occurred by radiolysis of the mixture in the solid state. \( \text{F}_2 \) evolution at 35°C corresponded to about 0.02 molecules per 100 eV absorbed, could be completely stopped by heating to 100°C or above, and could be reduced by chilling to -70°C.

<table>
<thead>
<tr>
<th>Fuel salt, mole %</th>
<th>Liquid phase</th>
<th>Solid phase</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>( T,^\circ\text{C} )</td>
<td>( G(\text{F}_2),10^{-5} \text{ mol/100eV} )</td>
</tr>
<tr>
<td>( ^{66}\text{LiF}-^{33}\text{BeF}_2-^{1}\text{UF}_4 )</td>
<td>615</td>
<td>7</td>
</tr>
<tr>
<td>( ^{69}\text{LiF}-^{31}\text{BeF}_2 )</td>
<td>680</td>
<td>2</td>
</tr>
<tr>
<td>( ^{71.7}\text{LiF}-^{16}\text{BeF}_2-^{12}\text{ThF}_4-^{0.3}\text{UF}_4 )</td>
<td>740</td>
<td>3</td>
</tr>
<tr>
<td>( ^{65.67}\text{LiF}-^{34.39}\text{BeF}_2-^{0.3}\text{UF}_4 )</td>
<td>740</td>
<td>0.2</td>
</tr>
<tr>
<td>( ^{73.6}\text{LiF}-^{25.9.2}\text{ThF}_4-^{0.5}\text{UF}_4 )</td>
<td>1200</td>
<td>2</td>
</tr>
<tr>
<td>( ^{74}\text{NaF}-^{25.9}\text{ThF}_4-^{0.9}\text{UF}_4 )</td>
<td>1150</td>
<td>0.15</td>
</tr>
</tbody>
</table>

These and subsequent experiences, including operation of the 8MWe MSRE at US ORNL, strongly indicate that radiolysis of the molten fuel at reasonable power densities is not a problem. It seems unlikely, though it is possible, that MSR fuels will evolve \( \text{F}_2 \) on cooling. If they do, arrangements must be made for their storage at elevated temperature until a fraction of the decay energy is dissipated.
Two fluid Th MOSART Flowsheet

**Core Li,Be,An/F**

**BLANKET Li,Be,Th/F**

**Fertile Stream Recycle**

- **Fertile make up**
  - NaF 600C
  - MgF₂
  - 70°C

- **Fluoride volatility**
  - 550°C
  - Make up Li,Be,Th/F

- **Reductive extraction**
  - UF₆ + Volatile FP

- **FP removal**

- **Pa conversion**

**Core Volatility Li,Be,An/F**

- **Core Volatility Li,Be,An/F**

**UF₆ recycle**

- **Sorbents**
  - NaF 600C
  - MgF₂
  - 70°C

- **Cold trap**

**UF₆ + Volatile FP**

- **AnFₙ make up and recycle to core**

**Li,Be,F addition**

**UF₆ → UF₄ reduction**

- **Valence adjustment**

**Waste storage**

- **LiF-BeF₂-AnFₙ recycle**

**FP removal**

- **Zr extraction**

- **Rare earth extraction**

- **Reduced Cr, Fe, Ni extraction**

- **Bi removal**

**Valence volatility extraction**

- **Reductive extraction**

**UF₄ → UF₆ recycle**

- **LiBe/F addition**
Reductive Extraction with Liquid Bi-Li

\( \text{LgD(Nd)} = 3\text{LgD(Li)} + 6.11 \)
\( \text{LgD(La)} = 3\text{LgD(Li)} + 5.91 \)

\( \text{D(Sm, Eu)} = 2\text{Lg D(Li)} + K'(\text{Sm, Eu}) \)
\( K'(\text{Sm}) = 3.65 \quad \text{and} \quad K'(\text{Eu}) = 3.15 \)

\( \text{LiCl at } 650^\circ \text{C} \)

\( \text{73LiF-27BeF}_2 \text{ at } 600^\circ \text{C} \)
Summary

- MSR concepts offer alternative options for new fuel breeding and long lived waste incineration with the added value of liquid fuel (intrinsic safety features, fuel cycle flexibility, simplified fuel processing, in-service inspection, no fuel transportation and refabrication required).

- Significant progress has been made on the resolution (or cancellation) of critical viability issues (material compatibility, salt physical & chemical properties, reprocessing feasibility, intrinsic safety).

- Pre-conceptual studies of the whole reactor and reprocessing unit must be performed to establish the MSR viability (reactor and fuel salt clean-up unit to be optimized together).

- Experimental infrastructures (analytical and integral salt loops with real fuel salts) are required to proceed further in the mastering of MSR technologies (e.g. tritium control) and components (long shuft pump, heat exchanger, etc.) testing.
Upcoming Webinars

30 July 2018  Astrid – Lessons Learned  Dr. Frederic Varaine, CEA, France

22 August 2018  BREST-300 Lead-Cooled Fast Reactor  Dr. Valery Rachkov, IPPE, Russia

26 September 2018  Advanced Lead Fast reactor European Demonstrator – ALFRED project  Dr. Alessandro Alemberti, Ansaldo Nucleare, Italy