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

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Meet the presenter

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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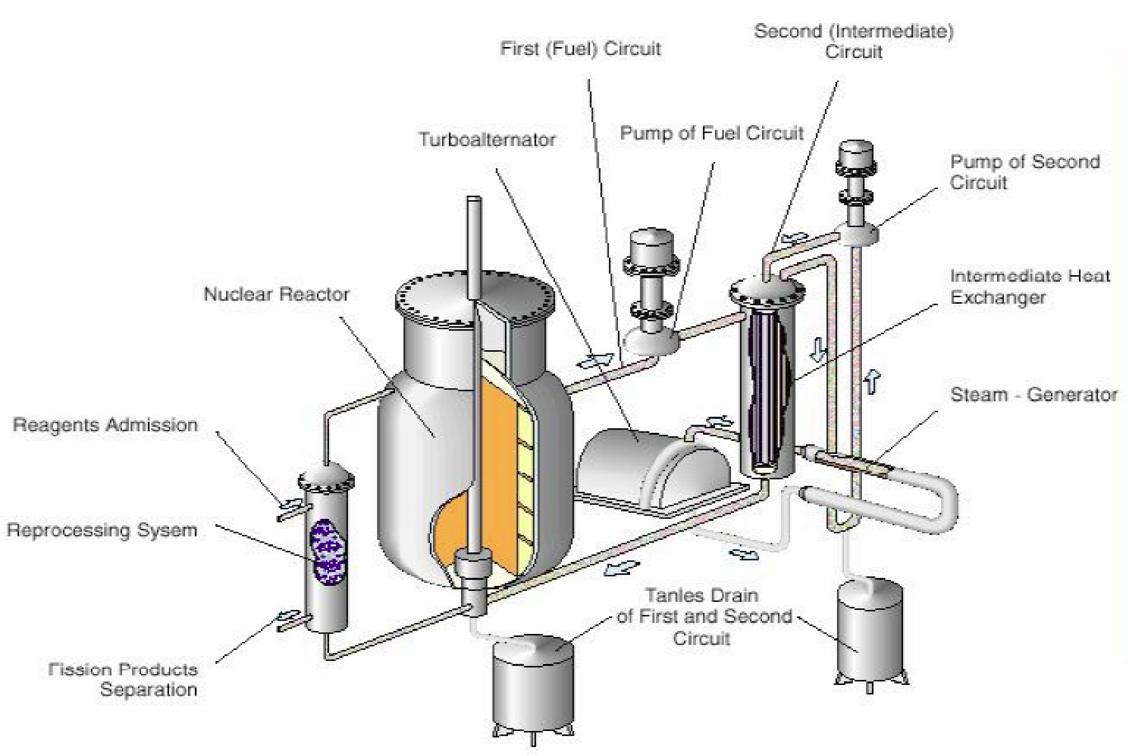
Contents

- 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



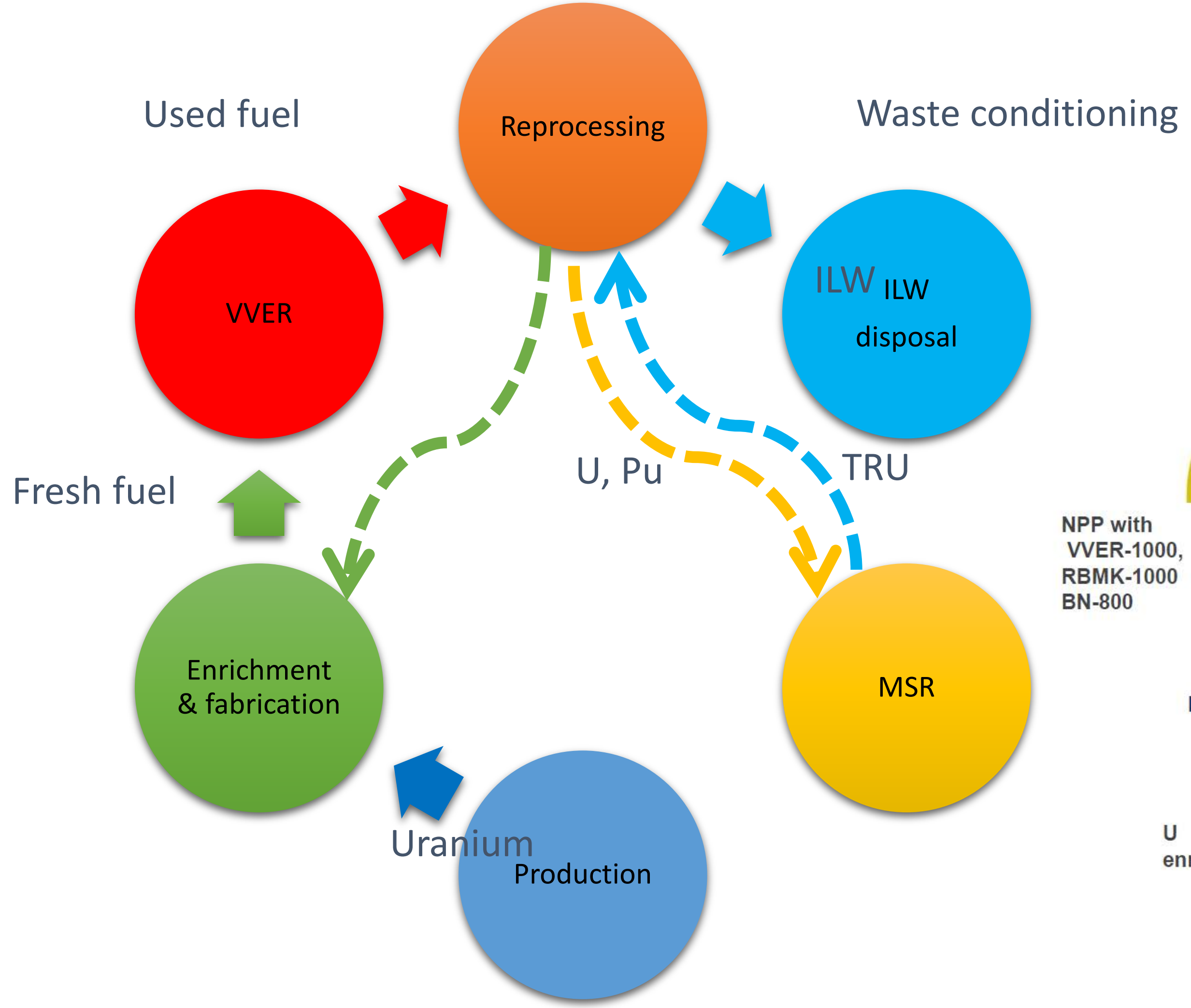
Mechanical engineering device presumes that the fuel (solid) has to be used in a max condensed form that excludes reprocessing and has advantage of technical simplicity while reactor operating.

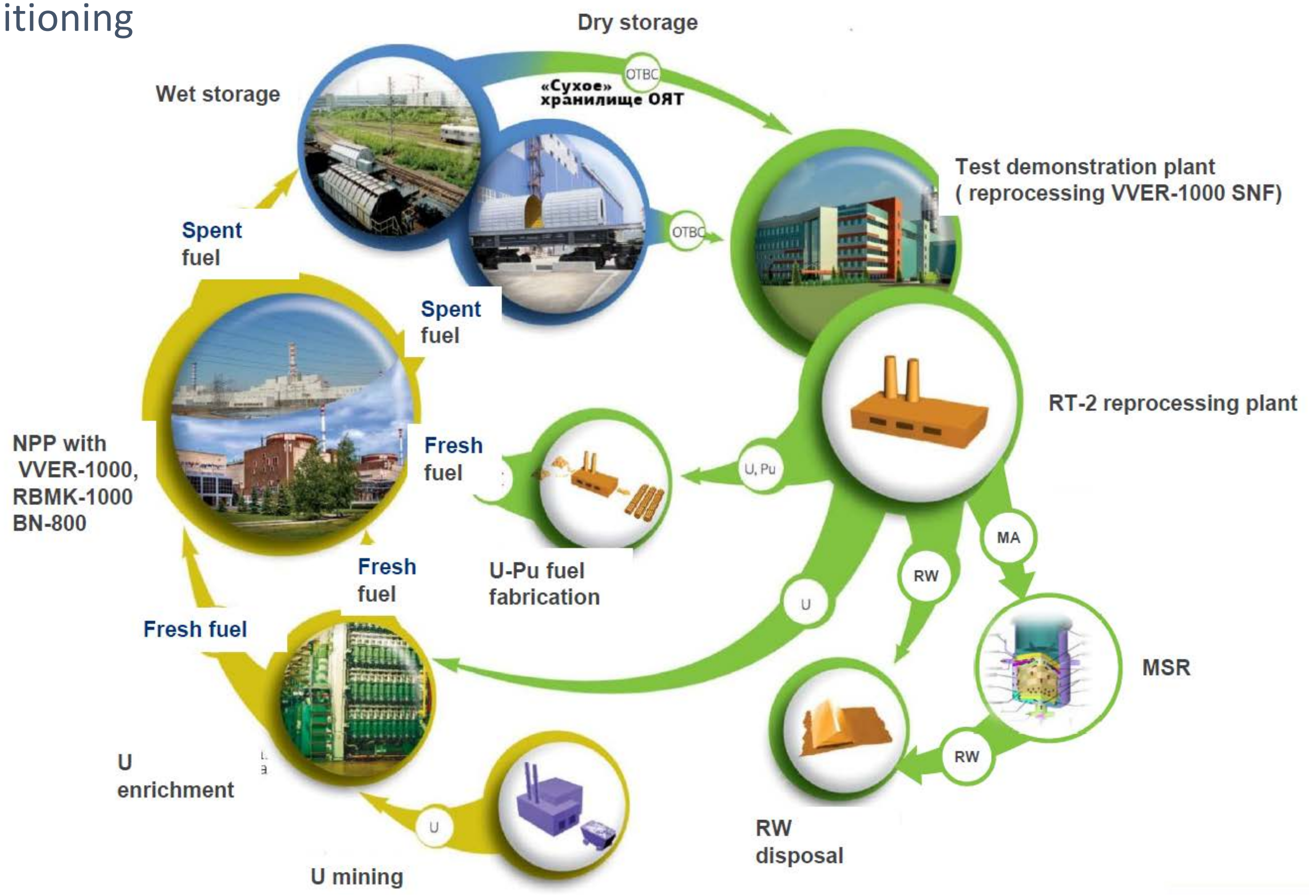
Chemical engineering device has not only possibilities of general benefits such as unlimited burn-up, easy and relatively low cost of purifying and reconstituting the fuel (fluid), but also there are some more specific potential gains.

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

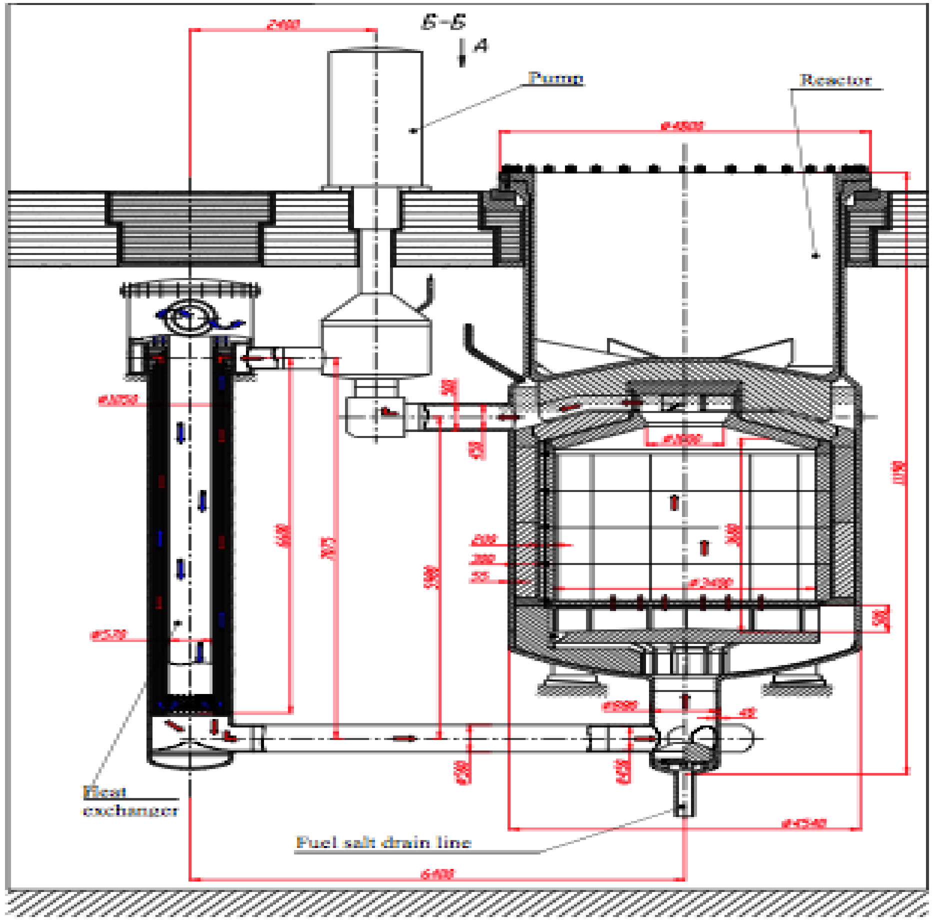




MOSART –Transforming Reactor System

HN80MTY
Circuits, Vessel, Heat exchangers *600 / 720° C* Creep, Creep-fatigue, Thermal fatigue, Aging, Welds…

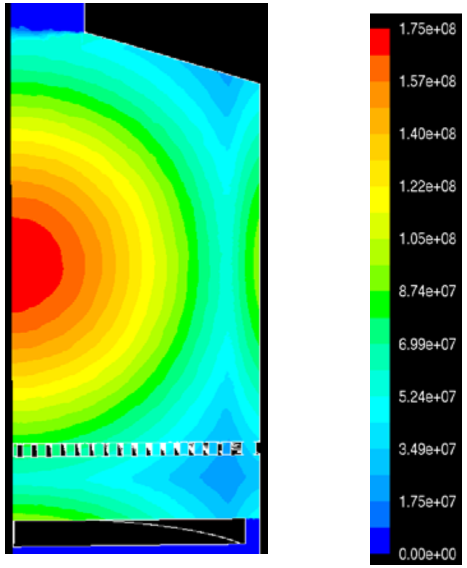
HN80MTY
Intermediate circuit - *455 / 620° C* Aging, Welds, Compatibility NaF-NaBF$_4$, Oxidation, Wastage…



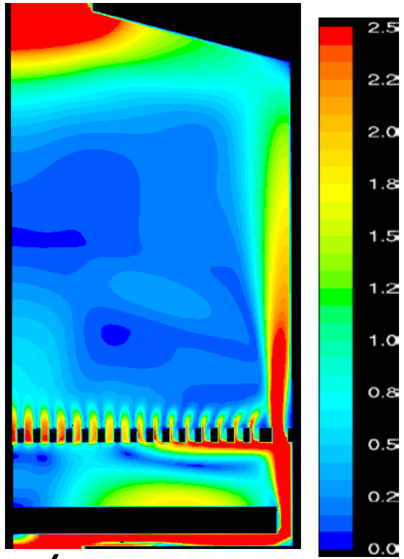
System	burner /	breeder
Fluid streams	1	2
Power capacity, MWt	2400	2400
Fuel salt inlet/outlet temperature, °C	600 /720	600 /720
Fuel salt composition, mole %	72LiF 27BeF$_2$ 1TRUF$_3$	75LiF 16.5BeF$_2$ 6ThF$_4$ 2.5TRUF$_3$
Blanket salt composition, mole %	no	75LiF 5BeF$_2$ 20ThF$_4$

- <u>Max temperature</u> of the fuel salt in the primary circuit made of special Ni- alloy is mainly limited by Te IGC depending on salt Redox potential
- <u>Min temperature</u> of fuel salt is determining not only its melting point, but also the solubility for AnF$_3$ in the solvent for this temperature

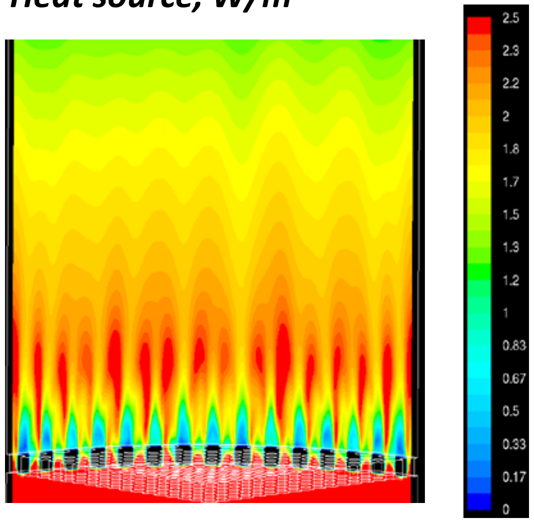
MOSART –Transforming Reactor System

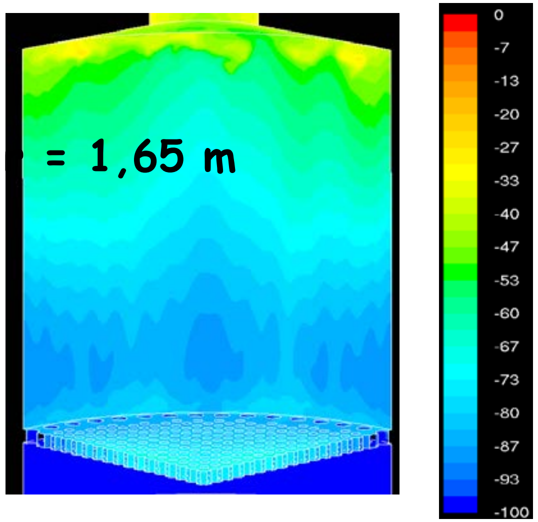


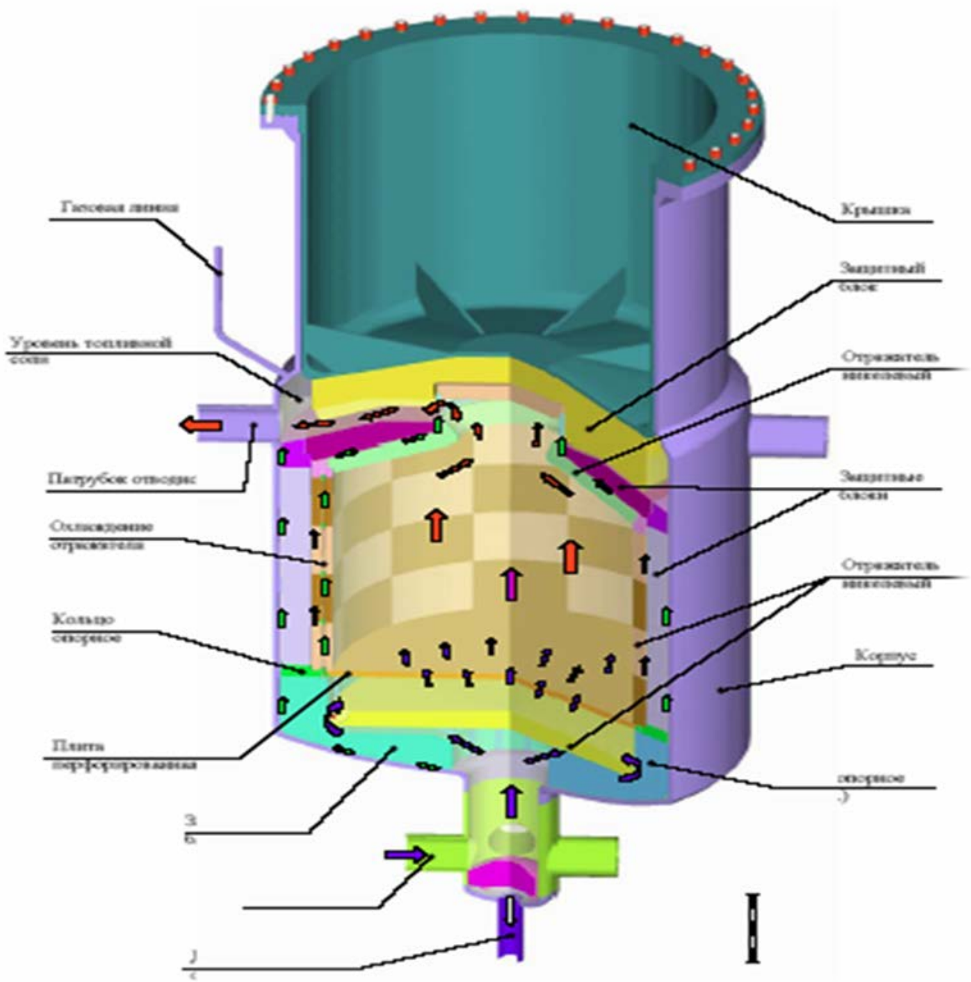
Heat source, W/m^3



Velocity, m/s

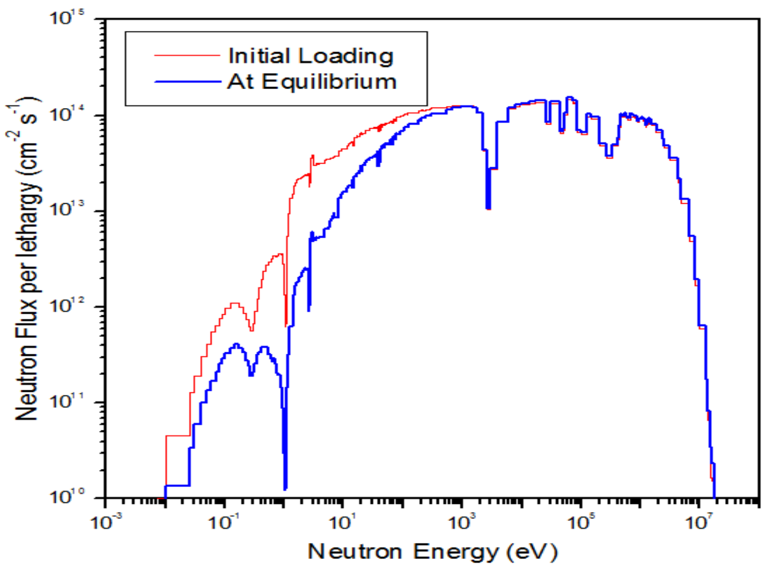






T max wall: 947 K; *T max fluid : 1192 K*

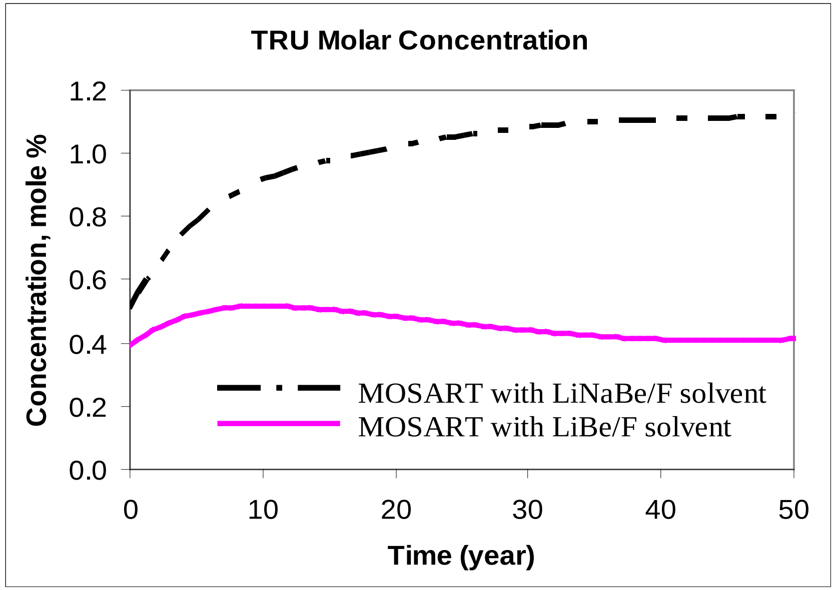
MOSART Fuel Cycles

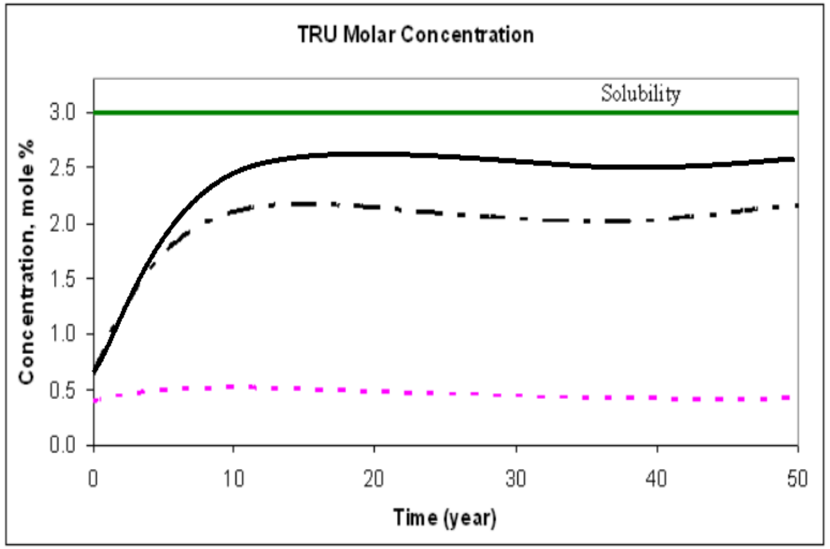


Solvent, mole %	Feed MA/TRU	Loading (EOL), t	TRU/MA, kg/yr
15LiF-58NaF-27BeF_2	0.1	7.7	730/73
73LiF-27BeF_2	0.1	3.9	730/73
73LiF-27BeF_2	0.35	13.9	730/260
73LiF-27BeF_2	0.45	23.2	730/330

Contribution to Keff

Isotope	BOL	EOL
Pu 239	0,75	0,25
Pu 238	-	-
Pu 241	0,25	0,41
Cm 245	-	0,28

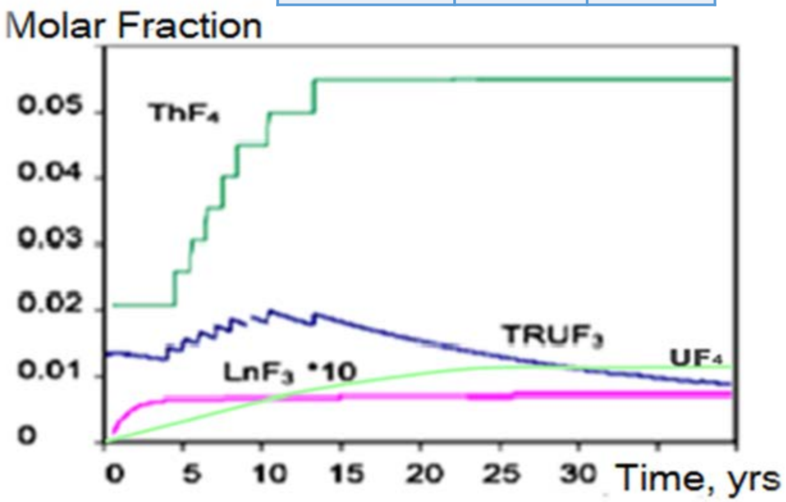




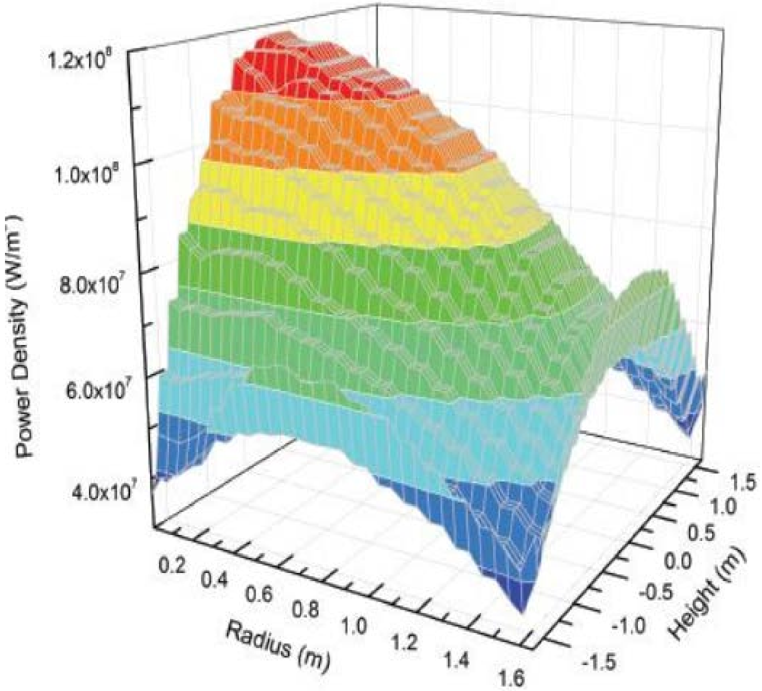
Feeding composition (Np+MA)/TRU=0.35

Feeding composition (Np+MA)/TRU=0.1

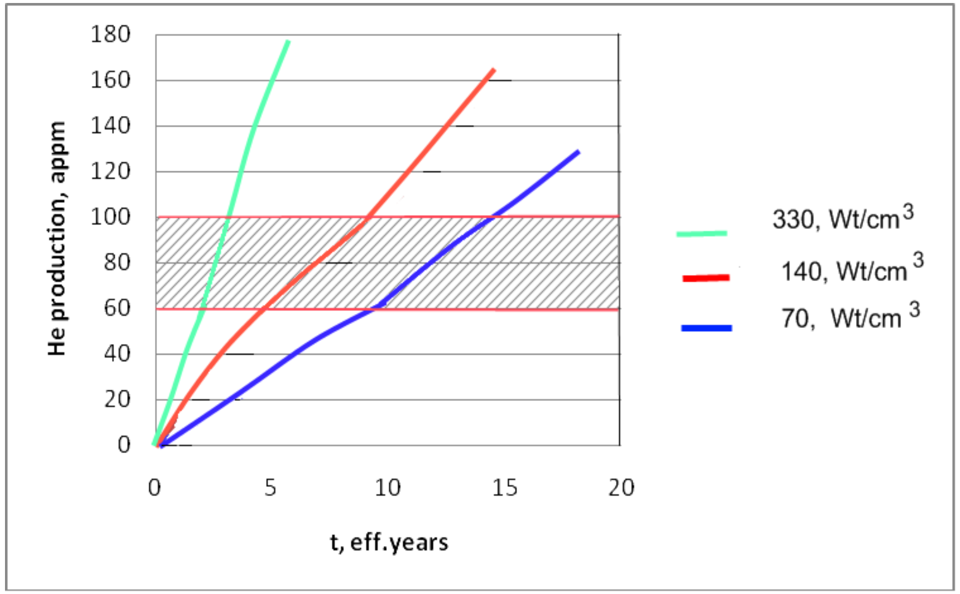
Feeding composition (Np+MA)/TRU=0.45 (with U-233 recycling from blanket to fuel stream).



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 / cm^3



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



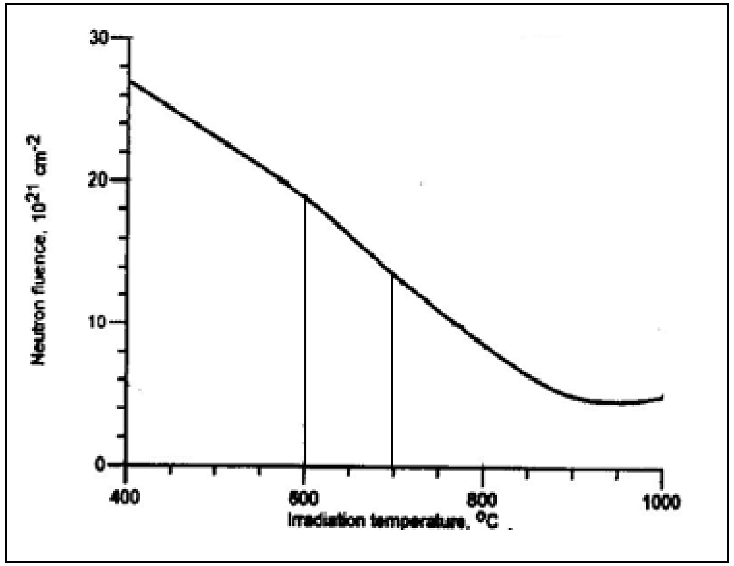
He embrittlement for Ni-base alloy at T > 500°C

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

$^{60}Ni + n \rightarrow {}^{57}Fe + {}^{4}He$;

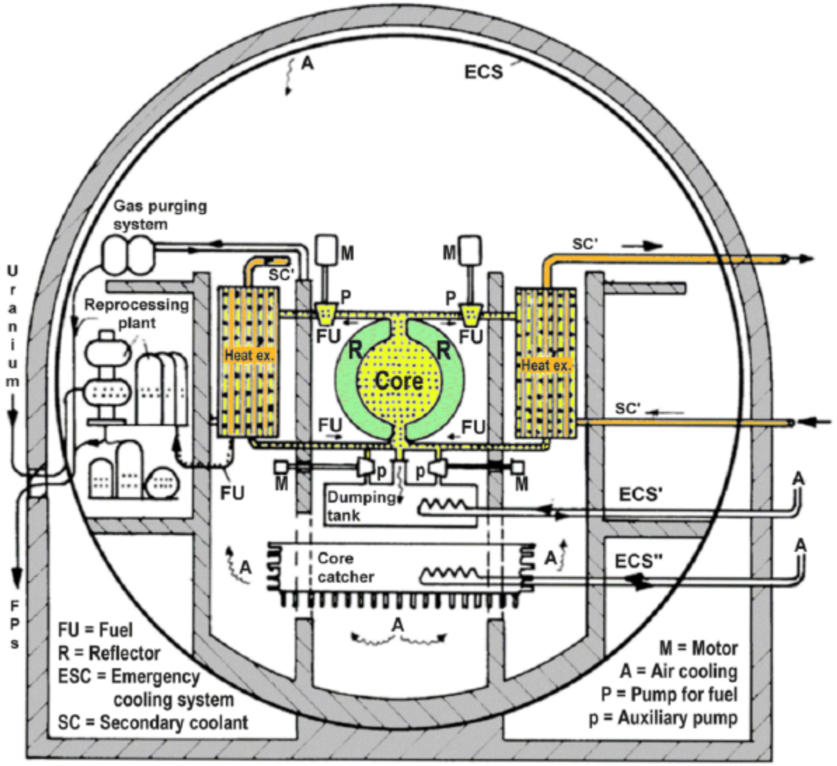
$^{10}B + n \rightarrow {}^{7}Li + {}^{4}He$.

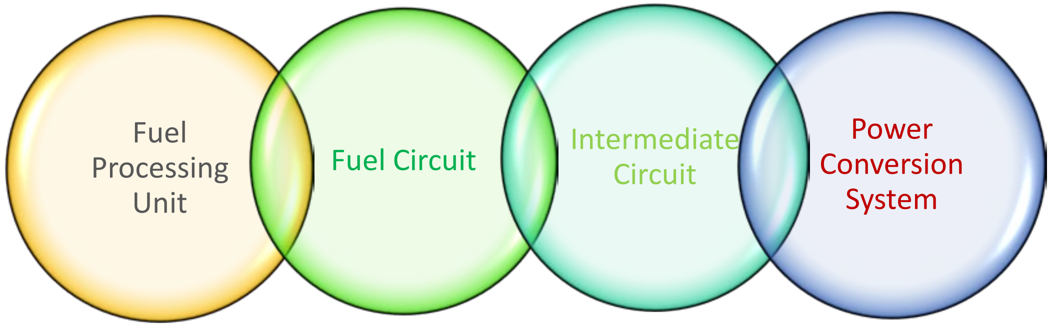
$^{58}Ni + n \rightarrow {}^{59}Ni + \gamma$, $^{59}Ni + n \rightarrow {}^{56}Fe + {}^{4}He$.



Basing on neutron fluence ($3{,}8 \cdot 10^{21} n/(cm^2 yr)$) and temperature (860-1000K) reflector should be changed in 5 yrs

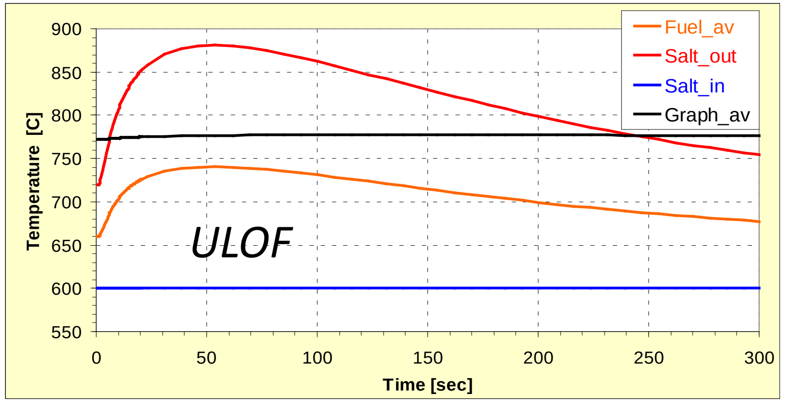
MSR Engineered Safety Features

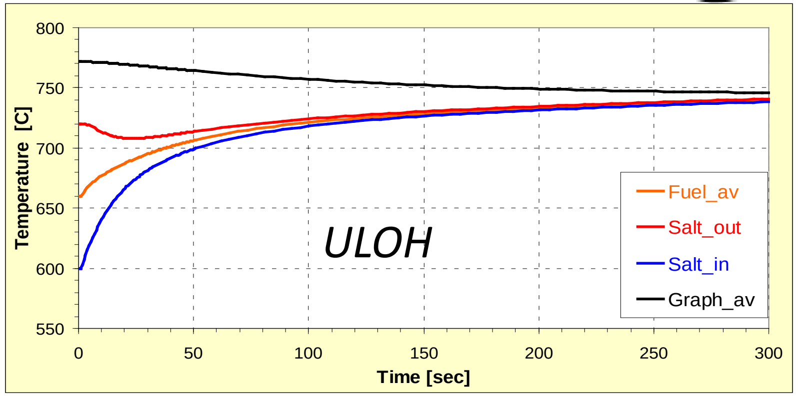


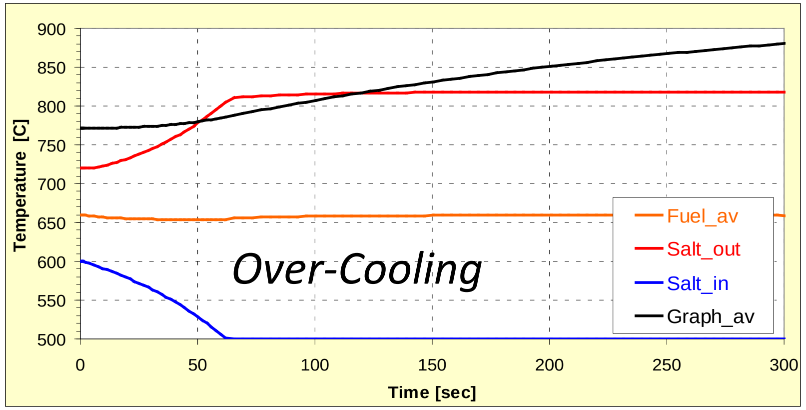


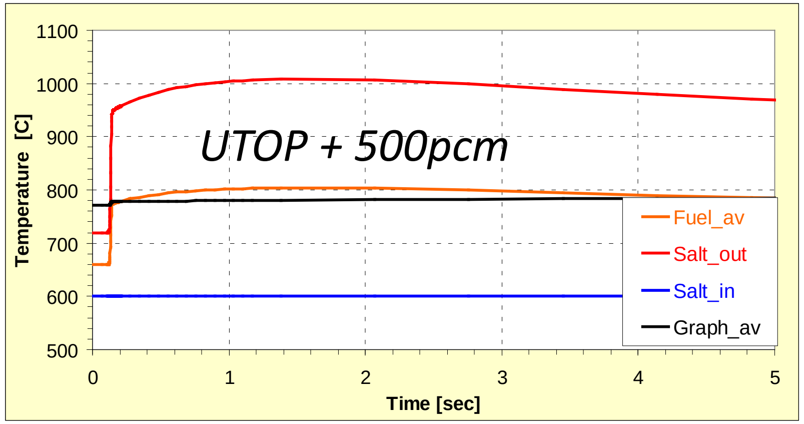
- Nuclear fuel is a fluid. It circulates throughout the RCS, transfers heat to heat exchanger and becomes critical only in core
- Possible initiators of RCS breach accident: pipe failure missiles, and pressure or temperature transients in RCS, failure of the boundary between the 1st and 2nd salts in 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

MOSART Transients Analysis









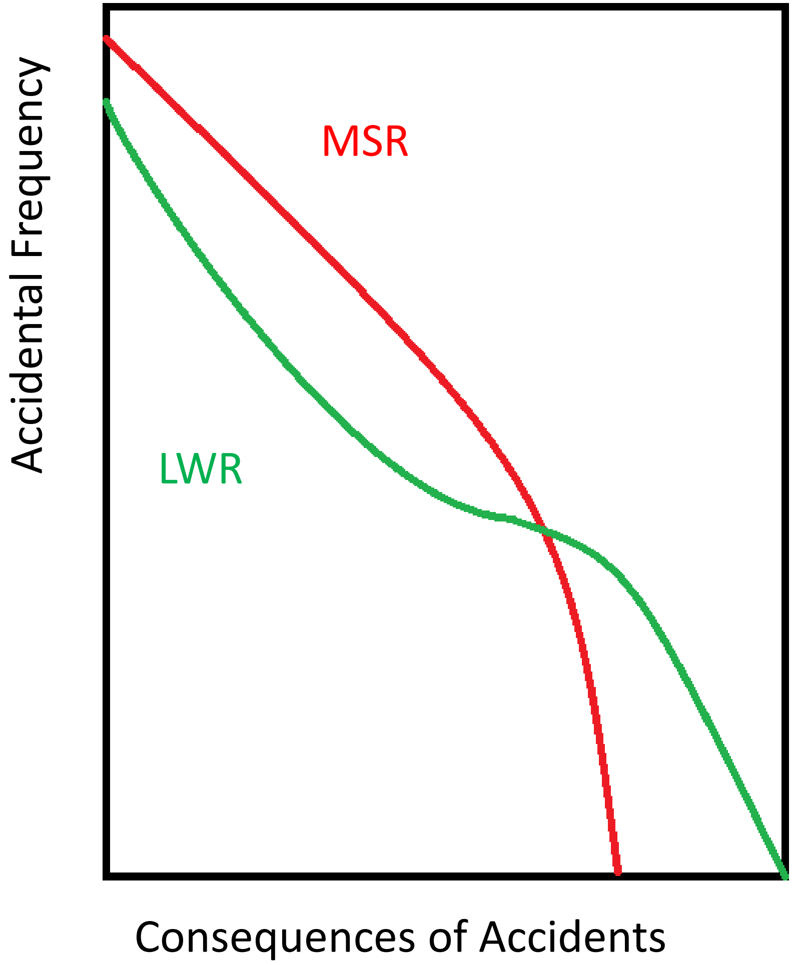
The MOSART is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP. The system was shown to buffer reactivity insertion of up to + 1.5$. System temperatures are expected to rise only ~300C above nominal under this severe transient 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_0)
- 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 I - 2 orders of magnitude compared to LWR), though for several particular nuclides such I^{131} and I^{133} the differences are smaller

Isotope	A_s/A_o	A_g/A_s	A_g/A_o
Te129	0,25	1	0,25
Te132	0,005	1	0,005
Ru103	0,01	1	0,01
Ru106	0,001	1	0,001
Nb95	0,034	1	0,034
Zr95	0,99	0,0011	0,0011
Sr89	0,99	0,00046	0,00046
Sr90	0,98	0,00046	0,00046
La140	0,98	0,026	0,025
Ce141	0,99	0,0024	0,023
Ce144	0,96	0,0024	0,023
I131	0,62	0,43	0,27
I133	0,94	0,43	0,43
Cs137	0,7	0,016	0,011

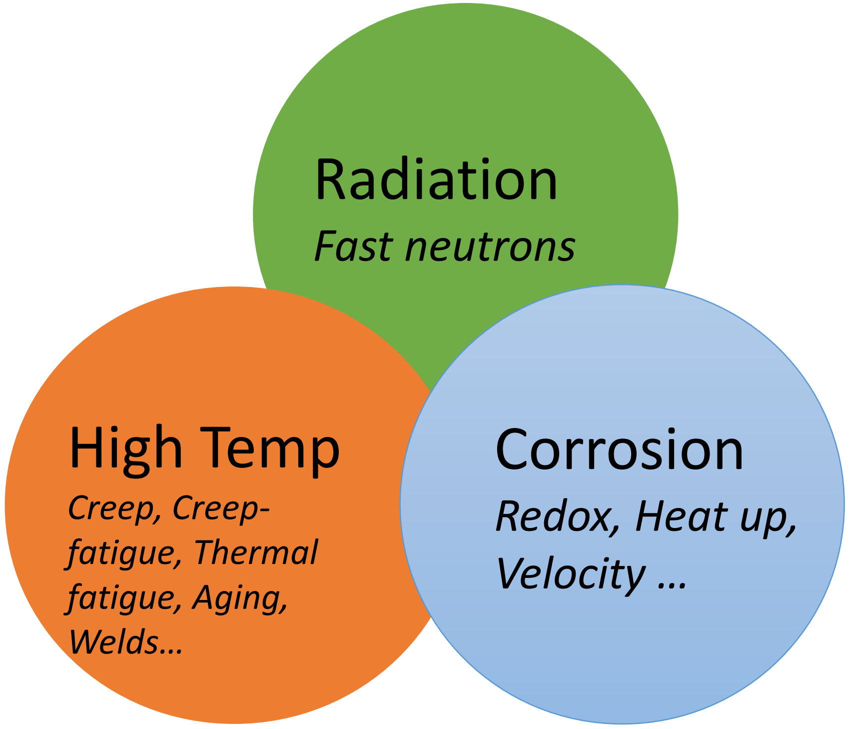
Frequency Distribution for the Probability of Accidents in the MSR



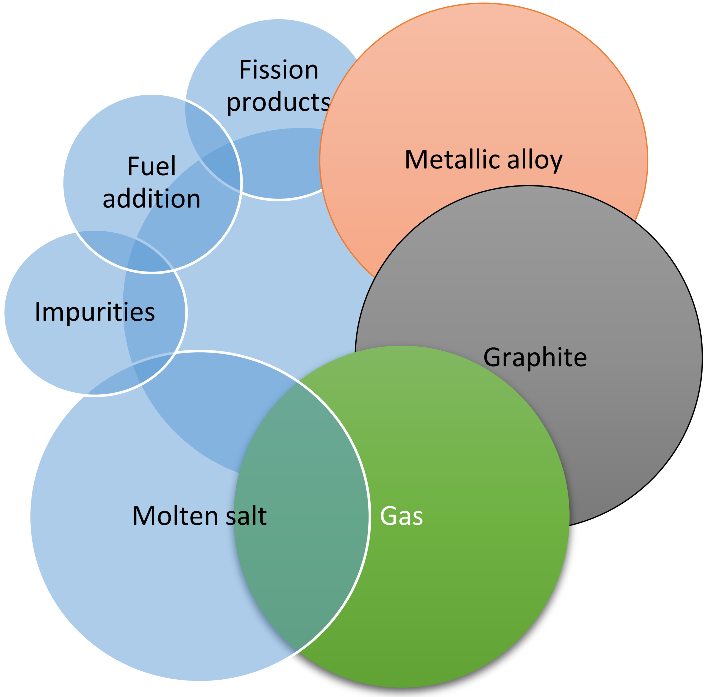
- Probability of an accident with a relatively low impact for MSR is higher than for LWR. This is due to the possibility of leakage of radioactive liquid fuel in case of accidents in the pump, piping, valves.
- The consequences of severe accidents in particular, leading to the release of radioactive 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



Corrosion effects

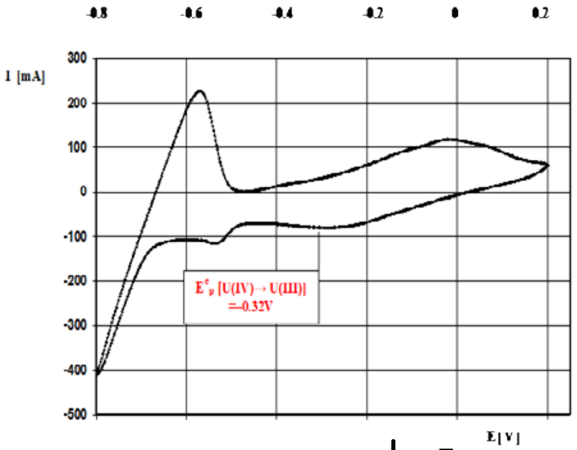


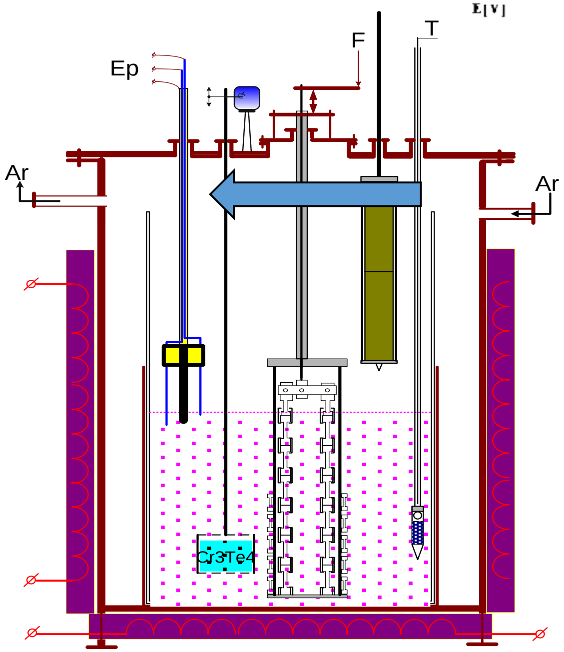
In temperature range 500-800^{0}C about 70 differently alloyed specimens of HN80MT were tested. Among alloying elements there were W, Nb, Re, V, Al and Cu

Element	Hasteloy N US	Hasteloy NM US	HN80M-VI Russia	HN80MTY Russia	MONICR Czech Rep	E-721 France
Ni	base	base	base	base	base	base
Cr	7,52	7,3	7,61	6,81	6,85	8
Mo	16,28	13,6	12,2	13,2	15,8	0.7
Ti	0,26	0,5–2,0	0,001	0,93	0,026	0.3
Fe	3,97	< 0,1	0,28	0,15	2,27	0.63
Mn	0,52	0,14	0,22	0,013	0,037	0.26
Nb	-	-	1,48	0,01	< 0,01	-
Si	0,5	< 0,01	0,040	0,040	0,13	0.25
Al	0,26	-	0,038	1,12	0,02	0.05
W	0,06	-	0,21	0,072	0,16	10

Experiments results in polythermal loops with redox potential measurement demonstrated that operations with Li,Be/F salt, fuelled by UF_4 or PuF_3, are feasible using carefully purified molten salts and loop internals. Corrosion rate of HN80M-VI and HN80MTY in the LiF-BeF_2+UF_4, LiF-BeF_2-ThF_4+UF_4 и LiF-NaF-BeF_2+PuF_3 melts was <5μm/yr. No intergranular corrosion of alloys is observed. Alloys modified b y Ti, Al and V have shown the best post irradiation properties

Te Corrosion in Li,Be,Th,U/F salt

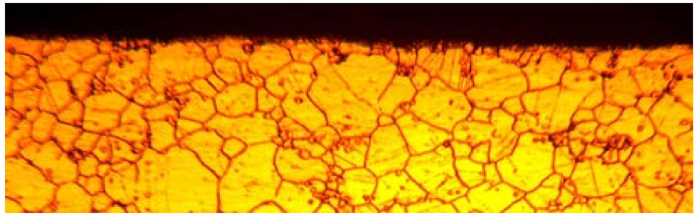
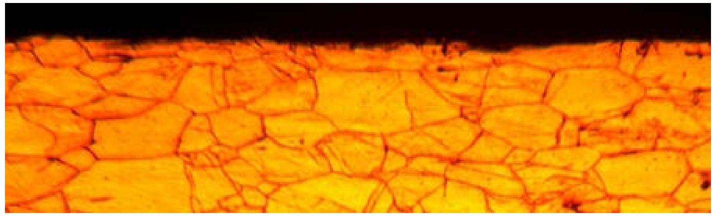
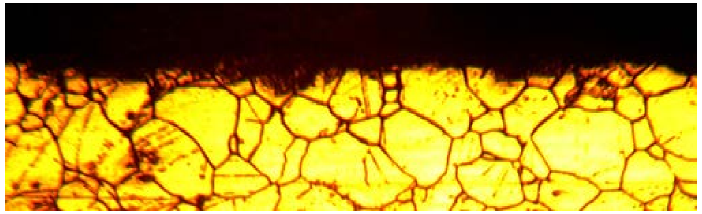
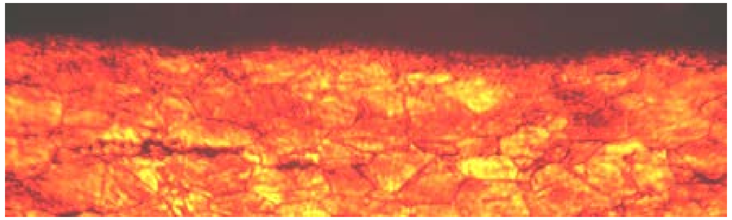
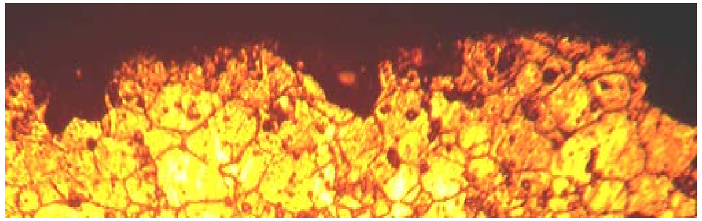
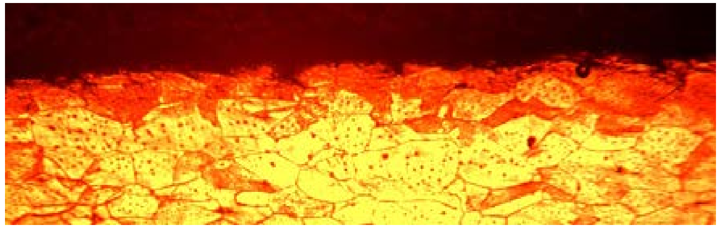


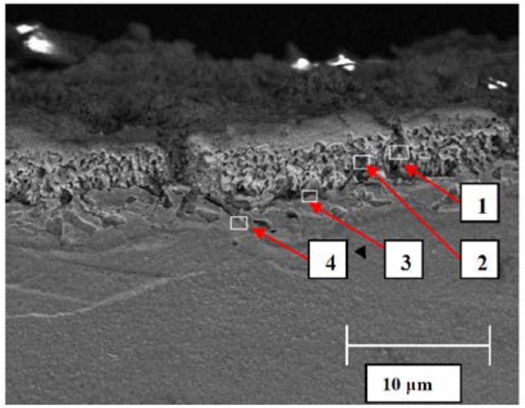


Test	$[UF_3+UF_4]$ mole %	[U(IV)]/[U(III)]	T °C	Impurity content in the fuel salt after test, wt. % Ni	Cr	Fe	Cu	Te
1	0.64	0.7	735	0.0034	0.0018	0.054	0.002	**0.015**
2	2.1	4	735	0.0041	0.0019	0.006	0.0012	**0.0032**
3	2.1	20	735	0.009	0.0055	0.003	0.001	**0.015**
4	2.0	500	735	0.26	0.024	0.051	0.019	**0.013**
5	**2.0**	**100**	**750**	**0.22**	**0.031**	**0.065**	**0.055**	**0.034**

- 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_3Te_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 [U(IV)]/[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 [U(IV)]/[U(III)] ratio from 30 to 90

Te corrosion in LiF-BeF_2-UF_4

U(IV)/(UIII)	Alloy N	HN80MTY
30 without loading at 760°C	no 	no 
60 without loading at 760°C	K = 3500pc × µm/cm; l = 69µm 	no 
90 without loading at 800°C	K = 4490pc × µm/cm; l = 148µm 	K = 530pc × µm/cm; l = 26µm 



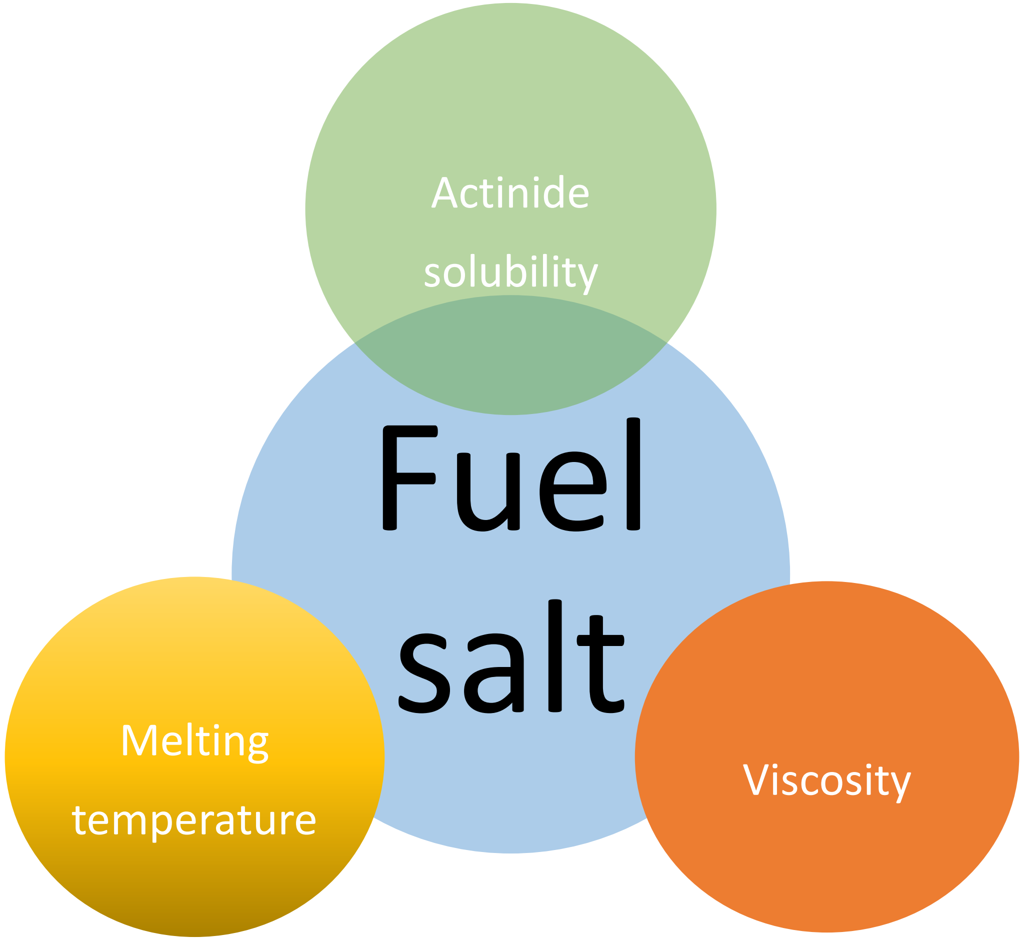
Element/ wt%	1	2	3	4
C	37	26	-	-
Ni	19	25	69	66
Al	2	3	2	1
Si	1	3	-	-
Mo	15	17	11	9
U	1	1	-	-
Te	4	3	-	-
Ti	-	1	-	8
Cr	21	22	18	17

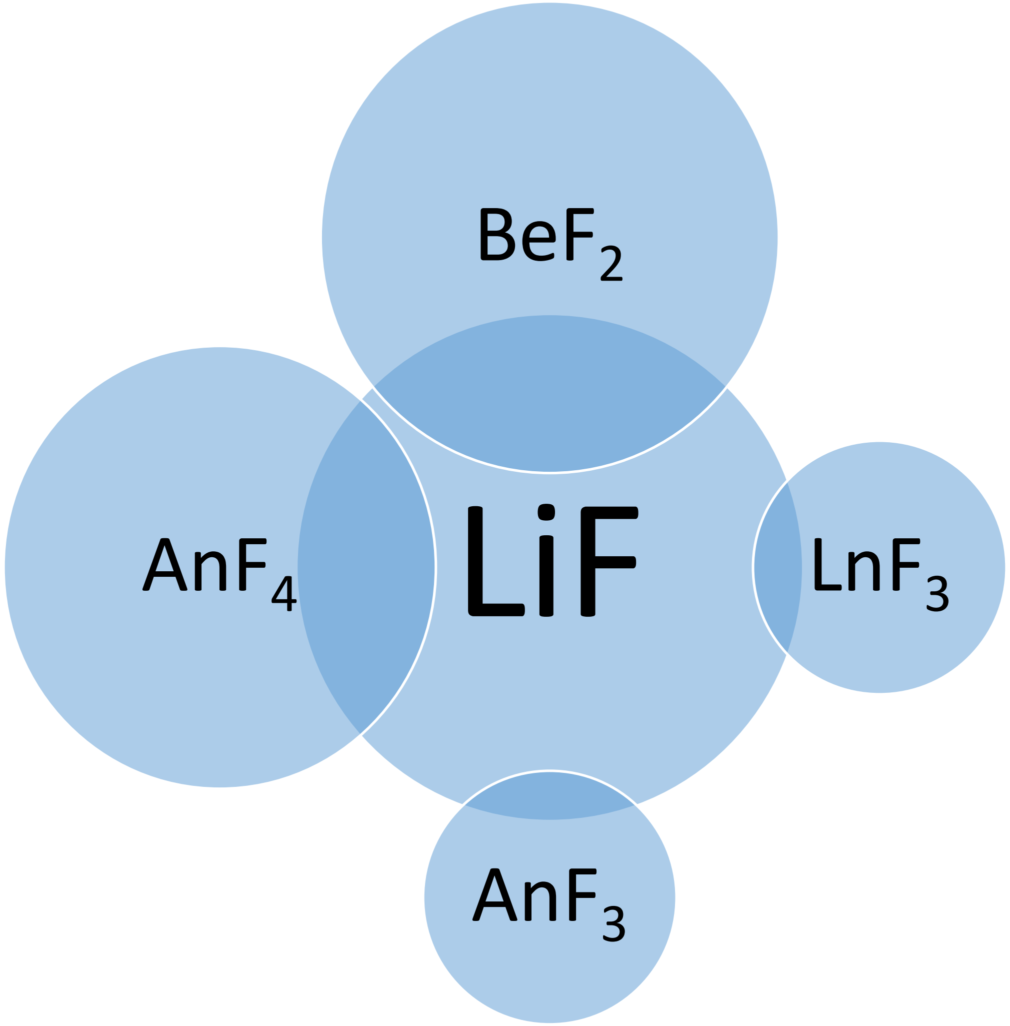
HN80MTY U(IV)/(UIII)=90 without loading at 750°C

$$2UF_3 + CrF_2 + Te^0 \rightarrow 2UF_4 + CrTe$$ (U(IV)/(UIII) <100)

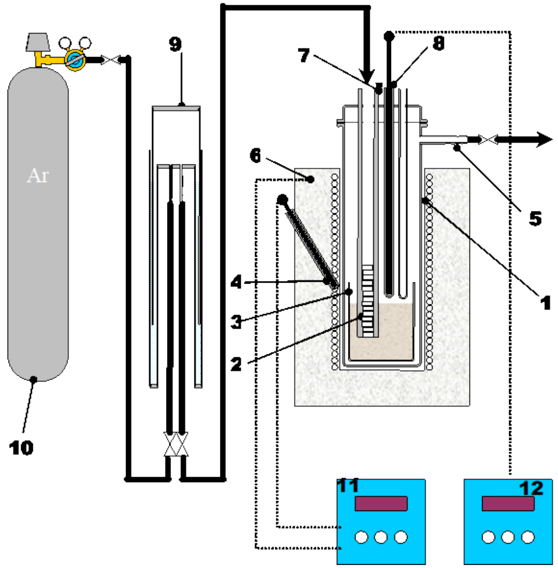
Reaction blocks transfer of free Te to the structural material and prevents such a corrosion of alloy

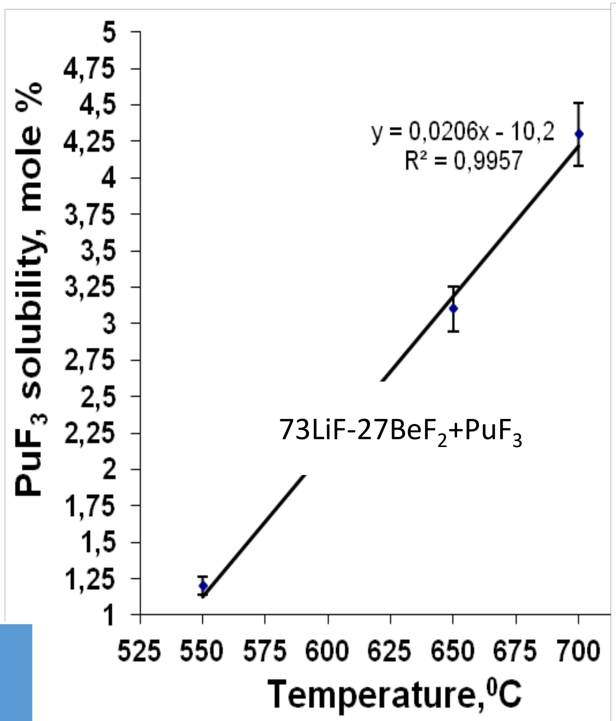
In most cases the base-line fuel / coolant salt is lithium-beryllium fluoride salt as it has best properties

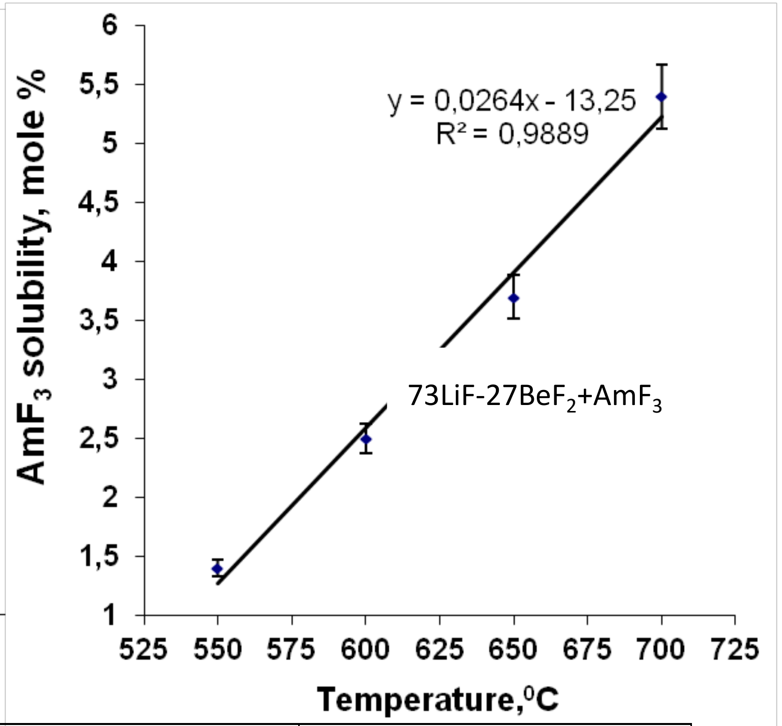




An and Ln trifluoride solubility





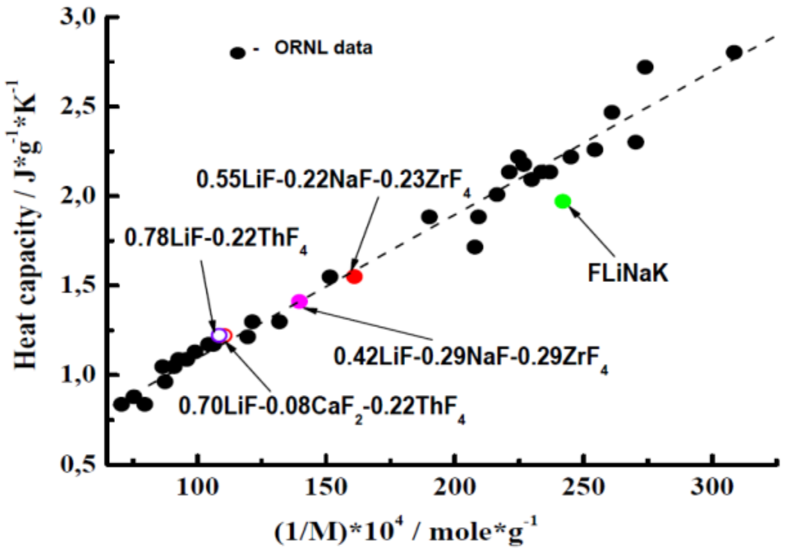


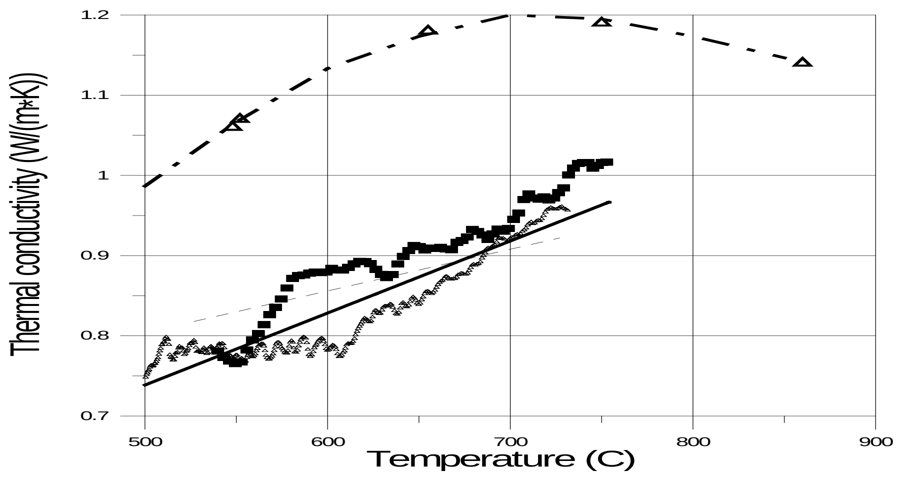
LiF	NaF	KF	BeF_2	ThF_4	T_{melt}, K	S, mole%		Mode of operation
						873K	923K	
46.5	11.5	42	0	0	722	11.3	21.5	MA transmuter
73	0	0	27	0	853	2.1	3.1	TRU transmuter
15	58	0	27	0	752	1.9	2.9	TRU transmuter
78	0	0	0	22	843	4.0	5.2	Th-U breeder
77	0	0	17	6	870	3.4	4.0	Th-U breeder

Temperature, K	72,5LiF-7ThF_4-20,5UF_4		78LiF-7ThF_4-15UF_4	
	PuF_3	CeF_3	PuF_3	CeF_3
873	0,35±0,02	1,5±0,1	1,45±0,7	2,6±0,1
923	4,5±0,2	2,5±0,1	5,6±0,3	3,6±0,2
973	8,4±0,4	3,7±0,2	9,5±0,5	4,8±0,3
1023	9,4±0,5	3,9±0,2	10,5±0,6	5,0±0,3

Near the liquidus temperature for 78LiF-7ThF_4-15UF_4 and 72,5LiF-7ThF_4-20,5UF_4 salts, the CeF_3 significantly displace PuF_3

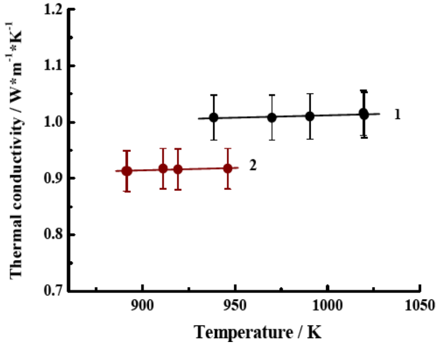
Fuel Salt Transport Properties

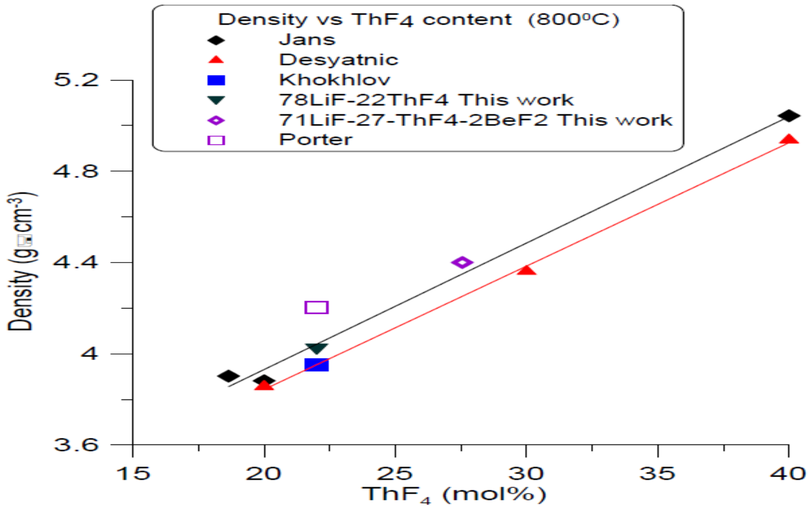


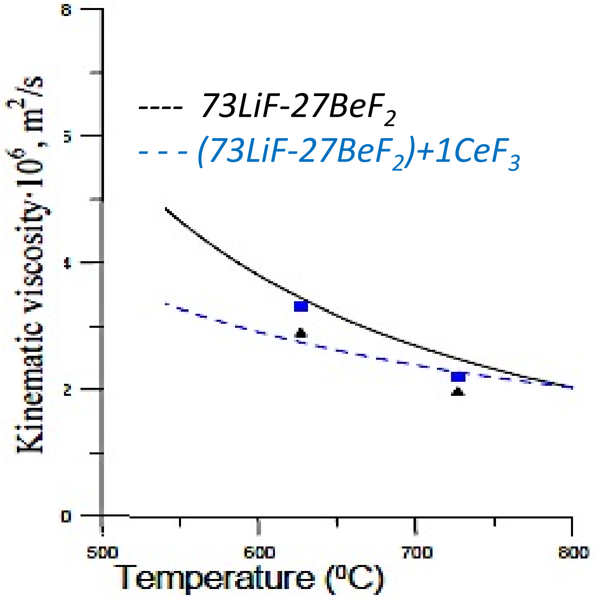


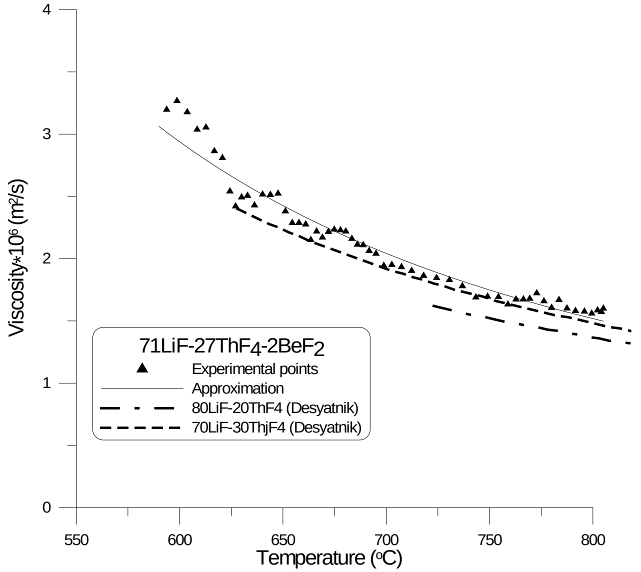
$\lambda = 0.928 + 8.397 \cdot 10^{-5} T \pm 0.054$ (0.78LiF - 0.22ThF4) и

$\lambda = 0.842 + 8.062 \cdot 10^{-5} T \pm 0.038$ (0.70LiF-0.08CaF2-0.22ThF4).



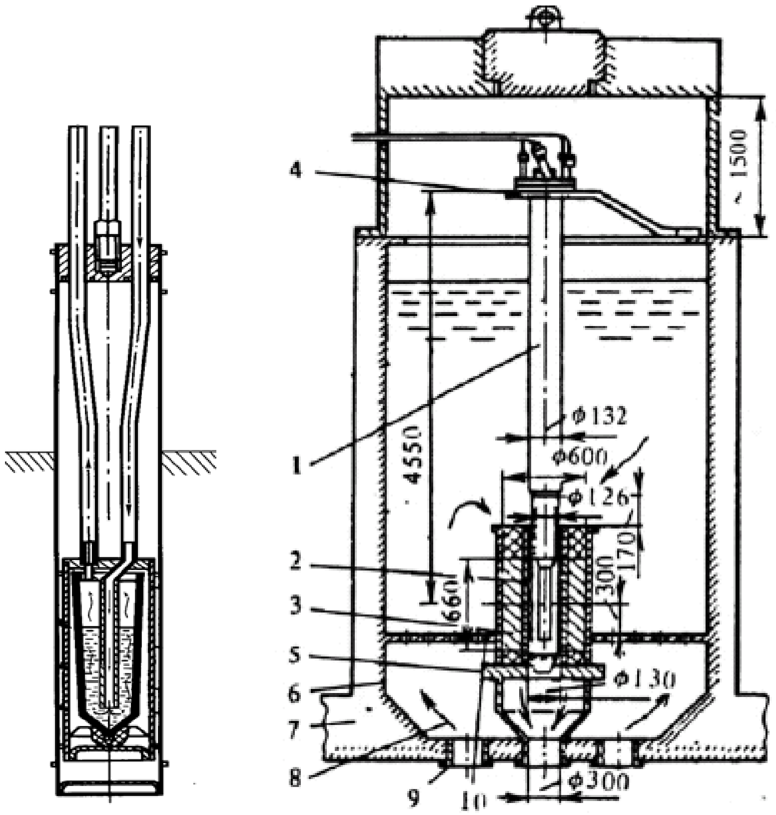






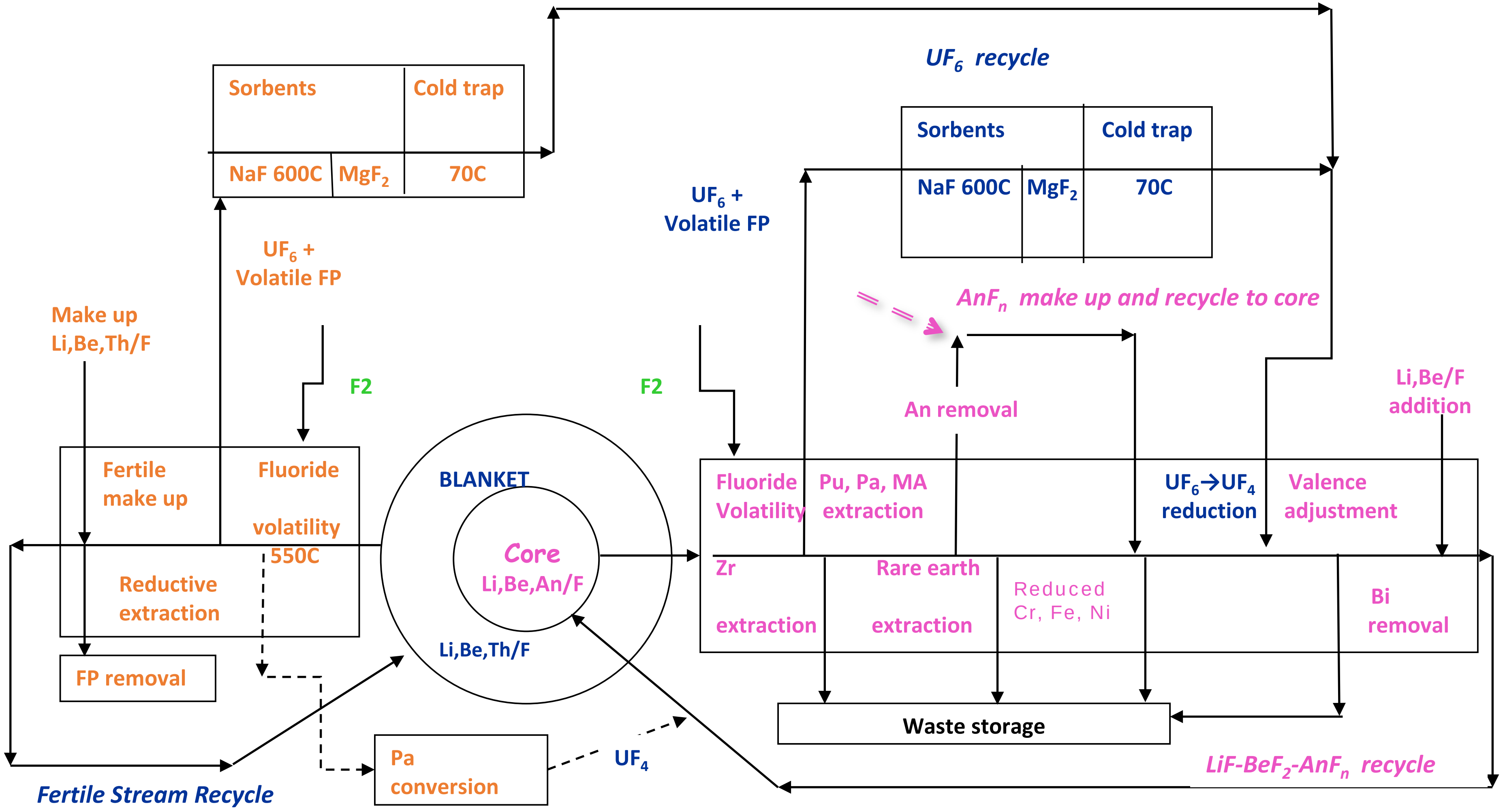
ORNL and KI tests strongly suggested that the F_2 generation had not occurred at the high temperature (gas was generating mainly via reaction $^{6}Li(n,\alpha)T$), but had occurred by radiolysis of the mixture in the solid state. F_2 evolution at 35°C corresponded to about 0,02 molecules per 100 eV absorbed, could be completely stopped by heating to 100°C or above, and could be reduced by chilling to -70°C.

KI tests Fuel salt, mole %	Liquid phase		Solid phase	
	T,°C	$G(F_2)$, 10^{-5} mol/100eV	T,°C	$G(F_2)$, 10^{-2} mol/100eV
66LiF-33BeF$_2$-1UF$_4$	615	7	50	1
69LiF-31BeF$_2$	680	2		0.2
71.7LiF-16BeF$_2$-12ThF$_4$-0.3UF$_4$	740	3	25	0.6
65.6^{7}LiF-34.39BeF$_2$-0.3UF$_4$	740	0.2	25	-
73.6iF-25.9.2ThF$_4$-0.5UF$_4$	1200	2	-	2.5
74NaF-25.9ThF$_4$-0.9UF$_4$	1150	0.15	50	2

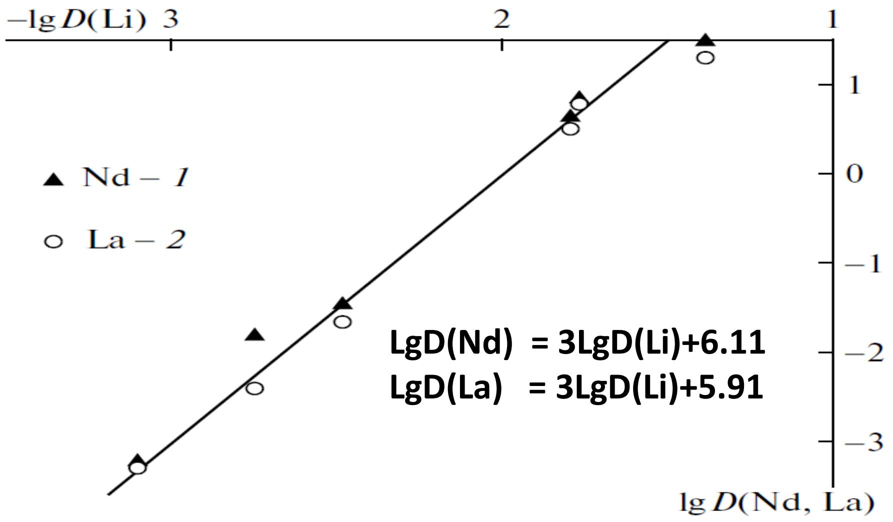


These and subsequent experiences, including operation of the 8MWe MSRE at US ORNL, strongly indicate that radiolysis of the molten fuel at reasonable power densities is not a problem. It seems unlikely, though it is possible, that MSR fuels will evolve F_2 on cooling. If they do, arrangements must be made for their storage at elevated temperature until a fraction of the decay energy is dissipated

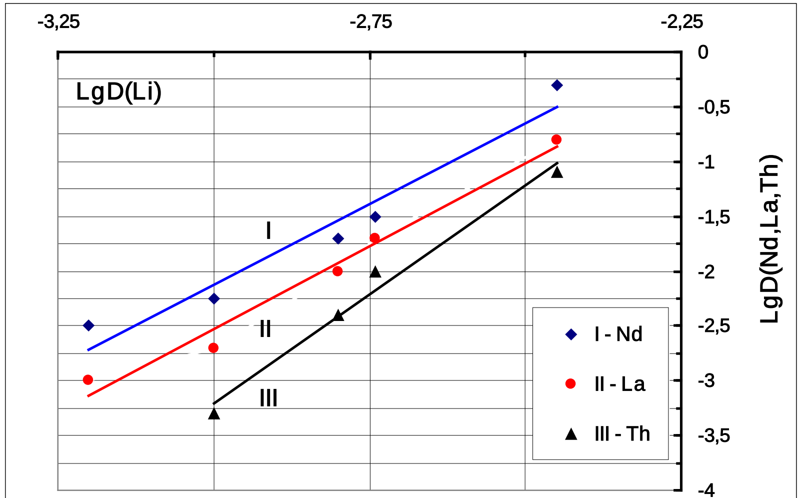
Two fluid Th MOSART Flowsheet



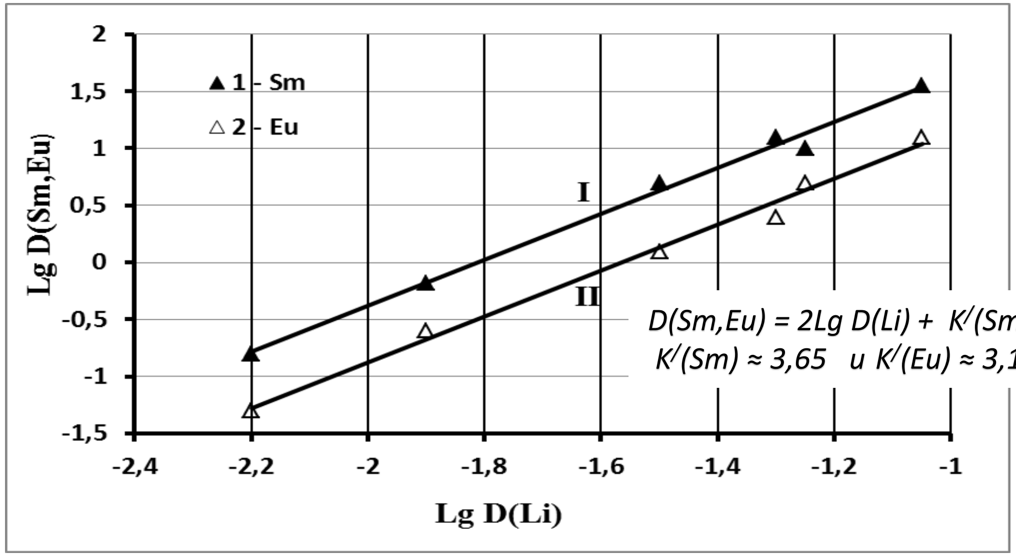
Reductive Extraction with Liquid Bi-Li



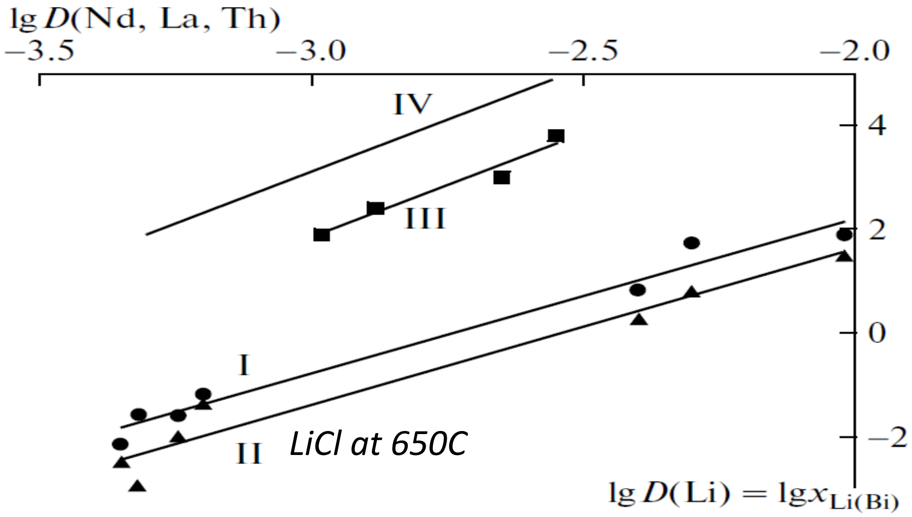
73LiF-27BeF$_2$ at 600C



74LiF-22BeF$_2$-4ThF$_4$ at 650C



D(Sm,Eu) = 2Lg D(Li) + K$^{/}$(Sm,Eu)
K$^{/}$(Sm) ≈ 3,65 u K$^{/}$(Eu) ≈ 3,15.



LiCl at 650C

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