Molten Salt Reactor Workshop, PSI, Switzerland, 24 January 2017

"Developing the Next Generation of Molten Salt Reactor Systems in Russian Federation"

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Introduction

From 1976 MSR study in Russia was organized around the following issues:

- *exploration of possible use and niches for MSR concepts*
  - *Efficient electricity production in Th-U Converter / Breeder designs*
  - *Consumption of TRU's while extracting their energy*
  - *High temperature Fluoride Salt Cooled Reactor*
  - *Fusion hybrid blankets*
  - *Isotopes production for medicine*
- *reactor physics, thermal hydraulics, fuel cycles and safety*
- *container materials for fuel and coolant salts*
- *physical and chemical properties of molten salt mixtures*
- *heat transfer and hydraulics of fuel and coolant salts*
- *handling and circulation of fuel and coolant salts*
- *process and radiochemical tests of model installations*
- *radiation chemistry of fuel salt*

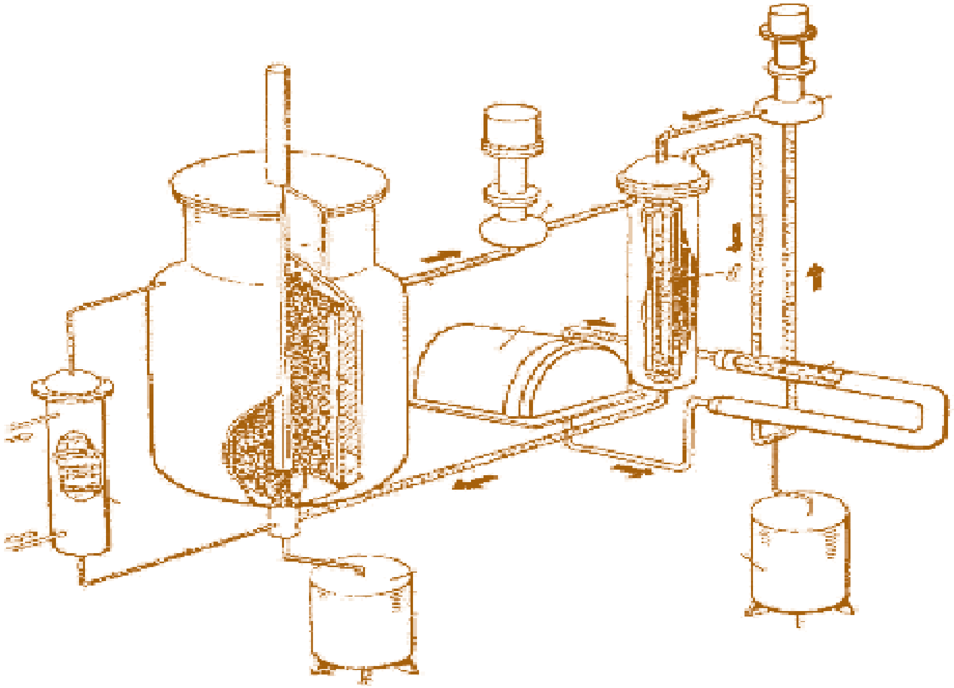
An extensive review of MSR development in Russia through 1989 is given in the book "Molten salt nuclear energy systems - perspectives and problems" V. Novikov, V. Ignatiev, V. Fedulov, V. Cherednikov, Moscow, 1990

Introduction of MSR Technology in Nuclear Power: From FSCR to MSR

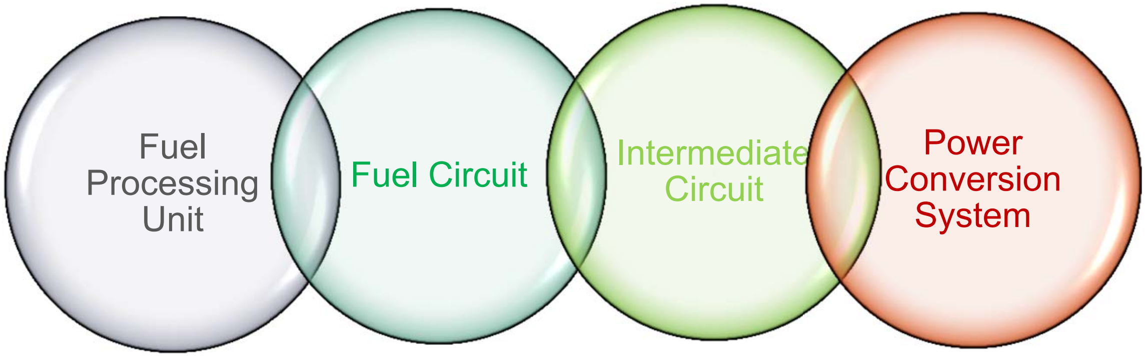
Stage 1 (FSCR):
Fluoride salt as coolant + solid HTR fuel

Stage 2 (MSR-converter):
- *Fluoride salt as coolant & liquid fuel*
- *Graphite as moderator*
- *No fuel salt processing*

Stage 3 (MSR-breeder):
- *Fluoride salt as coolant & liquid fuel*
- *Fuel salt processing*
- *Homogenious core (?)*



Boundaries and Interfaces



Fuel Processing Unit

Fuel Circuit

Intermediate Circuit

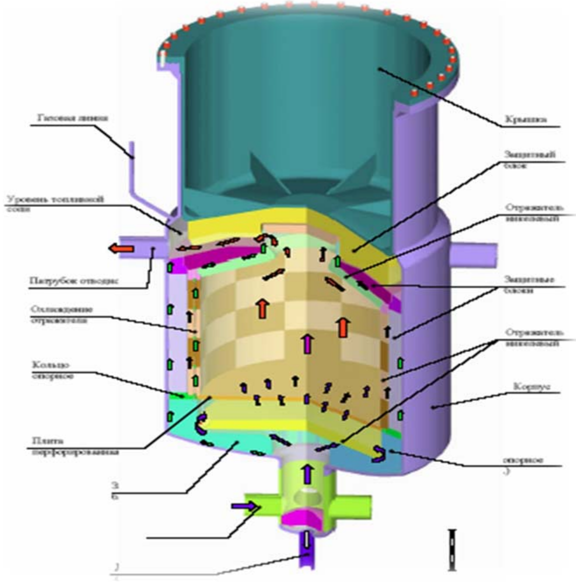
Power Conversion System

Today different fast reactor concepts using molten salt are under consideration in Russia

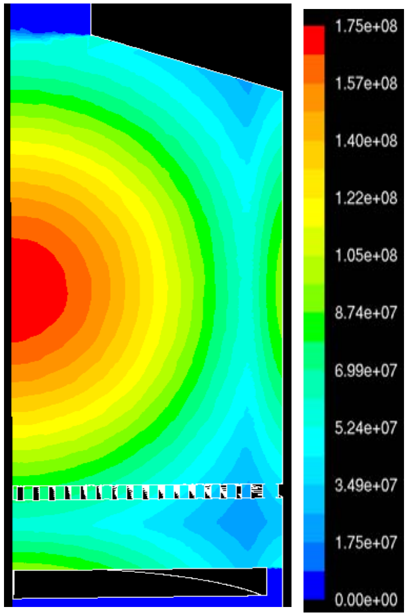
- *R&D studies are on-going in order to verify that fast spectrum MSR systems satisfy the goals of Gen-IV in terms of <u>technical feasibility, sustainability, safety and waste management</u>*
- *MOSART studies are done within ISTC#1606 and #3749 tasks as well as Rosatom MARS and Euratom FP6 EVOL projects*
- *Kurchatov Institute is a partner for EU Horizon 2020 SAMOFAR project focused on MSFR safety*
- *RFBR supports some experimental studies at Kurchatov Institute concerning fuel salt & material properties for MOSART*
- *Other groups are still interested to consider concepts with a very low technology readiness levels: the MSFR operating in U-Pu fuel cycle and MSR with direct lead cooling in primary circuit*

MOSART –Transforming Reactor System

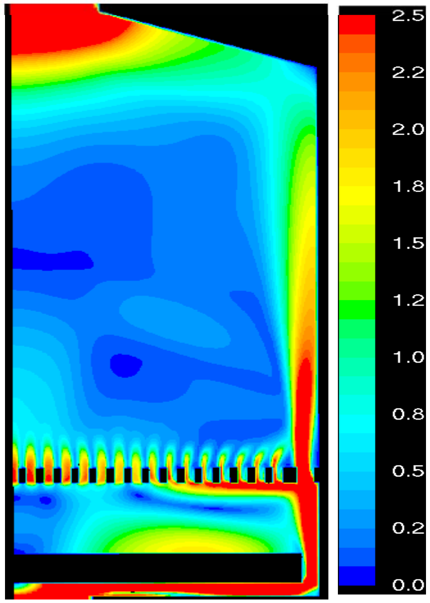
Started with TRU's Fluorides from LWR used nuclear fuel has flexible fuel cycle and can operate in different modes:
- *Transmuter*
- *Self-sustainable*
- *Breeder*



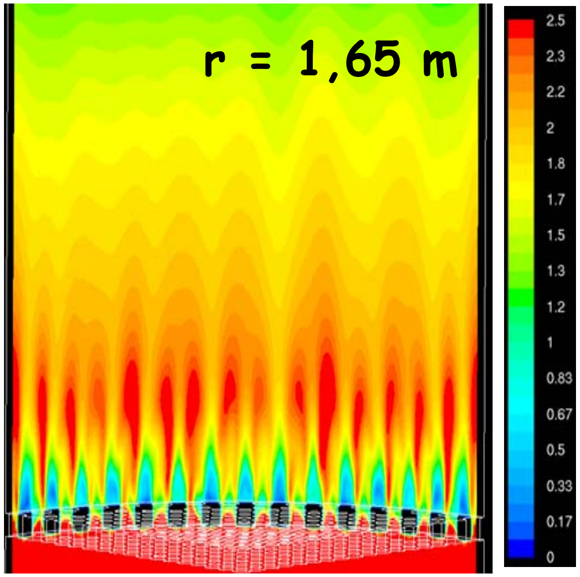
System	*burner / breeder*	
Fluid streams	*1*	*2*
Power capacity, MWt	*2400*	*2400*
Fuel salt inlet/outlet temperature, °C	*600 /720*	*600 /720*
Fuel salt composition, mole %	*72LiF* *$27BeF_2$* *$1TRUF_3$*	*75LiF* *$16.5BeF_2$* *$6ThF_4$* *$2.5TRUF_3$*
Blanket salt composition, mole %	*no*	*75LiF* *$5BeF_2$* *$20ThF_4$*



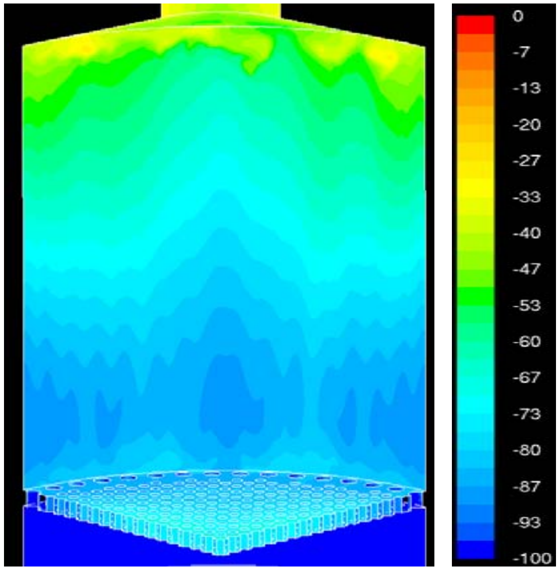
Heat source, W/m³



Velocity, m/s



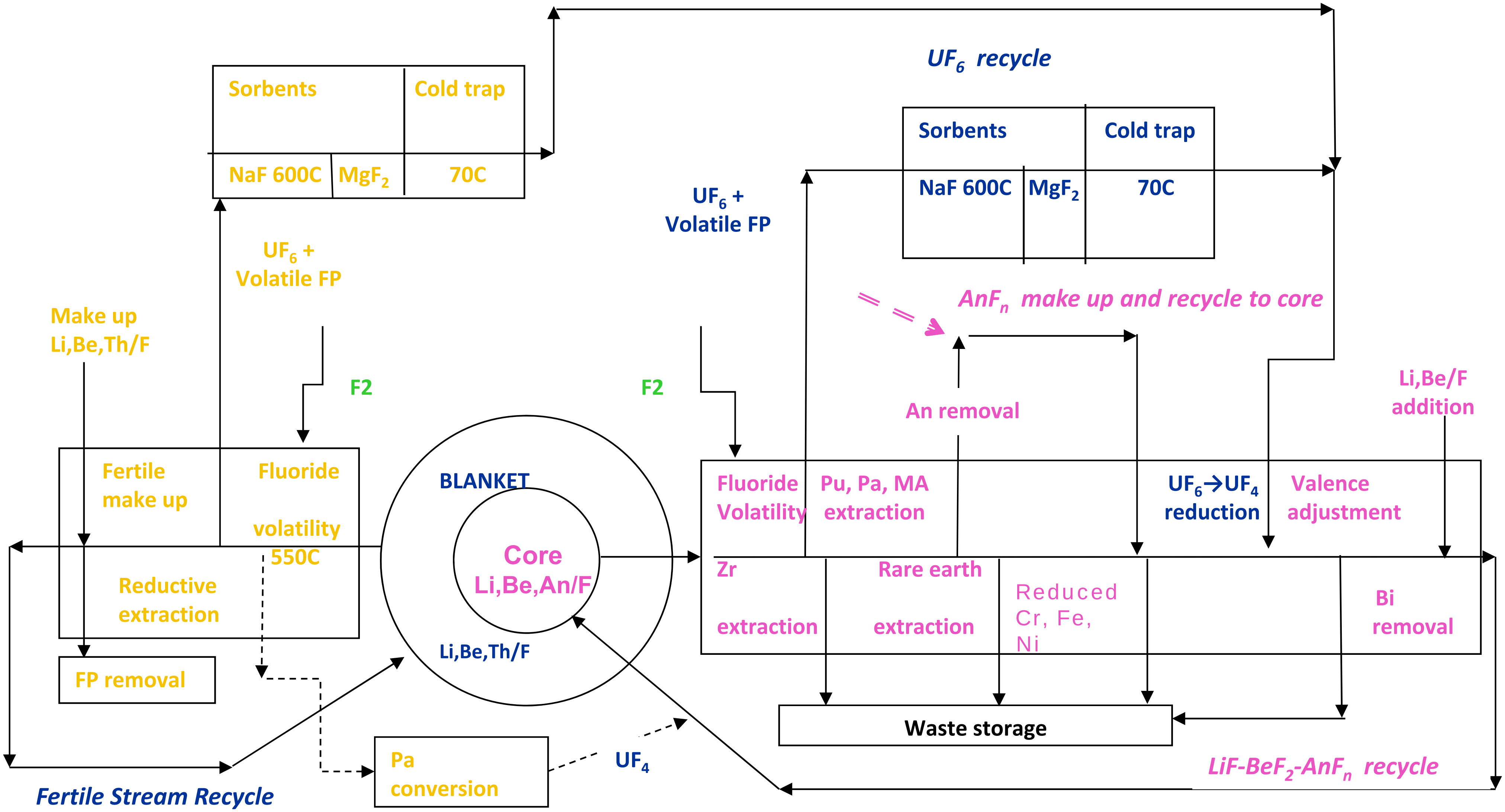
T max wall: 947 K;



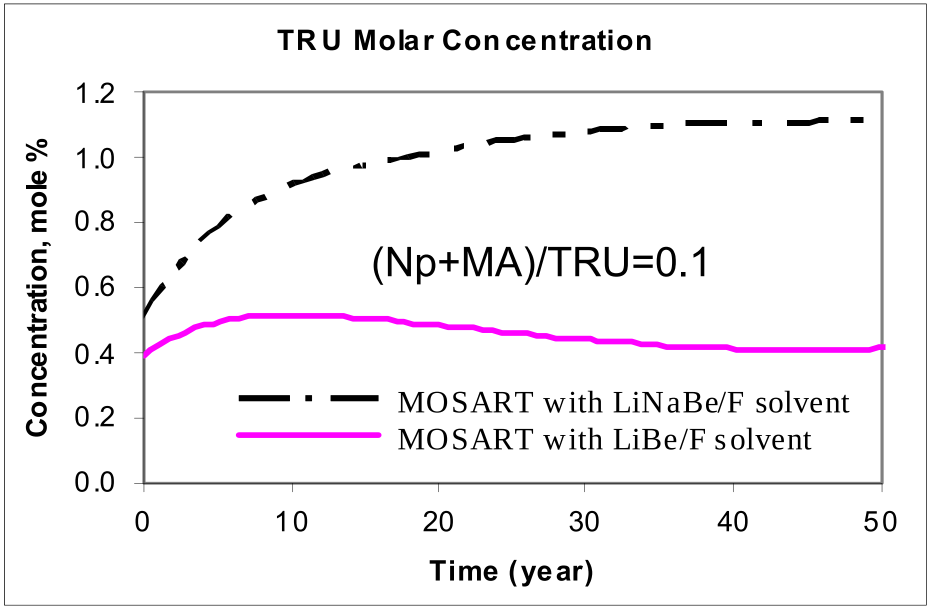
T max fluid : 1192 K

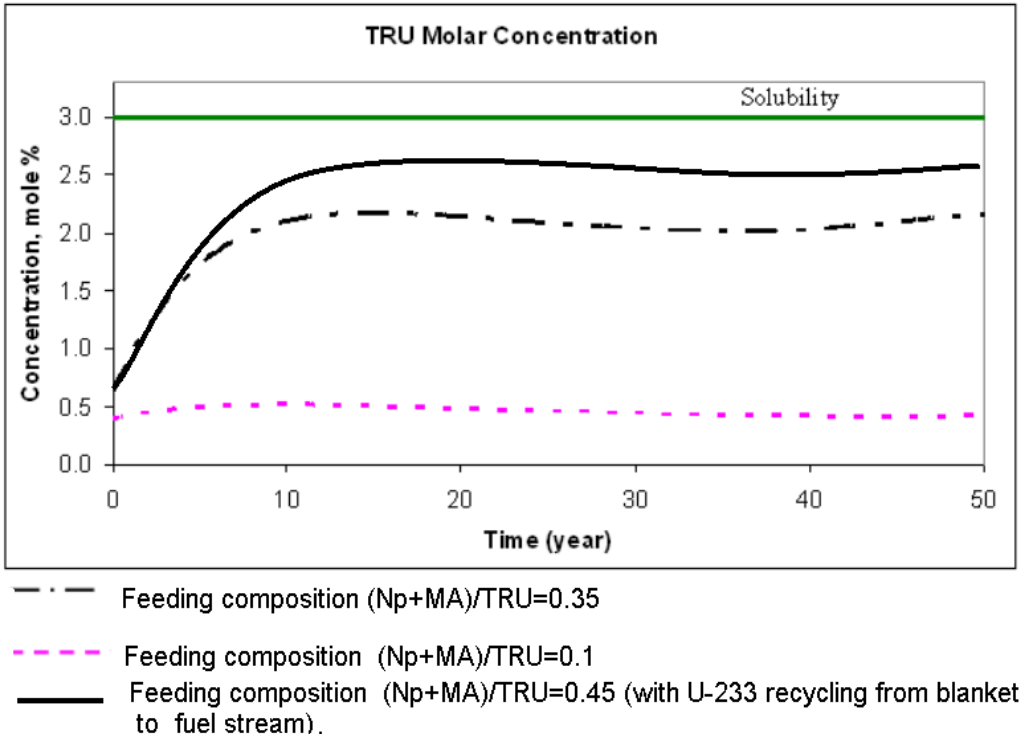
Two Fluid Th MOSART Flowsheet

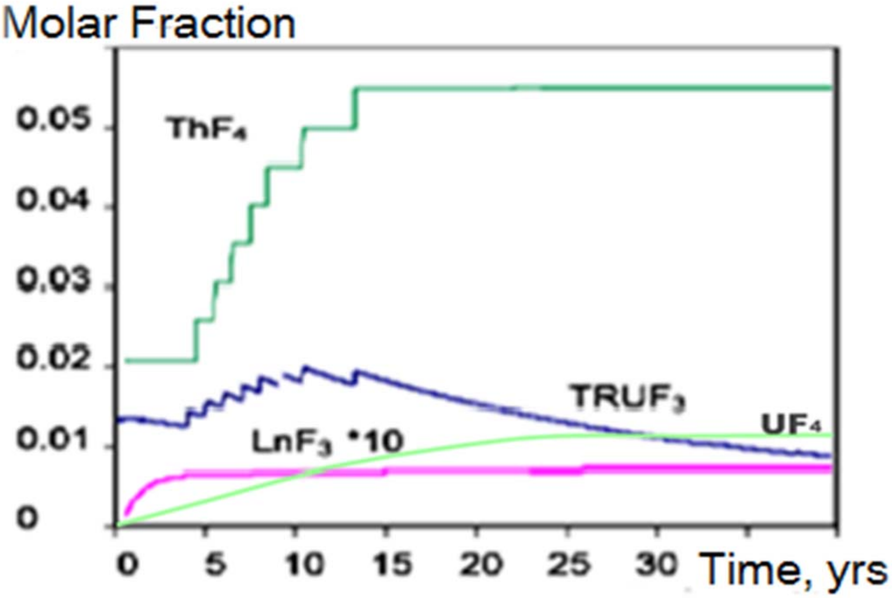
Thermal power: 2.4 GWt; Core height / diameter: 2-3.6m/2-3.4m; Radial blanket thickness: 0.2-0.6m;
FP removal time: 300 efpd; TRU loading at BOL / EOL – 1.5t/3.5t; ^{233}U production from 150 up to 300 kg/yr;
Fuel salt:73LiF-27BeF$_2$; T$_{inlet}$=600C; Blanket salt: 75LiF-5BeF$_2$-20ThF$_4$; T$_{inlet}$=600C



MOSART Fuel Cycles





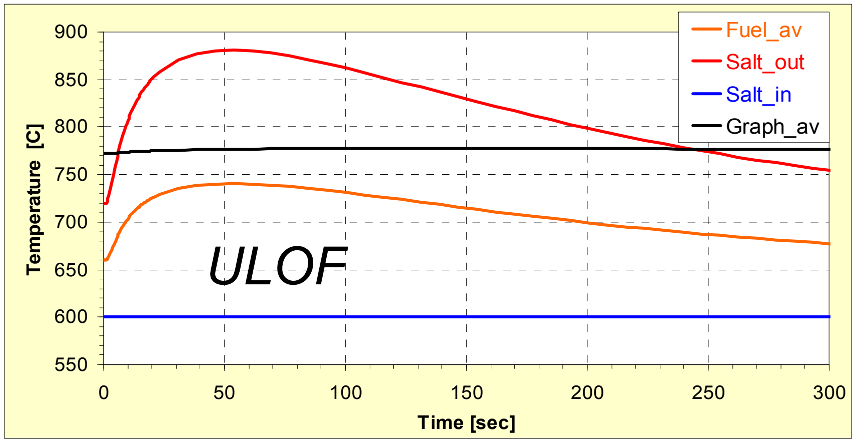


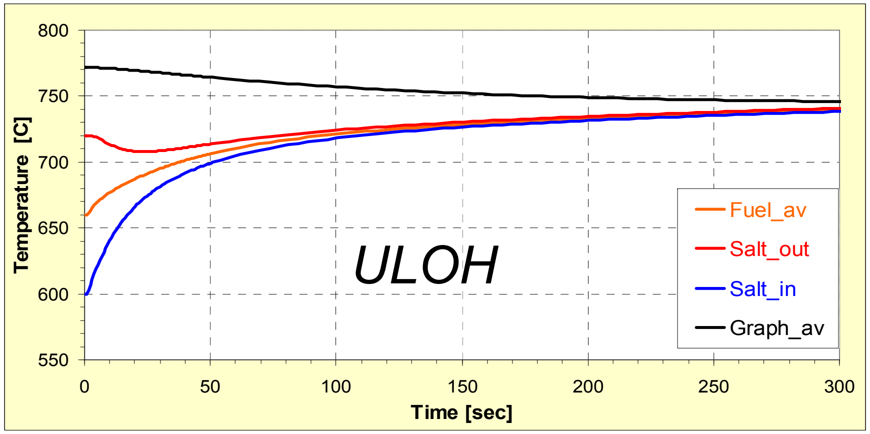
- *Single fluid 2400MWt MOSART core containing as initial loading 2 mole% of ThF_4 and 1.2 mole % of $TRUF_3$, with the rare earth removal cycle 300 epdf after 12 years can operate without any $TRUF_3$ make up basing only on Th support as a self-sustainable system.*
- *At equilibrium molar fraction of fertile material in the fuel salt is near 6 mole % and it is enough to support the system with CR=1.*

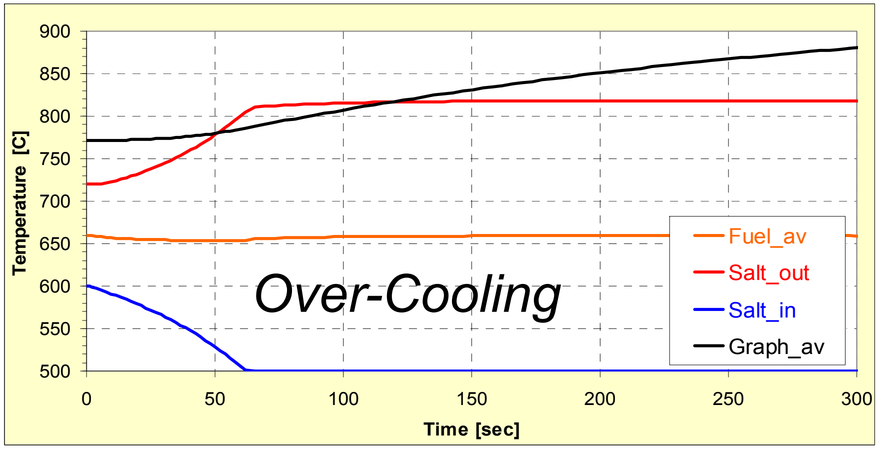
Preliminary Process Hazards Analysis

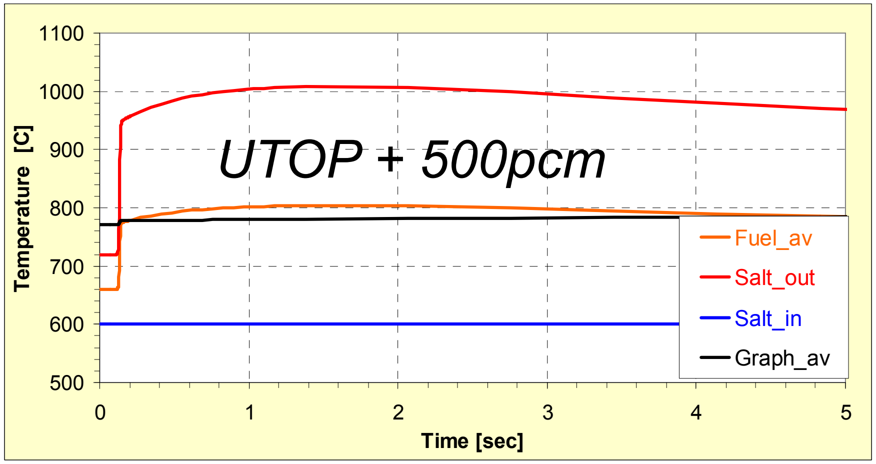
System / Component	***Initiating Event***
Primary circuit / containment	• *Failure in the main circulation pipe* • *Failure in the primary heat exchanger* • *Failure in the main circulation pump* • *Unintentional control rod withdrawal* • *Loss of blanket salt* • *Premature criticality during filling* • *Inflow of contaminants or unexpected isotopic ratio in the fuel salt* • *Inadvertent release of fission gas from reactor cell and / or containment*
Intermediate circuit	• *Failure in the main circulation pipe* • *Failure in the main circulation pump*
Salt chemical processing	• *Hydrogen reacts with fluorine in chemical processing system* • *Excess pressure in the helium bubbler* • *Inadequate removal of Pa or U*
Drain Tank	• *Improper or inadequate cooling of the drained fuel salt* • *A partially thawed piece of the salt plug or other solid mass obstructs piping to the drain tank*

MOSART Transients Analysis



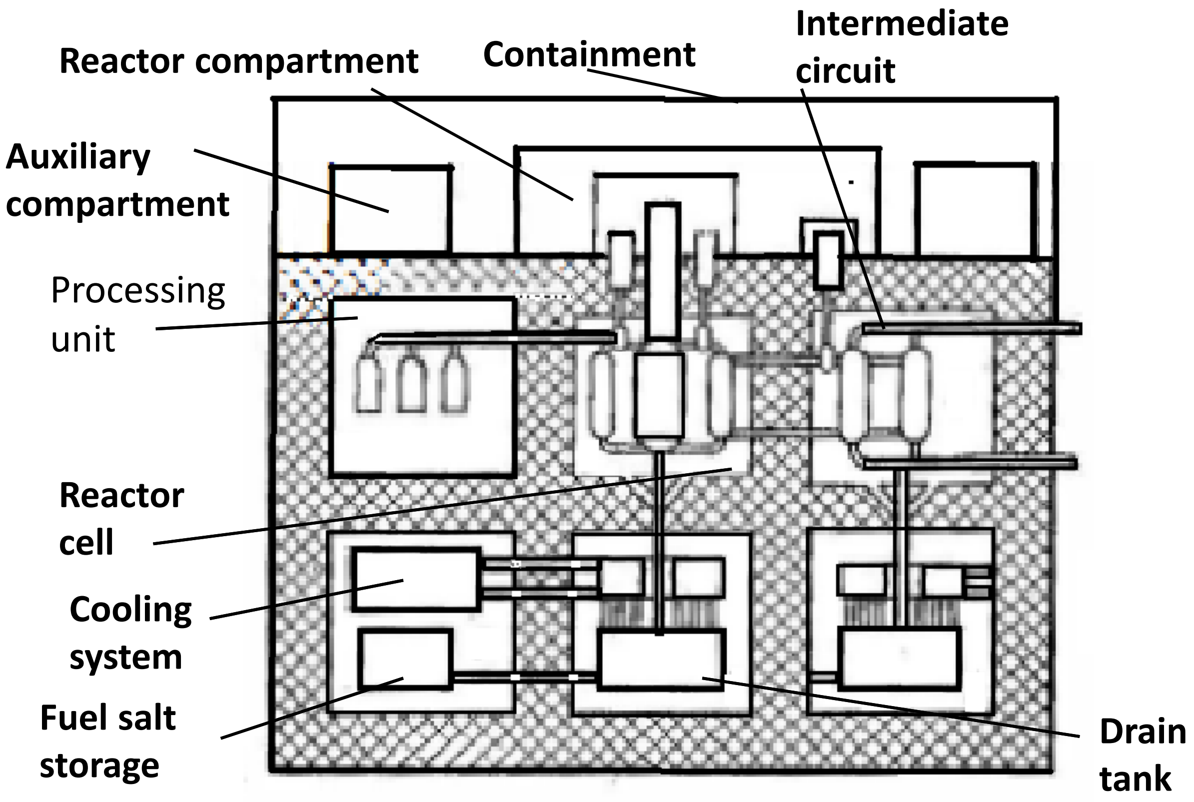




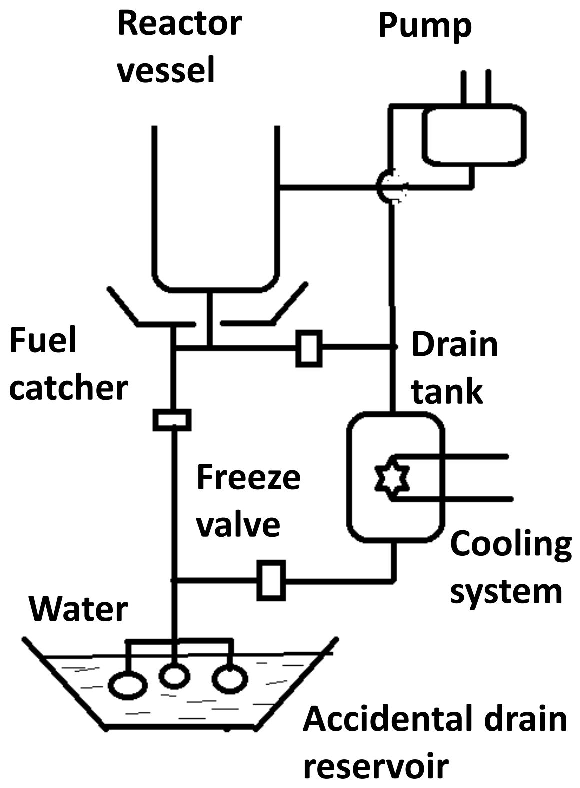


The MOSART is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP. The system was shown to buffer reactivity insertion of up to + 1.5$. System temperatures are expected to rise only ~300C above nominal under this severe transient conditions. The mechanical and structural integrity of the system is not expected to be impaired.

Design Aspects of MSR Safety



Safety barriers: wall of fuel circuit, double wall of the cell, containment



$t = 20\,min$

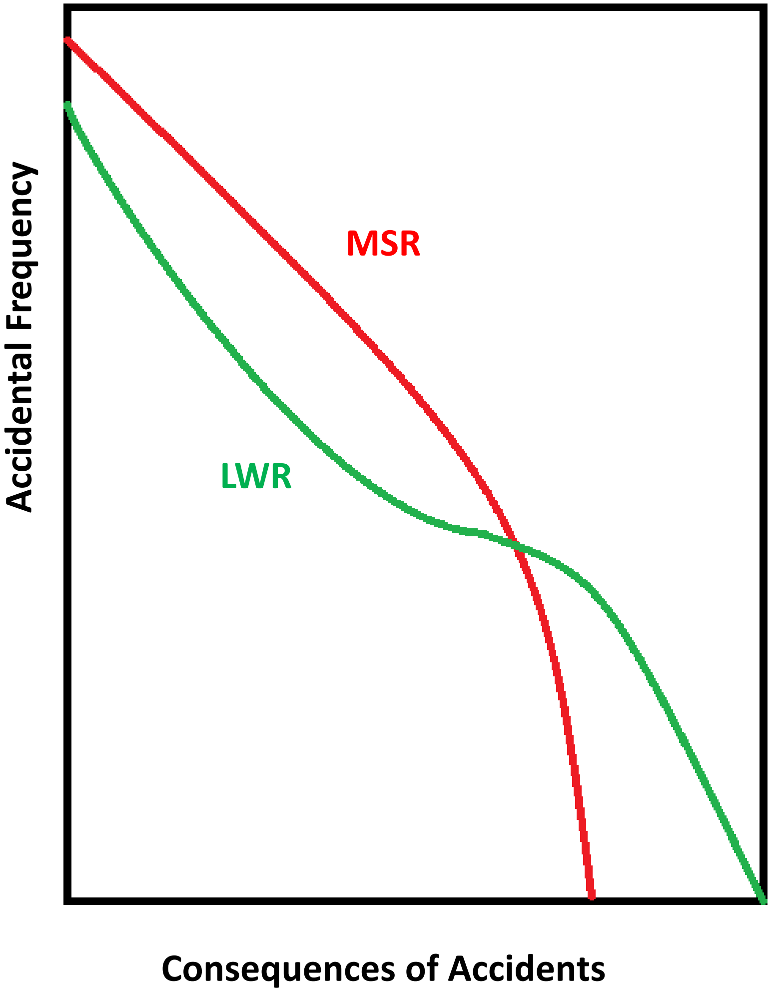
Severe accident with the rupture of the main fuel salt pipe and fuel discharged on the reactor box bottom

- *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 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 during the normal operation these nuclides are almost completely leave the fuel salt.*

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	0,0005	0,0005
Sr90	0,98	0,0005	0,0005
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

❖ *For MSR the total release of radioactivity would be significantly lower (by I - 2 orders of magnitude compared to PWR), though for several particular nuclides such I^{131} and I^{133} the differences are smaller*

Frequency distribution for the probability of accidents in the MSR and the LWR on the degree of contamination risk

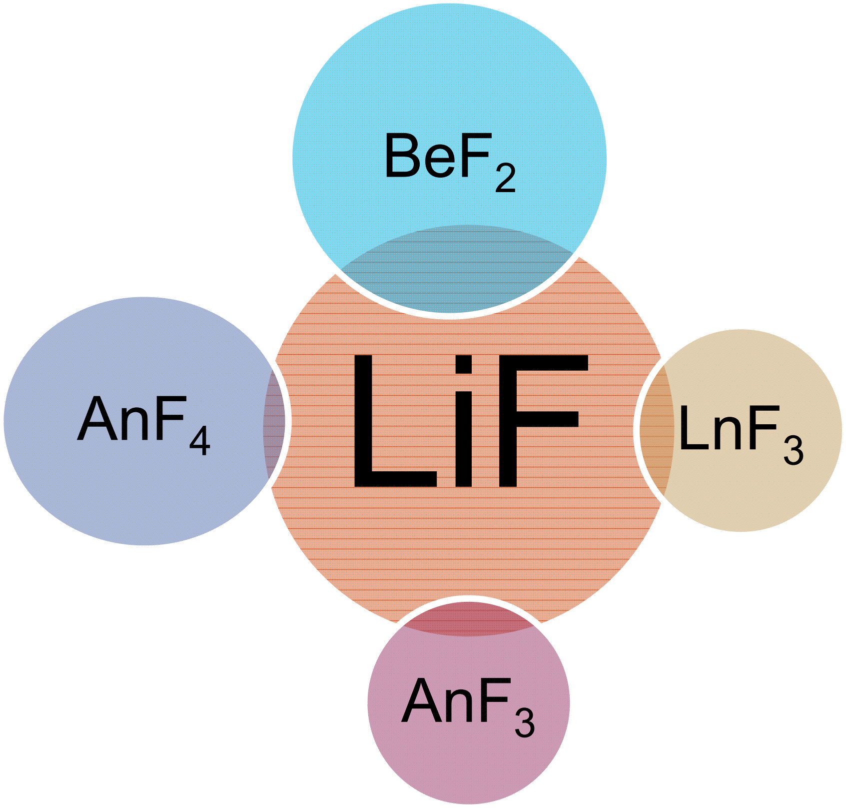


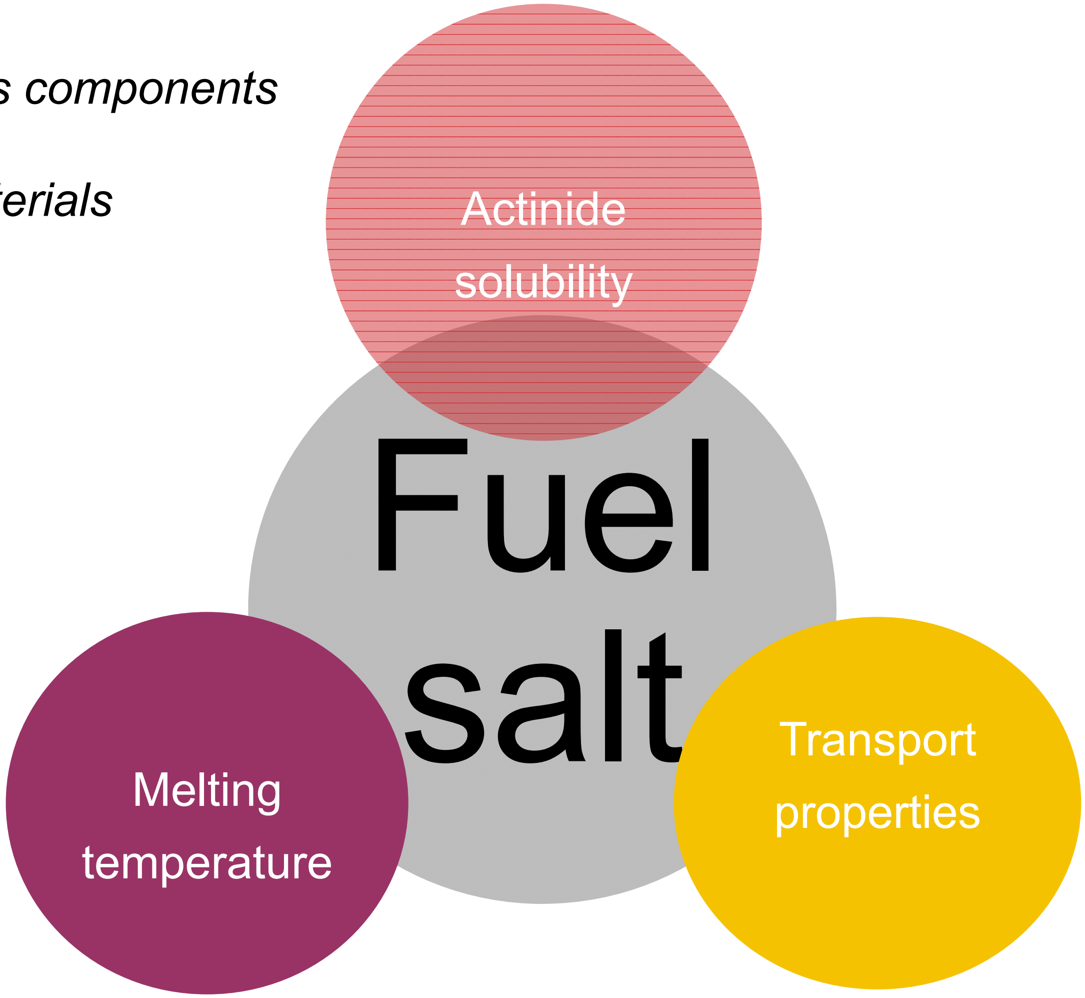
- ✓ *Probability of an accident with a relatively low impact for MSR is higher than for LWR. This is due to the possibility of leakage of radioactive liquid fuel in case of accidents in the pump, piping, valves.*
- ✓ *The consequences of severe accidents in particular, leading to the release of radioactive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

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

- *Low neutron cross section for the solvent*
- *Thermal stability of the salt components*
- *Low vapor pressure*
- *Radiation stability*
- *Adequate solubility of fuel and FP's components*
- *Adequate transport properties*
- *Compatibility with construction materials*
- *Low fuel and processing costs*





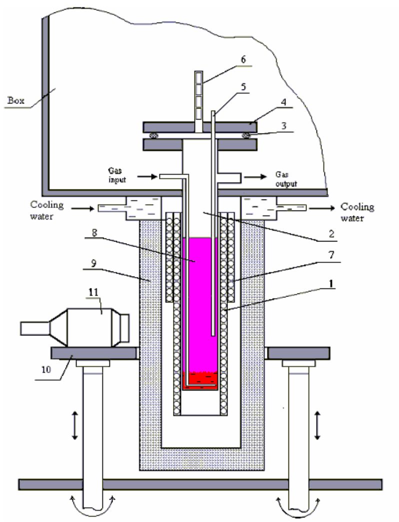
Materials Involved in Experimental Studies

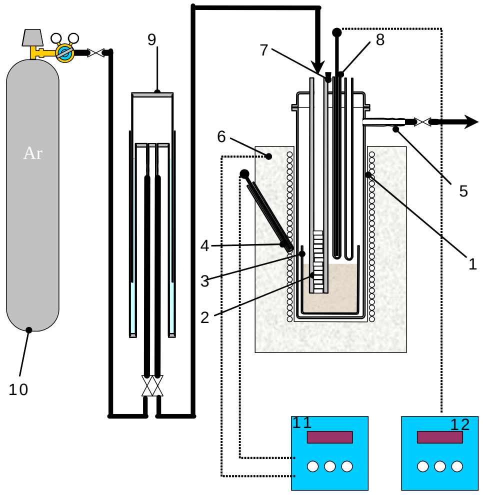
Components	*Breeder*	*Burner*
Solvent major constituents	LiF, BeF_2	LiF, NaF, KF, BeF_2
Fertile additives	ThF_4	*Fertile-free,* UF_4
Fissile additives	UF_4, PuF_3	PuF_3, AmF_3
Soluble fission products	CeF_3, NdF_3, LaF_3	CeF_3, NdF_3, LaF_3
Gaseous reaction products	T (D)	T(D)
Alloys major constituents	*Ni-Mo; Ni-W*	*Ni-Mo; Ni-W*

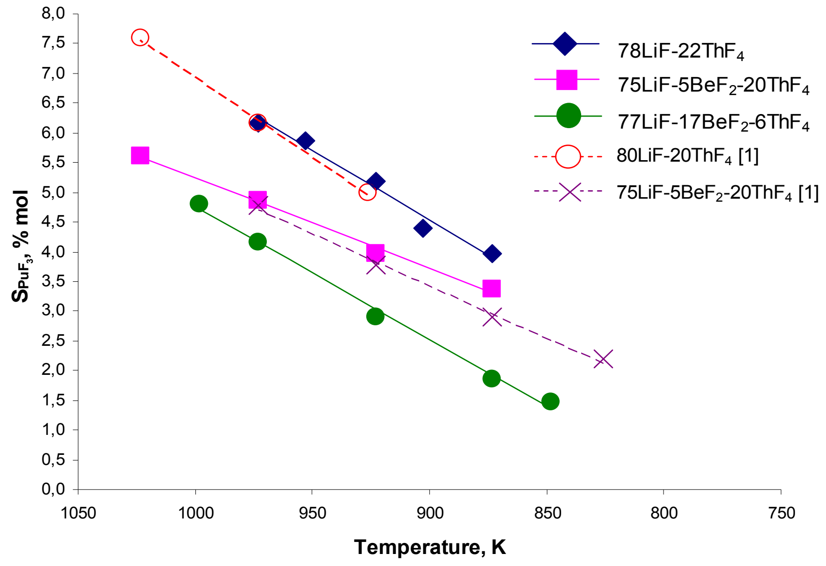
The data on PuF_3 solubility in molten salt fluorides appear to follow a linear relationship within the experimental accuracy of the measurements when plotted as log of molar concentration of AnF_3 vs. 1/T(K)

LiF	NaF	KF	BeF_2	ThF_4	T, K	A	$-B \cdot 10^{-3}$
46.5	11.5	42	0	0	823-973	5.59	3.949
73	0	0	27	0	825-1000	3.927	3.099
66	0	0	34	0	800-900	3.231	3.096
15	58	0	27	0	825-925	3.639	2.750
17	58	0	25	0	800-900	3.253	2.578
78	0	0	0	22	873-973	2.58	1.73
75	0	0	5	20	873-1023	2.06	1.34
77	0	0	17	6	848-998	3.61	2.91

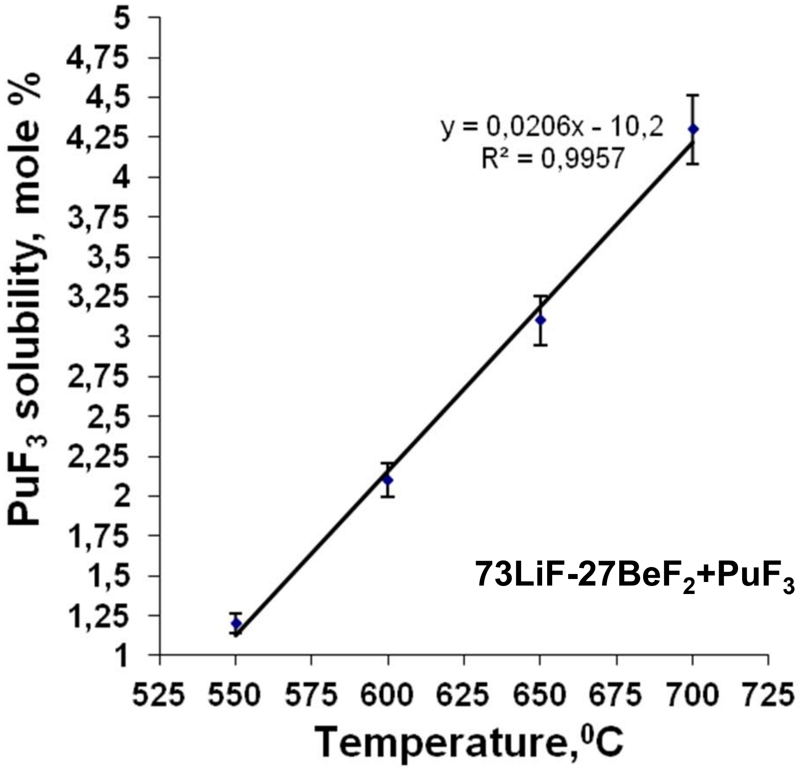
$$\log S_{PuF_3}, \text{мол\%} = A + B/T, K$$

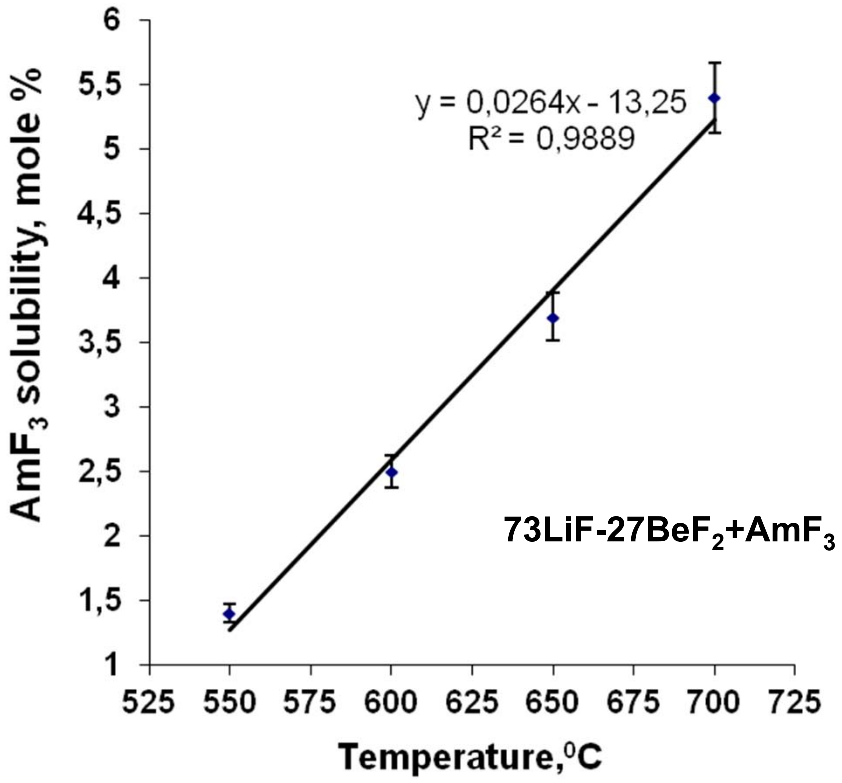


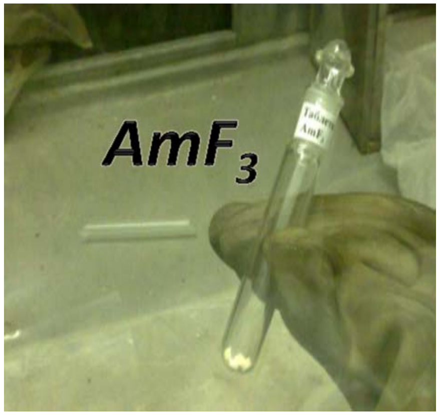




An and Ln Trifluoride Solubility



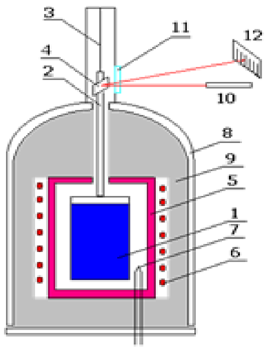


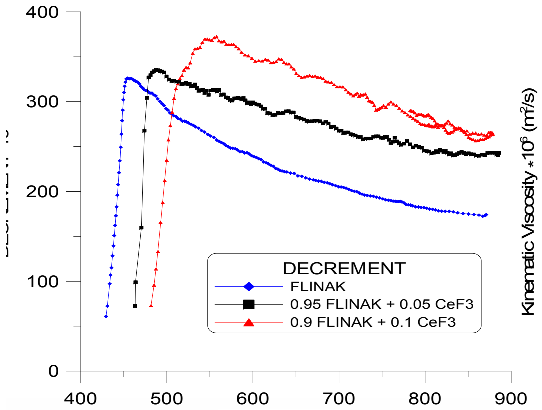


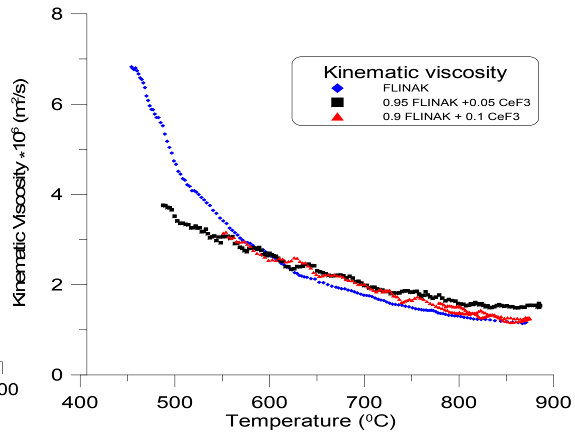
Temperature, K	72,5LiF-7ThF$_4$-20,5UF$_4$		78LiF-7ThF$_4$-15UF$_4$	
	PuF$_3$	CeF$_3$	PuF$_3$	CeF$_3$
873	0,35±0,02	1,5±0,1	1,45±0,7	2,6±0,1
923	4,5±0,2	2,5±0,1	5,6±0,3	3,6±0,2
973	8,4±0,4	3,7±0,2	9,5±0,5	4,8±0,3
1023	9,4±0,5	3,9±0,2	10,5±0,6	5,0±0,3

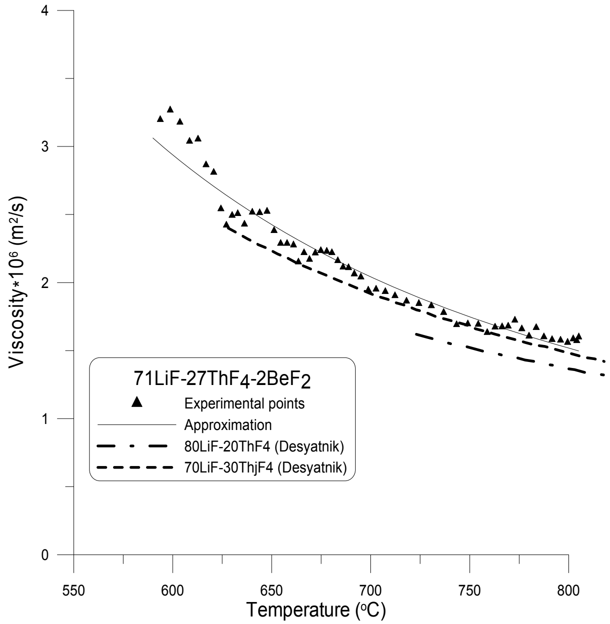
Near the liquidus temperature for 78LiF-7ThF$_4$-15UF$_4$ and 72,5LiF-7ThF$_4$-20,5UF$_4$ salts, the CeF$_3$ significantly displace PuF$_3$

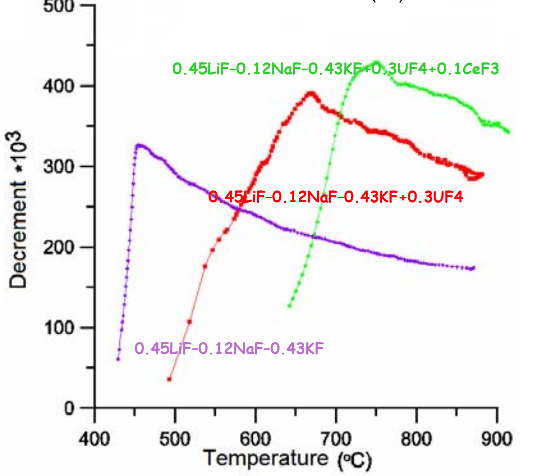
Viscosity Measurements

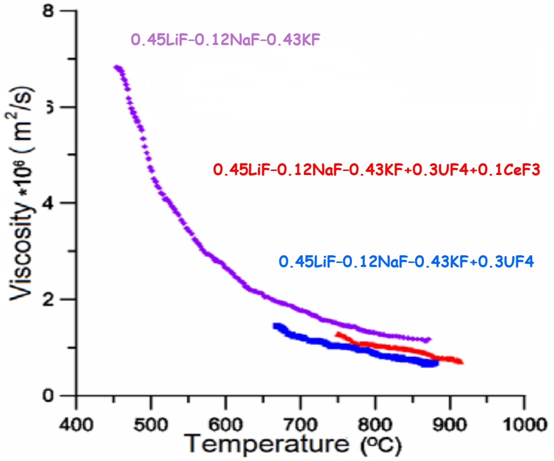


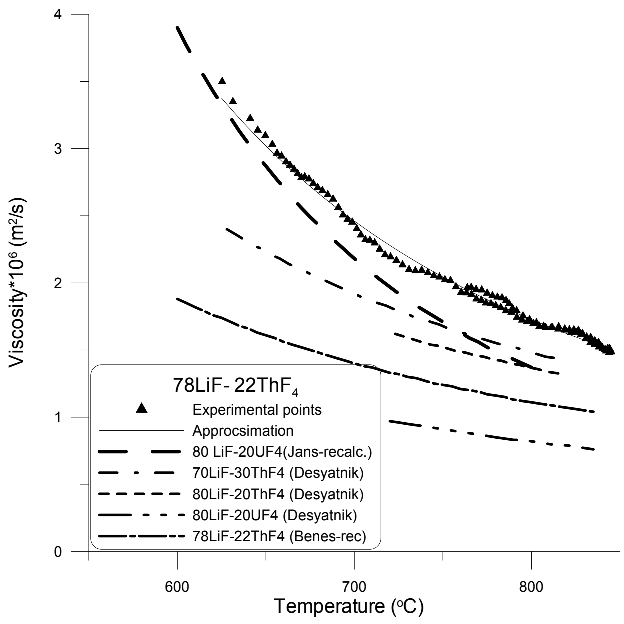


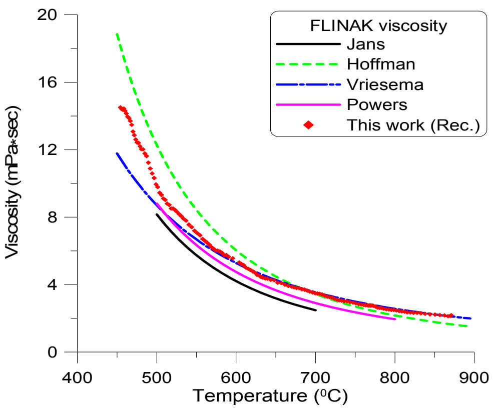


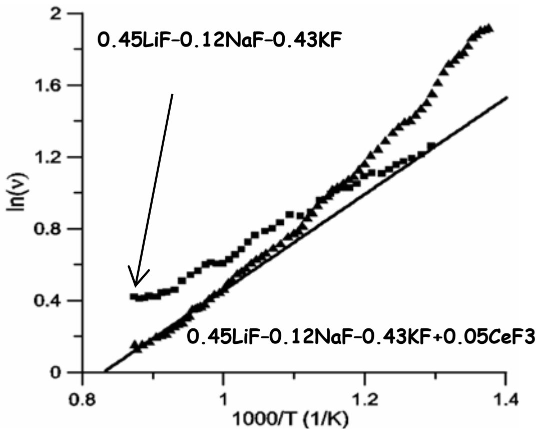




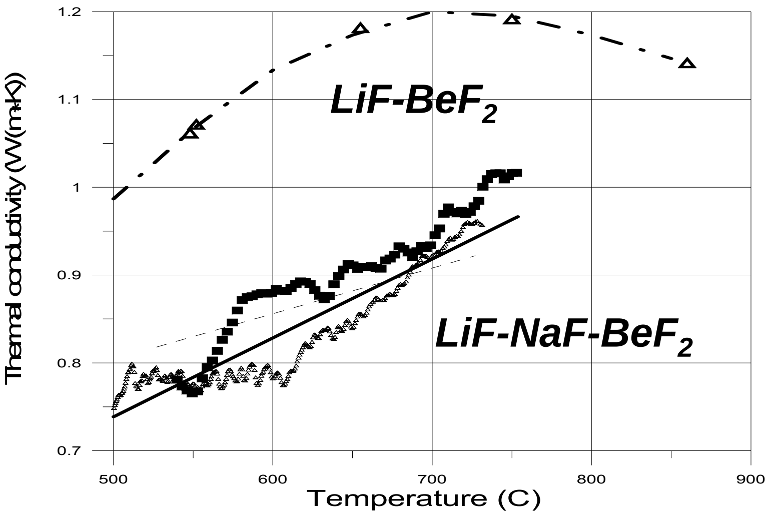


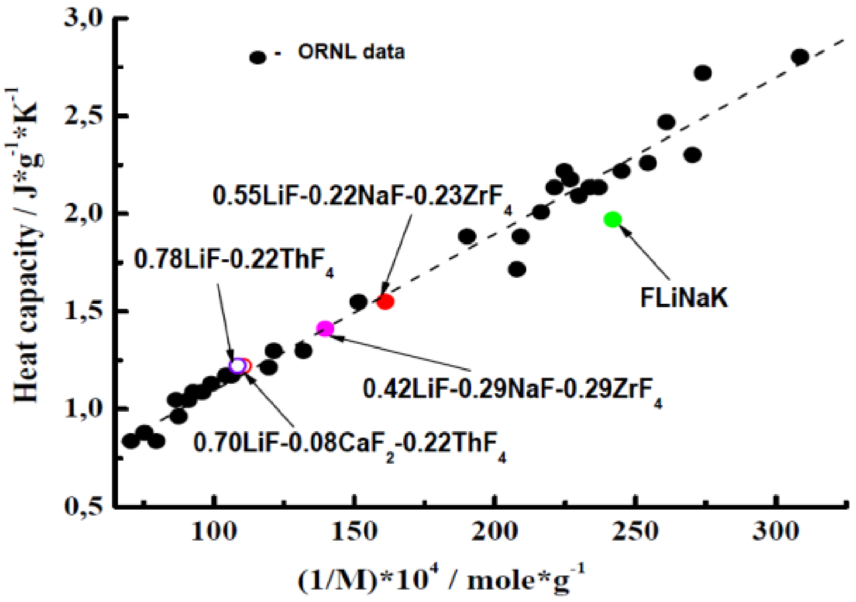






Fuel Salt Transport Properties

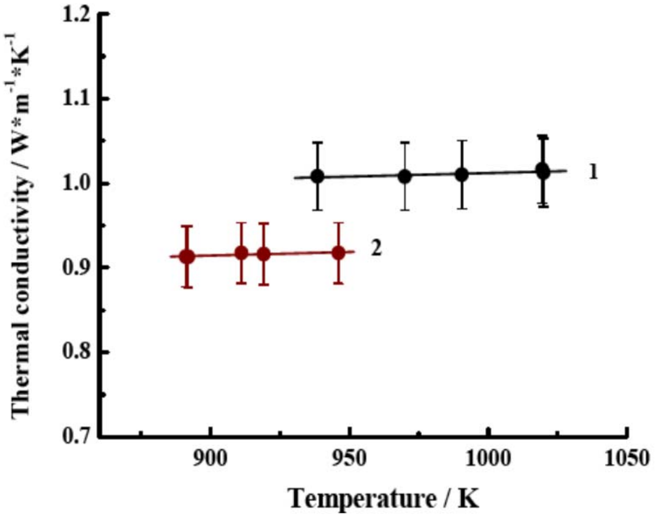


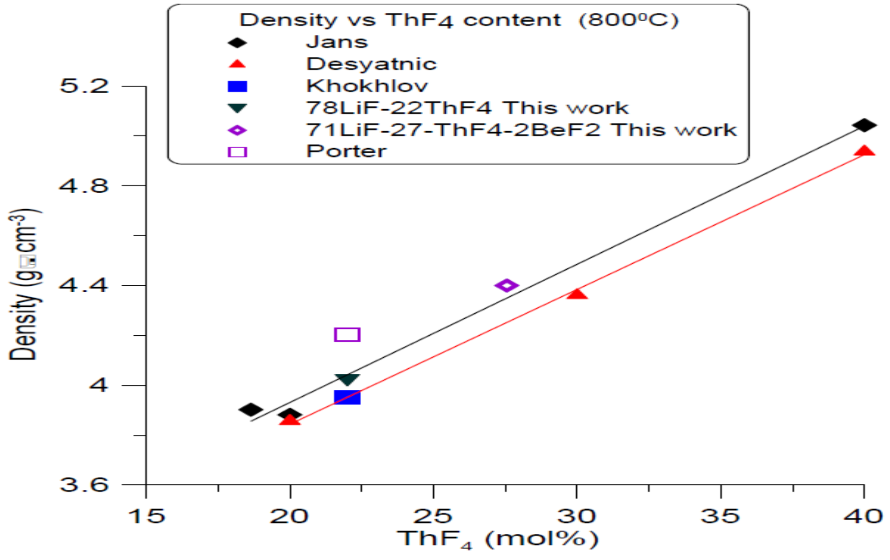


$$\lambda = -0.34 + 0.5 \cdot 10^{-3} T(K) + 32/M$$

$$\lambda = 0.928 + 8.397 \cdot 10^{-5} T \pm 0.054 \qquad (0.78LiF - 0.22ThF_4) \text{ и}$$

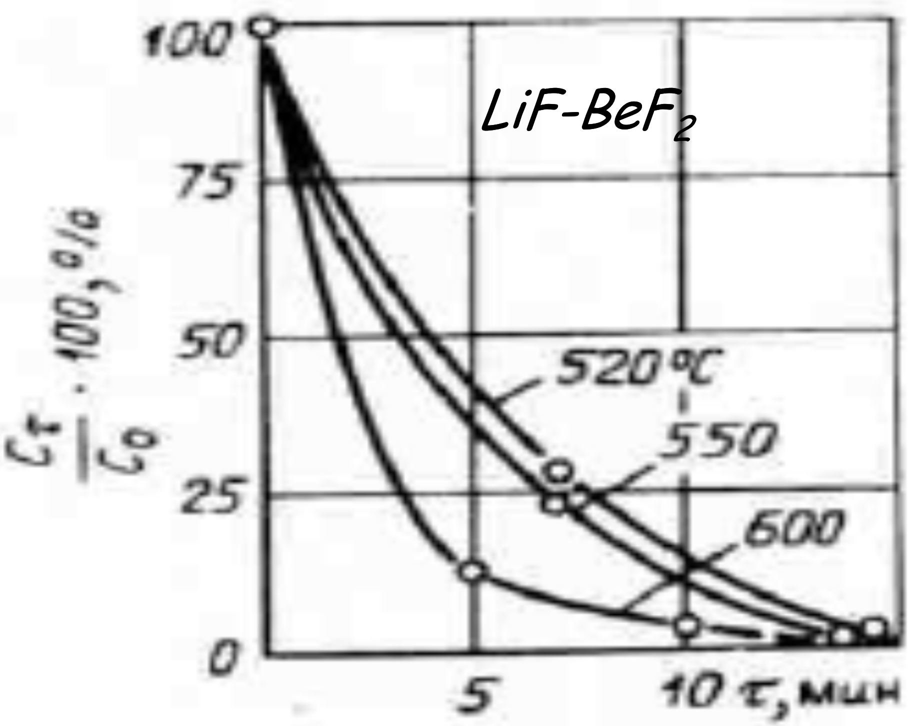
$$\lambda = 0.842 + 8.062 \cdot 10^{-5} T \pm 0.038 \qquad (0.70LiF\text{-}0.08CaF_2\text{-}0.22ThF_4).$$

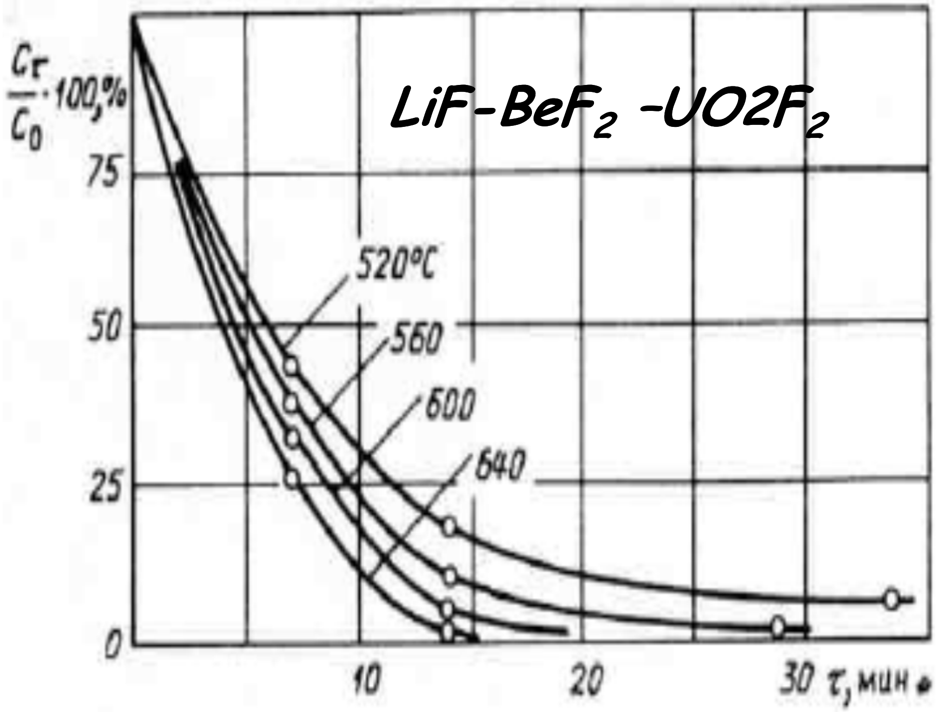




U and Pa Removal from Fuel Salt by Fluoride Bubbling

The rate of $UF_4 + F_2 \leftrightarrow UF_6$ reaction at 500—600°C is very high

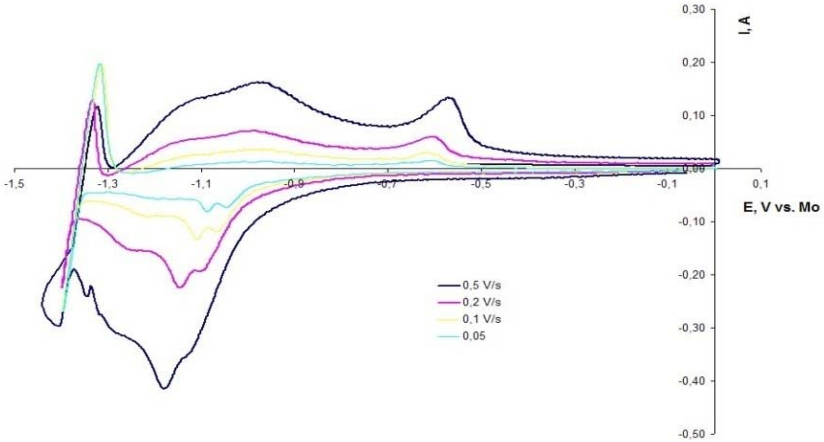


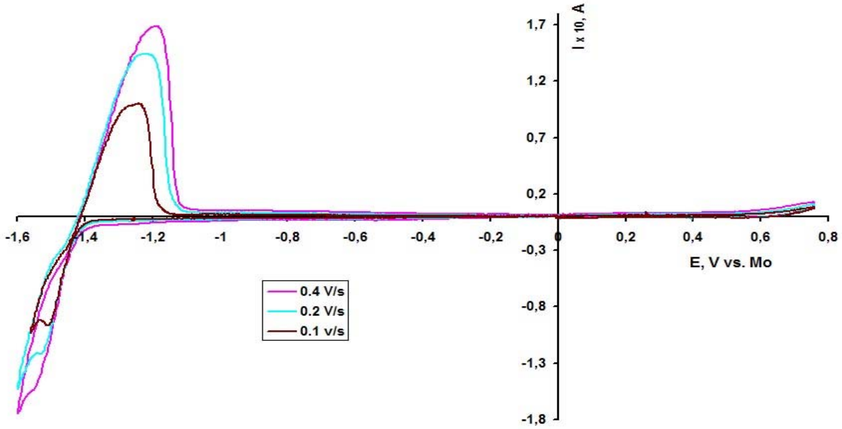


- *<u>In static conditions</u> at temperature of 750°C and fluorine pressure of 50 kPa the normalized rate of PaF_5 removal from the Li,Be/F salt surface into the fluorinator's "cold" zone with the wall temperature of 400°C was $\Delta m/m/\tau \approx 5 \cdot 10^{-2}$ hr^{-1}. In this case the ^{233}Pa recovery from the salt reached 98%.*
- *Using fluorine bubbling, the process of U and Pa removal can be intensified by an order of value as compared to static fluorination method.*

Electrochemical Properties

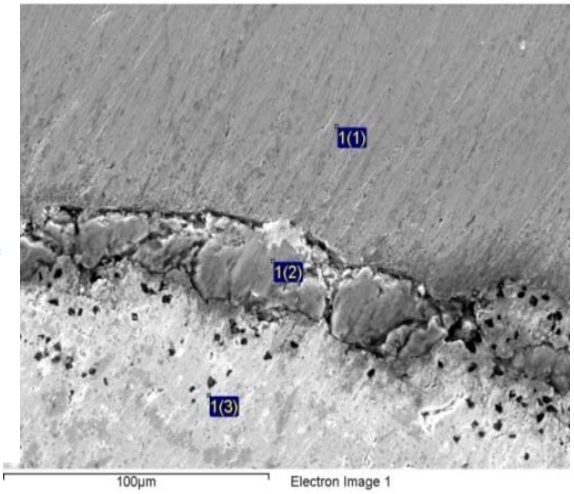
CV's of PuF_3 (0.05 mole %) in LiF-NaF for various potential scan rates. WE - Mo, RE – Mo. T=1023K





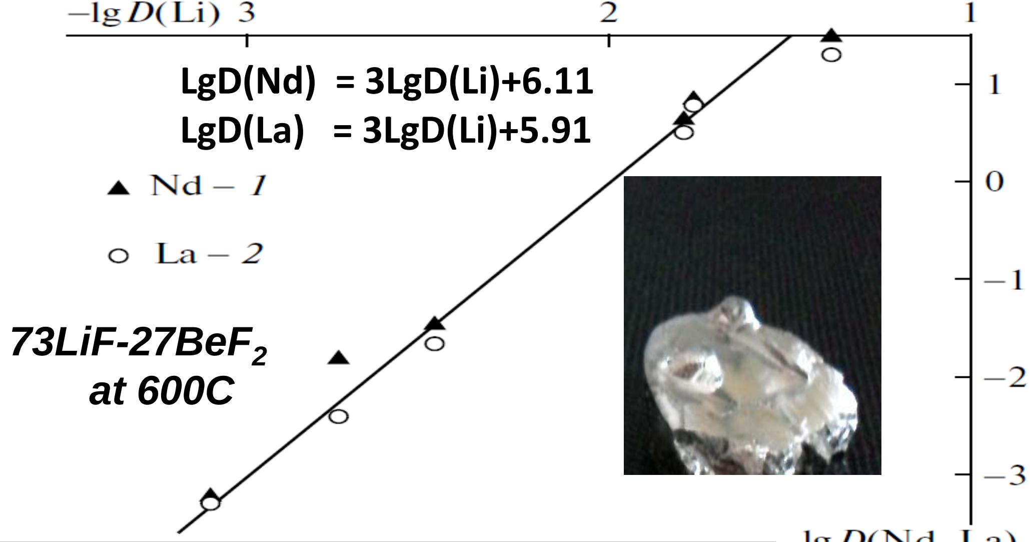
CV's of PuF_3 (0.1 mole %) in LiF-NaF-BeF_2 for various potential scan rates. WE - Mo, RE – Mo. T=853K

- E*quilibrium potentials of the Pu^{3+}/Pu, Zr^{2+}/Zr, Be^{2+}/Be, Na^+/Na couples in 15LiF-58NaF-27BeF_2 and 60LiF-40NaF (in mole %) melts were determined*
- *It was shown, that plutonium is deposited on molybdenum cathode earlier than beryllium and sodium. Difference of plutonium and beryllium deposition potentials is equal ~ 0.15 V, plutonium and sodium deposition potentials – ~ 0.30 V.*
- *Diffusion coefficients of zirconium and plutonium ions in 15LiF-58NaF-27BeF_2 and 60LiF-40NaF melts were estimated*



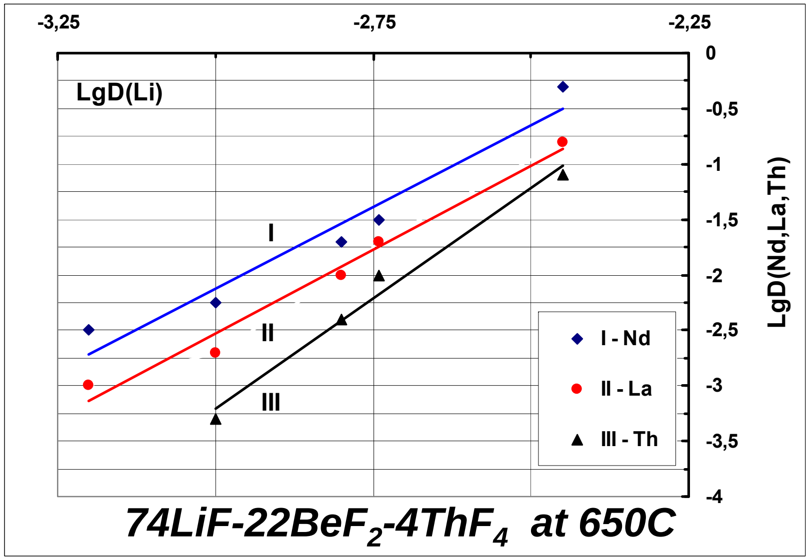
Reductive Extraction with Liquid Bi-Li

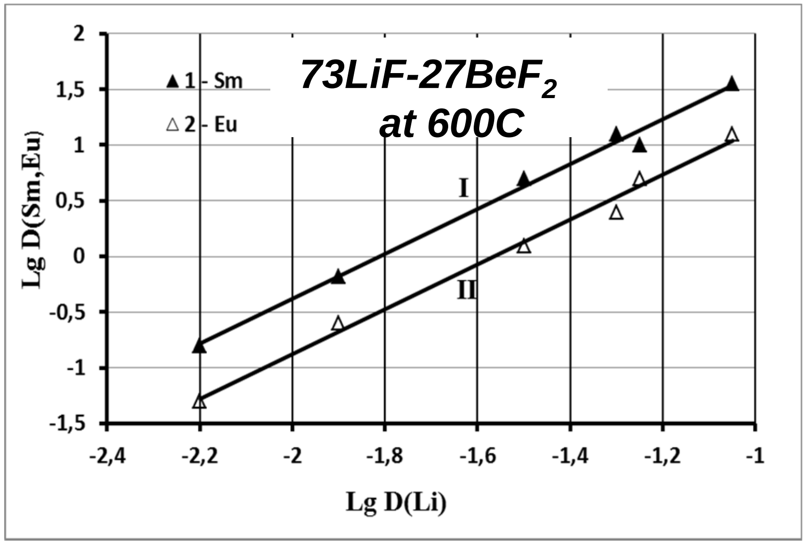
Reductive extraction of actinides from molten salt into liquid bismuth with their subsequent re-extraction into purified salt flow is the most acceptable technological method of actinide recycling. In order to determine particular parameters of this stage, it will be necessary to conduct additional experimental studies with selected salt composition on enlarged facilities


LgD(Nd) = 3LgD(Li)+6.11

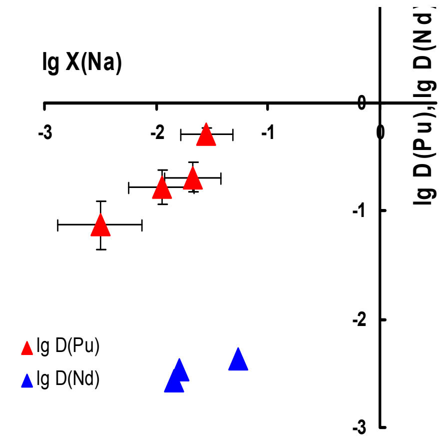
LgD(La) = 3LgD(Li)+5.91

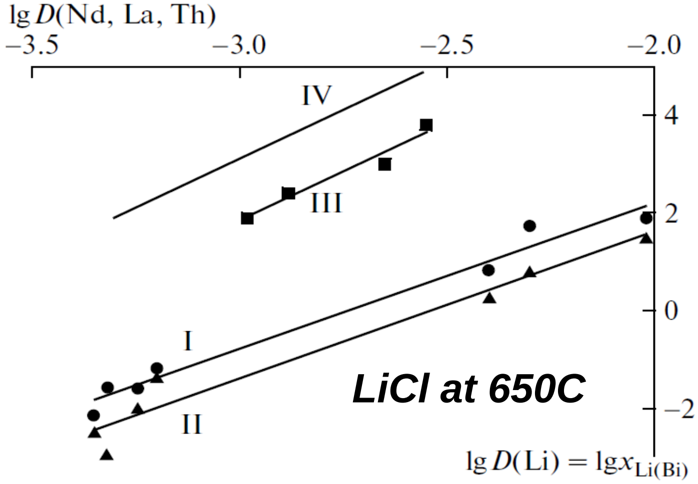
$73LiF\text{-}27BeF_2$ at 600C


$74LiF\text{-}22BeF_2\text{-}4ThF_4$ at 650C


$73LiF\text{-}27BeF_2$ at 600C

$15LiF\text{-}58NaF\text{-}27BeF_2$



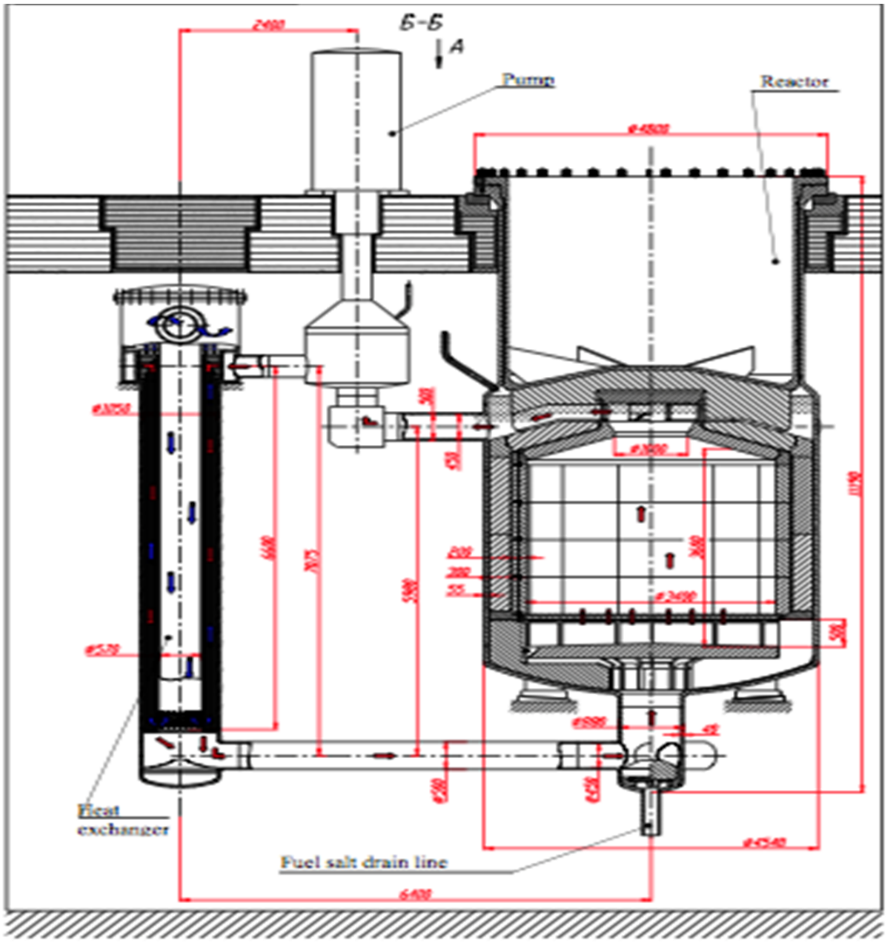

LiCl at 650C

Pu distribution coefficient in $15LiF\text{-}58NaF\text{-}27BeF_2/Bi$ is much less than in $LiF\text{-}BeF_2/Bi$ system

Selection of Materials for Components

Ni-based alloy
Circuits, Heat exchangers
600 / 720 ° C Creep, Creep-fatigue, Thermal fatigue, Aging, Welds...

Ni-based alloy
Intermediate circuit - *455 / 620 ° C* Aging, Welds, Compatibility NaF-NaBF$_4$, Oxidation, Wastage...



Ni-based alloy
/ SS
Vessel - *600 ° C*
Negligible creep

C, Ni
Reflector*600-750 ° C*
Negligible creep

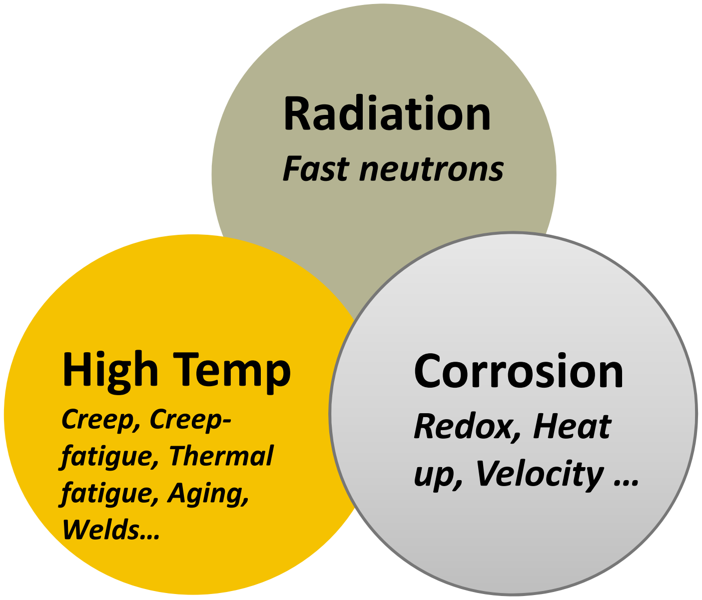
SiC
Distribution plate-
600 ° C,
High irradiation

- *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*

Gen IV MSR Container Materials

The success of MSR is strongly dependent on the compatibility of the container materials with the molten salts used in primary and secondary circuits

Combined environments

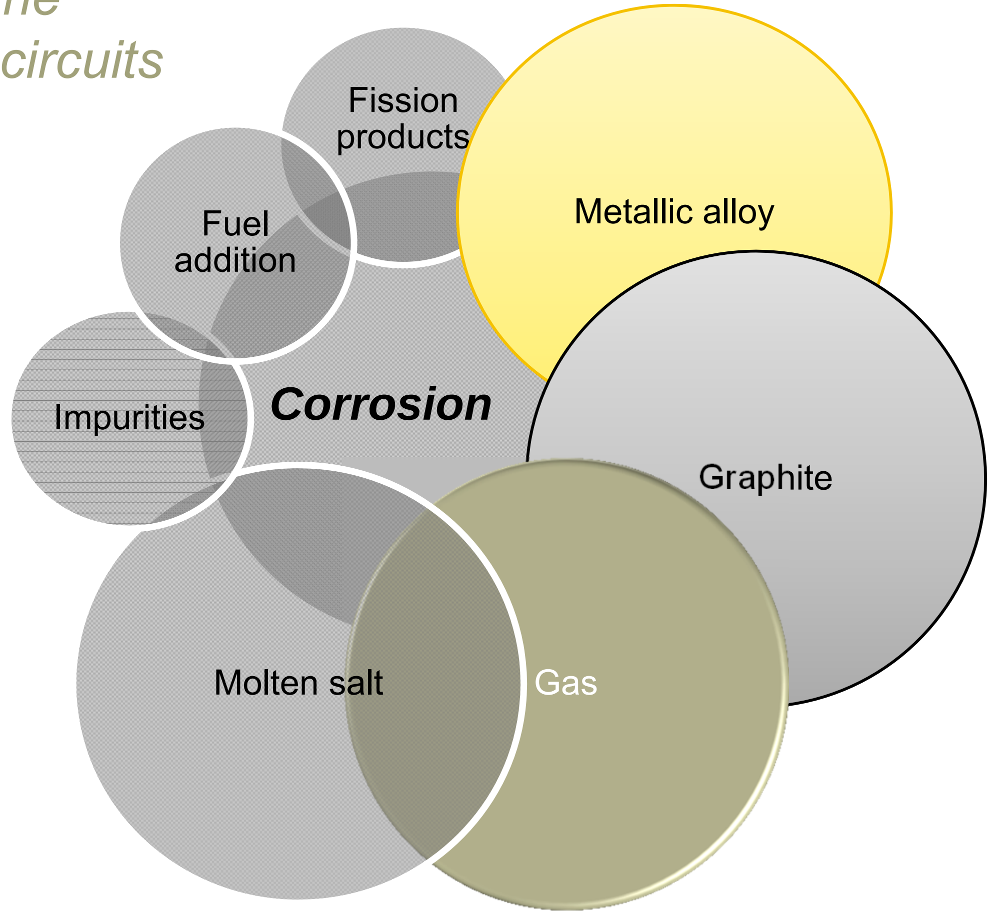


Radiation
Fast neutrons

High Temp
Creep, Creep-fatigue, Thermal fatigue, Aging, Welds…

Corrosion
Redox, Heat up, Velocity …

Corrosion effects

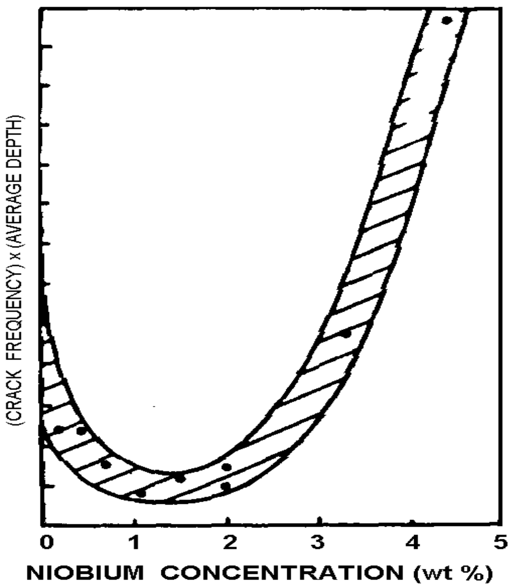


Fission products
Fuel addition
Metallic alloy
Impurities
Corrosion
Graphite
Molten salt
Gas

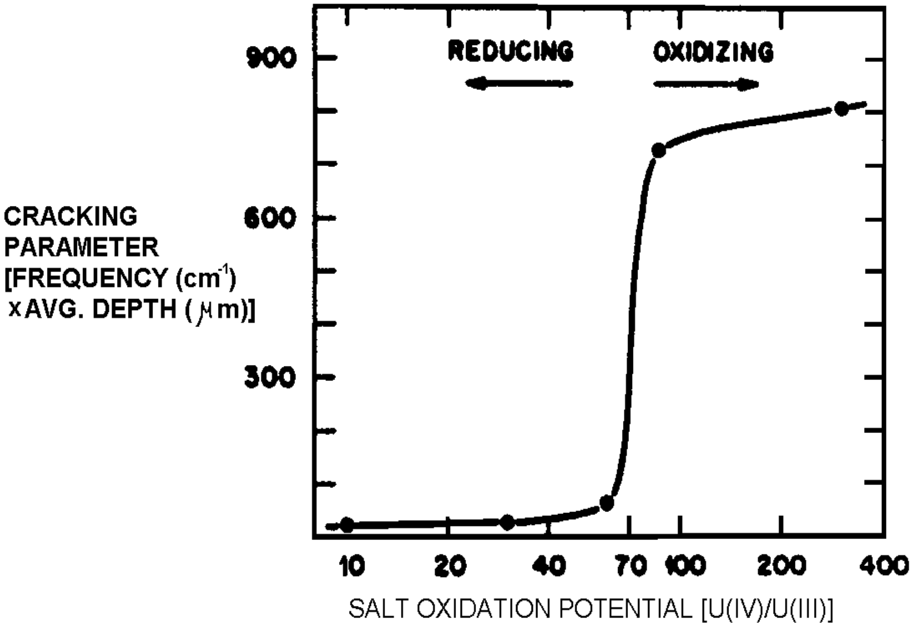
Because the products of oxidation of metals by fluoride melts are quite soluble in corroding media, passivation is precluded, and the corrosion rate depends on other factors, including: <u>Oxidants, Thermal gradients, Salt flow rate, Galvanic coupling</u>

MSRE Experience

- Hastelloy N was susceptible to damage from He formation in a fast neutron flux. Later it was found that modified alloys that had fine carbide precipitates within grains would hold He and restrain this migration to the grain boundaries
- The reasoning in 1970’s was that the 2% titanium addition would impart good resistance to irradiation embrittlement and that the 0 to 2% niobium addition would impart good resistance to intergranular tellurium embrittlement.
- Hastelloy N was deteriorating in its properties as a result of intergranular attack caused by Te. Later work showed that this Te attack could be controlled by keeping the fuel on the reducing side. This is done by adjustment of the chemistry so that: about 2%UF_3 as opposed to 98%UF_4



Effect of Nb in modified Hastelloy N grain boundary cracking when exposed in salt Cr_3Te_4 + Cr_5Te_6 for 250 hr at 700 °C

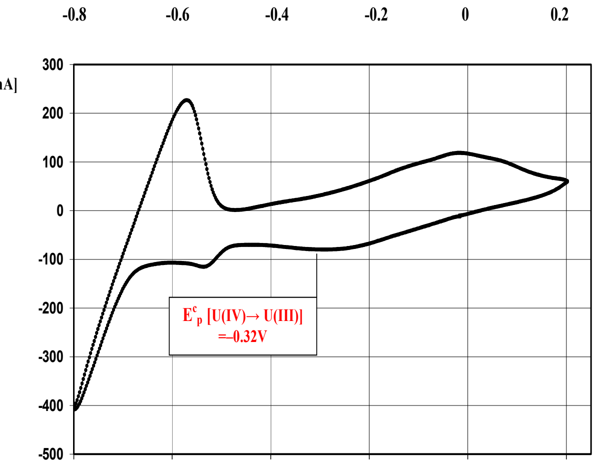


In temperature range 500-800^{0}C about 70 differently alloyed specimens of HN80MT were tested. Among alloying elements there were W, Nb, Re, V, Al and Cu

Element	Hasteloy N US	Hasteloy NM US 1972	Hasteloy NM US 1976	HN80M-VI Russia	HN80MTY Russia	MONICR Czech Rep	E-721 France
Ni	base	base	base	base	base	base	base
Cr	7,52	7,3	7,3	7,61	6,81	6,85	8
Mo	16,28	13,6	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	0,1	0,28	0,15	2,27	0.63
Mn	0,52	0,14	0,14	0,22	0,013	0,037	0.26
Nb	-	-	1 - 2	1,48	0,01	< 0,01	-
Si	0,5	< 0,01	< 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

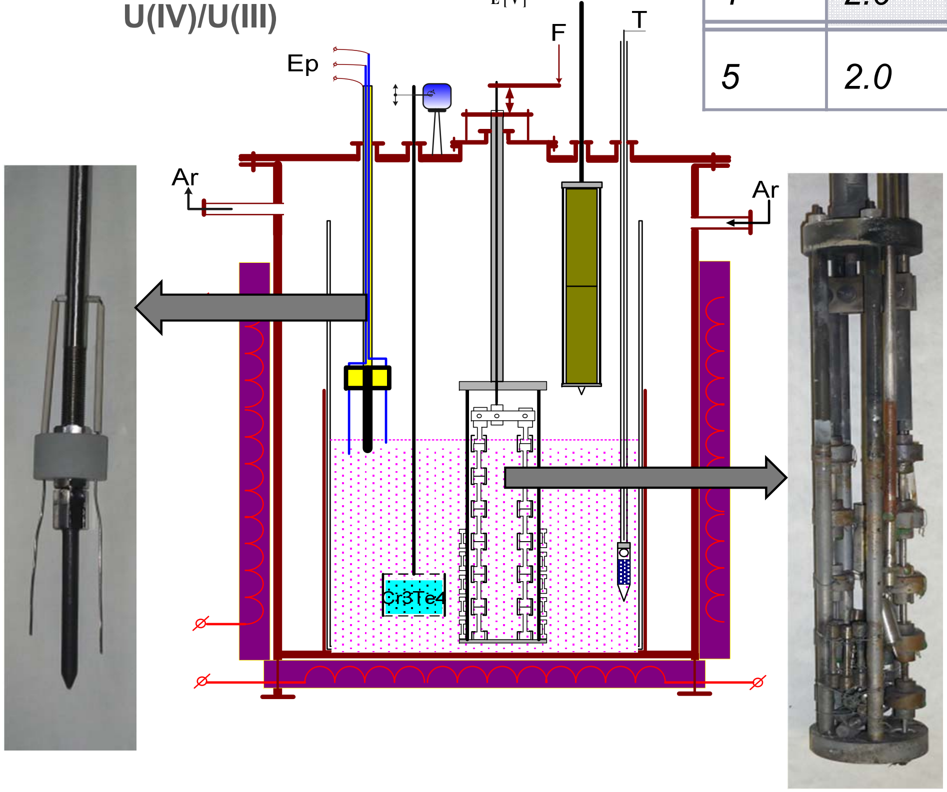
- *Experiments results in polythermal loops with redox potential measurement demonstrated that operations with Li,Be/F salt, also fuelled by UF$_4$ or PuF$_3$, are feasible using carefully purified molten salts and loop internals.*
- *Russian HN80MTY alloy with 1% added aluminum is the most resistant with fuel Na,Li,Be,Pu/F; Li,Be,U/F; Li,Th,U/F and Li,Be,Th,U/F salt mixtures up to temperature 750°C with [U(IV)]/[U(III)] ≤ 100. Corrosion rate was <5µm/yr. No intergranular corrosion of alloy 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



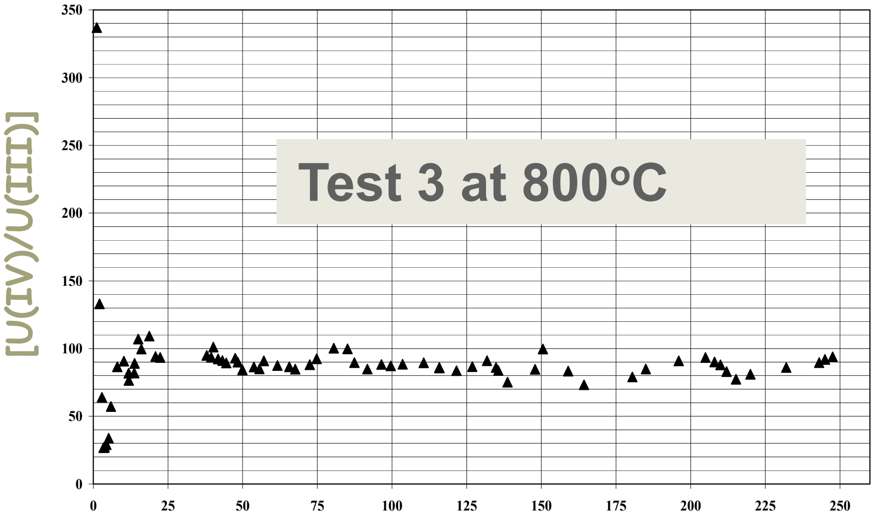
U(IV)/U(III)

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

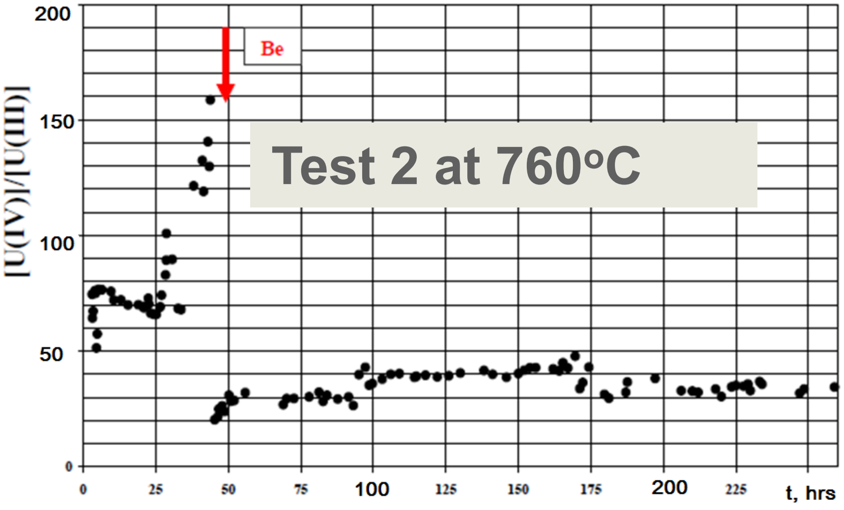


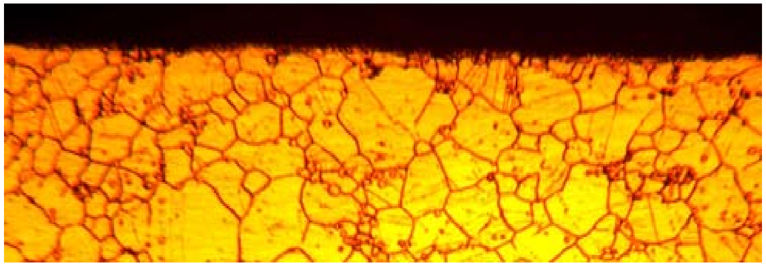
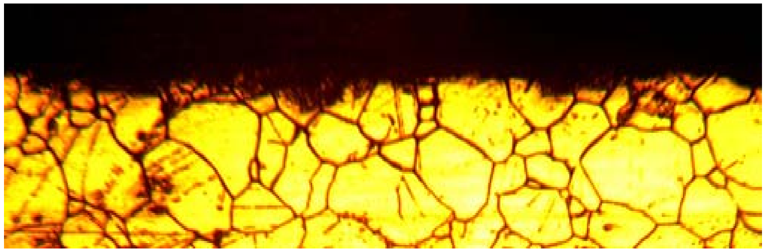
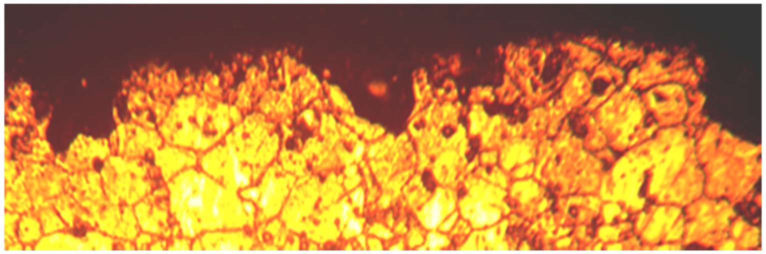
- *The experimental facility was developed to study compatibility of Ni-based alloys under various mechanical loads to the materials specimens with fuel salts containing Cr_3Te_4 with redox potential measurement*
- *$LiF-BeF_2-ThF_4-UF_4$: 5 tests of 250 hrs each at fuel salt temperature till to 750°C 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*

N-alloys Compatibility With Fuel Salts Strongly Depends on Redox Potential

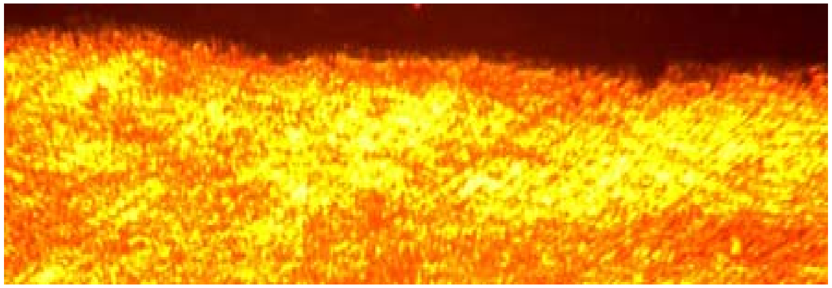
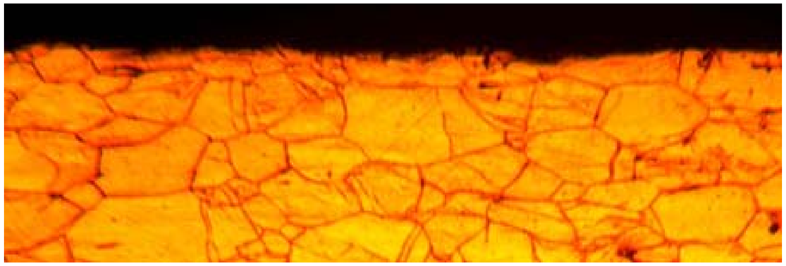
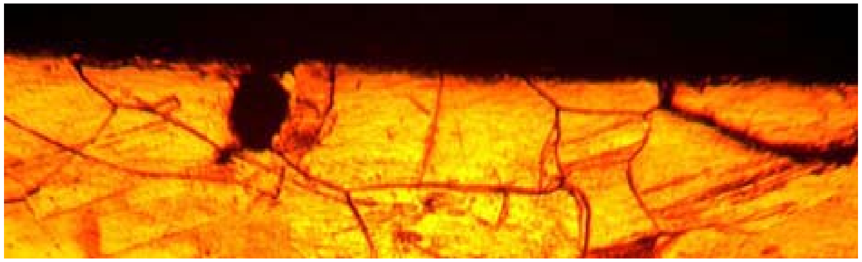
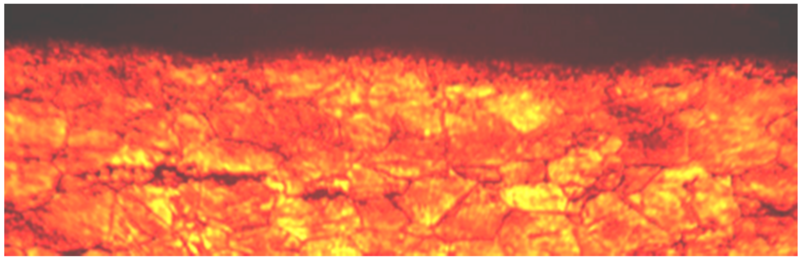
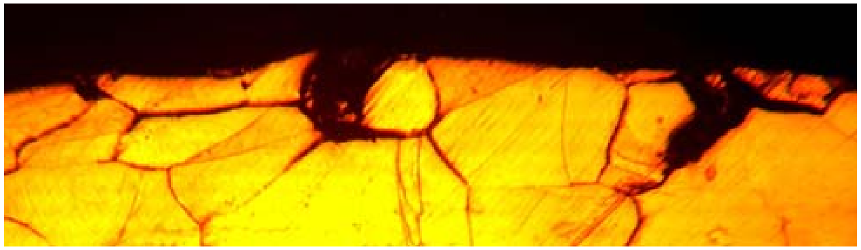
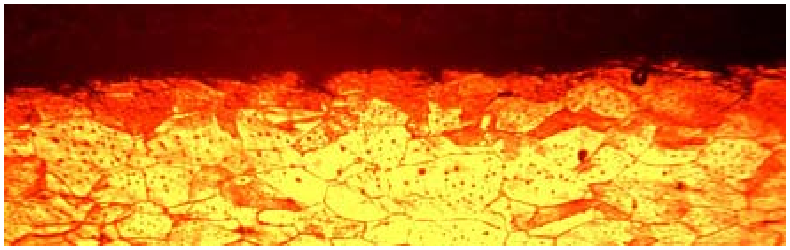


Decrease of RedOx potential by metallic Be

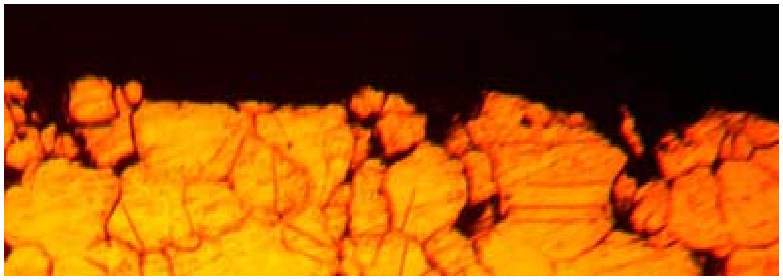
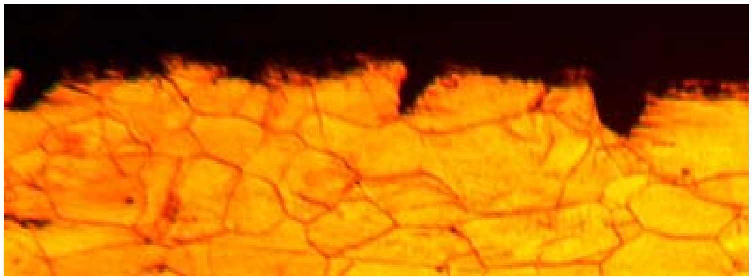
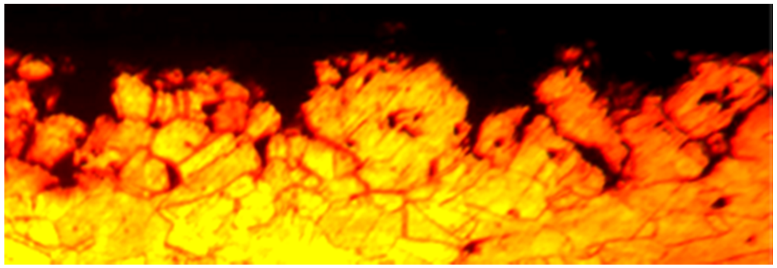
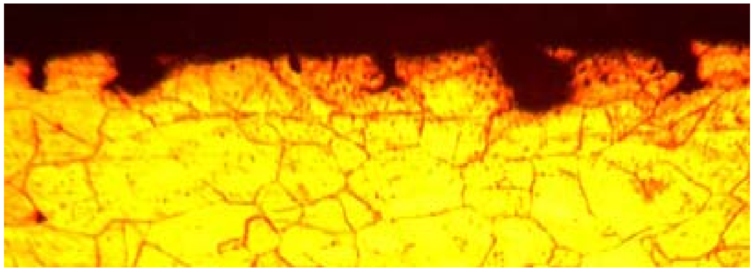
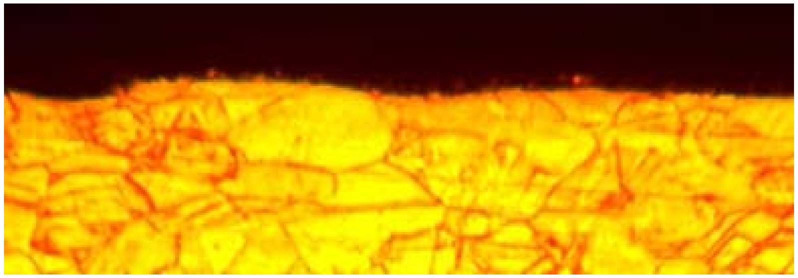
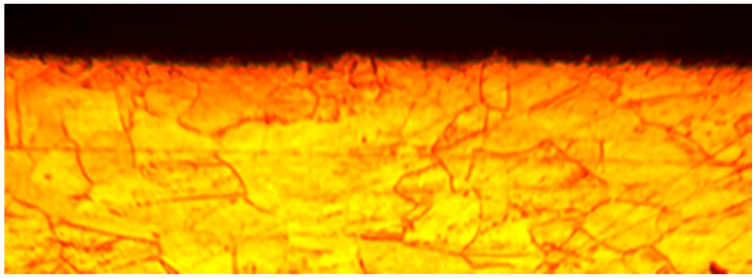


Li,Be,U/F U(IV)/(UIII)	*Hastelloy N, enlargement ×160*
30 without loading at 760°C	*No* 
60 without loading at 760°C	*K = 3500pc×µm/cm ; l = 69µm* 
90 without loading at 800°C	*K = 4490pc×µm/cm ; l = 148µm* 

N-alloys Compatibility With Fuel Salts Strongly Depends on Redox Potential

U(IV)/(UIII)	EM-721, enlargement ×160	HN80MTY, enlargement ×160
30 *without loading at 760^{o}C*	*No* 	*No* 
60 *without loading at 760^{o}C*	*K = 3380pc ×μm/cm ; l = 117μm* 	*No* 
90 *without loading at 800^{o}C*	*K = 5830pc ×μm/cm ; l = 286μm* 	*K = 530pc ×μm/cm ; l = 26μm* 

N-alloys Compatibility With Fuel Salts Strongly Depends on Redox Potential

Li,Be,Th,U/F	HN80MT-VI	HN80MTY
[U(IV)]/[U(III)] 500 without loading at 735°C	K =3360pc×μm/cm; l =166μm 	K=1660pc×μm/cm; l=68μm 
[U(IV)]/[U(III)] 500 Loading 25MPa 750°C	K =8300pc×μm/cm; l =180μm 	K = 1850pc×μm/cm ; l=80μm 
[U(IV)]/[U(III)] 100 Loading 25MPa 750°C	no 	no 

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).*
- *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.*
- *The innovative and commercially unproven nature of MSRs, as with many other advanced reactor concepts, presents significant challenges and risks in terms of financing, licensing, construction, operation and maintenance.*

Preparative Chemistry and Salt Purification

Most suppliers of halide salts do not provide materials that can be used directly

The major impurities that must be removed to prevent severe corrosion of the container metal are moisture/oxide contaminants

Once removed, these salts must be kept from atmospheric contamination by handling and storage in sealed containers

During the US MSR program, a considerable effort was devoted to salt purification by HF/H_2 sparging of the molten salt. In addition to removing moisture/oxide impurities, the purification also removes other halide contaminants, such as chloride and sulfur.

- *In our purifications the gaseous agent (HF) was in some cases replaced by solid ammonium hydrofluoride (NH_4HF_2, $T_m \approx 125\ ^0C$), which is safer and more convenient in use for the removal of impurity oxide compounds from metal fluorides and for the conversion of U and Th oxides to fluorides.*

$$MeO_2 + 4NH_4HF_2 \rightarrow MeF_4 + 4NH_4F\uparrow + 2H_2O\uparrow\ (T_{max} = 350\text{-}400\ ^0C)$$

- *To carry out these processes do not require expensive equipment and special safety measures. The purified anhydrous fluorides of metals was obtained, which are used for the preparation of fluoride salt melts of different composition.*

Production of anhydrous constituents->Melting -> Filtration -> Zone recrystallization -> met.Th or Zr or Be treatment

Hydrogen Isotopes Behavior

Main strategies for mitigation include: advanced materials for the piping and heat exchangers, inert gas sparging, additional coolant lines and metal hydride addition or chemical removal

Material	*Temperature / Pressure*	*Equations Activation energy in kJ/mole*
HN80MTY *Mo-13,2%$_{mass}$*	*473…1123K* *0.1…90 кPa*	*Diffusion* $D=1{,}1\cdot10^{-7}\exp(-47{,}7/R\cdot T)$, $[m^2/s]$ *Permeability* $P=3{,}5\cdot10^{-8}\exp(-55{,}0/R\cdot T)$, $[\,mole/m\times s\times Pa^{1/2}\,]$ *Solubility* $Ks=0.41\cdot\exp(-8{,}85/R\cdot T)$ $[mole/\,m^3\times Pa^{1/2}\,]$
HN80MTY oxidation film	*Same*	*Permeability* $P=3{,}8\cdot10^{-8}\exp(-48{,}8/R\cdot T)$, $[\,mole/m\times s\times Pa^{1/2}\,]$
EM721 *W-25,2%$_{mass}$*	*Same*	*Diffusion* $D=4{,}5\cdot10^{-7}\exp(-45{,}7/R\cdot T)$, $[m^2/s]$ *Permeability* $P=3{,}7\cdot10^{-8}\exp(-55{,}4/R\cdot T)$, $[\,mole/m\times s\times Pa^{1/2}\,]$ *Solubility* $Ks=0.13\cdot\exp(-9{,}6/R\cdot T)$ $[mole/\,m^3\times Pa^{1/2}\,]$
$77LiF-6ThF_4-17BeF_2$ *Tm =560°C (833K)*	*973-1073K* *4-70 кPa*	*Diffusion* $D=7{,}7\cdot10^{11}\exp(-393{,}5/R\cdot T)$, $[m^2/s]$ *Permeability* $P=1{,}5\cdot10^{-15}\exp(-38{,}8/R\cdot T)$, $[\,mole/m\times s\times Pa^{1/2}\,]$ *Solubility* $Ks=2{,}0\ \exp(-390{,}1/R\cdot T)$ $[mole/\,m^3\times Pa^{1/2}\,]$

- *Additional development of permeation-resistant coatings, including W-Si*
- *Ultrasonic degassing to facilitate removal of tritium, reducing required total bubble volume for gas sparging*
- *Refinement of geometric configuration of the intermediate heat exchangers, minimizing tritium flux*
- *Discovery of reusable solvents for direct tritium removal from molten salt*
- *The chemistry of sodium fluoroborate and the trapping process by which tritium is retained by the salt*