Progress in ISTC-1606

Presented by
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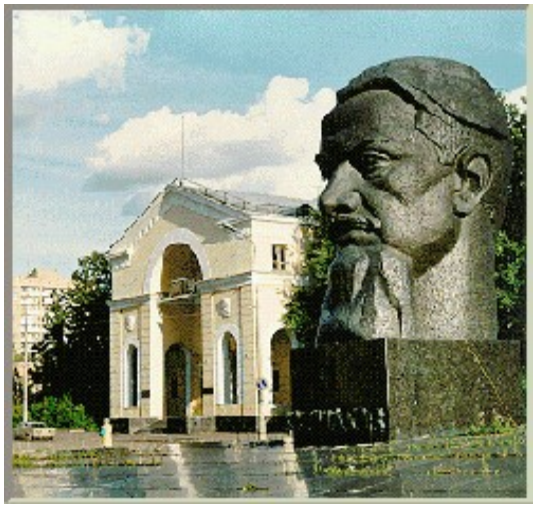


for ISTC CEG meeting, Brussels, EC,
January 22-23, 2007

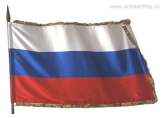
Integrated Task#1606

Supported by EU from 2001 – 2004 (Phase 1)– 2007 (Phase 2)

Partners of project consortium: RRC-KI, VNIITF , VNIIkHT, IHTE

- At current project stage focus is placed on experimental and theoretical evaluation of MOlten Salt Actinide Recycler & Transmuter (MOSART) system fuelled with different compositions of TRU's from LWR spent fuel without U-Th support
- Basic objective is to provide the fissile concentration and geometry of the fuel salt to obtain heat release of about 2400 MWt at conditions affording the effective transmutation of TRU's from LWR spent fuel

Cooperation with partners



ROSATOM Transmutation Task

ISTC#1606

Foreign collaborators:

CEA, BNFL, EdF, FZK, IAEA, KTH, NRI

IAEA CRP on "Studies of Advanced Reactor Technology Options for Effective Incineration of RadWaste" in 2004 -2007 , Domain VI MOSART concept:

RRC-KI, CEA, FZK, NRG, SCK CEN, Polito Univ, NRI

Euratom 5th FP MOST project, 2002-2004
Euratom 6th FP EUROPART, ALISIA, 2006-2007

ISTC "Training in modern experimental and analytical methods for study of actinide-containing molten salts properties" in 2006-2007

US Liquid Salt R&D activity
ORNL, UC - Berkeley

Integrated Task#1606

New experimental data received in our studies feed into MOSART design efforts as well as Gen IV R&D

- WP 1: Choice of molten salt system for detailed studies
- WP 2: Reactor physics, thermal hydraulics and fuel cycles consideration
- WP 3: Measurement of key physical and chemical properties for selected fuel salts
- WP 4: Experimental verification of Ni-Mo alloys for fuel circuit in corrosion facilities with on-line redox measurement
- WP 5: Final report and work plan development for new ISTC LIST Project
- WP 6: Supporting tasks

Phase 1: Choice of molten salt system

- Low neutron cross section for neutron typical of the chosen energy spectrum
- *Adequate solubility of fuel and FP's components*
- Thermal stability of the salt components
- Low vapour pressure
- Radiation stability
- Adequate melting point & transport properties
- Chemical compatibility with construction materials
- Low waste and cost fuel salt clean up

Note: Molten salts fluorides were developed originally for fluid fuel MSR to reflect *Gen II*, but not *Gen IV* objectives.

Phase 1: Useful salt composition

Alkali fluorides

LiF
743°C, 20 mole %

NaF
727°C, 24 mole %

LiF-KF
(50-50) 492°C

LiF-RbF
(44-56) 470°C

LiF-NaF-KF
(46.5-11.5-42) 454°C

LiF-NaF-RbF
(42-6-52) 435°C

ThF4-based

LiF-ThF4
(71-29) 555°C
3.69 mole %

LiF-BeF2-ThF4
(71-16-13) 499°C
1.41 mole %

LiF-BeF2-ThF4
(64-20-16) 460°C
1.21 mole %

LiF-BeF2-ThF4
(47-51.5-1.5)
360°C

ZrF4-based

LiF-ZrF4
(51-49) 509°C

NaF-ZrF4
(59.5-40.5) 500°C
1.8 mole %

LiF-NaF-ZrF4
(42-29-29) 460°C

LiF-NaF-ZrF4
(26-37-37) 436°C

NaF-RbF-ZrF4
(33-24-43) 420°C

RbF-ZrF4
(58-42) 410°C

KF-ZrF4
(58-42) 390°C

NaF-KF-ZrF4
(10-48-42)
385°C

BeF2-based

LiF-NaF-BeF2
(15-58-27) 479°C
1.94 mole %

LiF-BeF2
(66-34) 458°C
0.47 mole %

LiF-BeF2-ZrF4
(64.5-30.5-5) 428°C

NaF-BeF2
(57-43) 340°C
0.26 mole%

LiF-NaF-BeF2
(31-31-38) 315°C
0.43 mole %

Fluoroborate

KF-KBF4
(25-75) 460°C

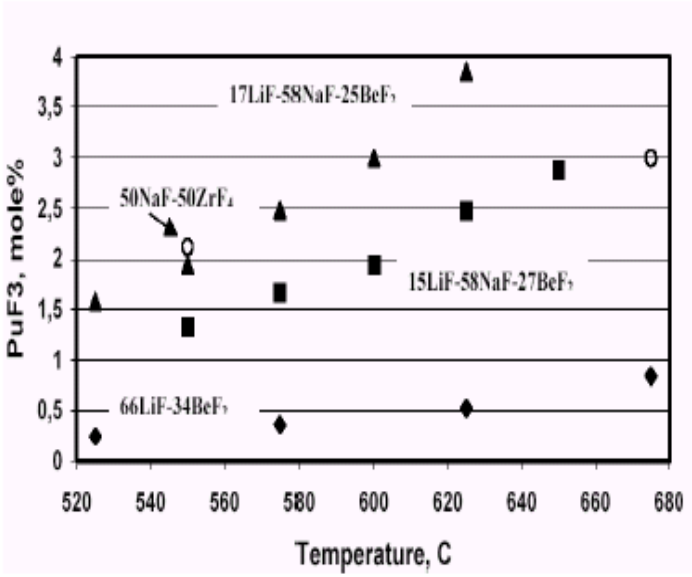
RbF-RbBF4
(31-69) 442°C

NaF-NaBF4
(8-92) 384°C

Phase 1: Physical properties

LiF	NaF	BeF_2	T_{melt}, ^0C
7	64	29	515-536
13	58	29	486-500
15	58	27	479
17	58	25	494-496
22	57	21	570

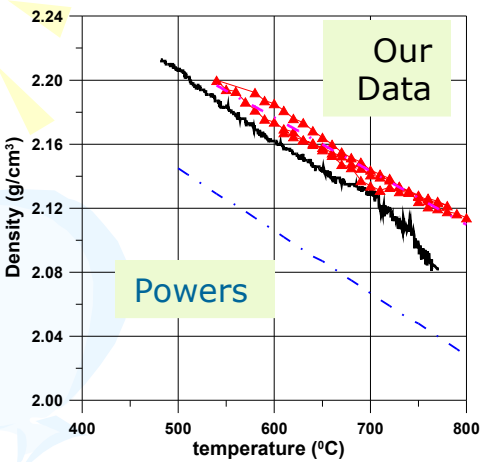
Composition, mole%	$(58NaF\text{-}15LiF\text{-}27BeF_2)+1.1AnF_3$	600C
PuF_3 solubility, mole%	$\ln P = 7.49 - 5936/T[K]$	1.94
Viscosity, m^2/s	$\nu = 0.136 \cdot \exp\{2914/T[K]\}$	3.75
Heat capacity, $J \cdot kg^{-1} \cdot K^{-1}$	$C_p = 2090$	2090
Thermal conductivity, W/(mK)	$\lambda = 0.838+0.0009\ (t[C]-610.3)$	0.837
Density, g/cm^3	$\rho = 2.163-0.406(t[C]-601.4)\bullet10^{-4}$	2.164
Vapor pressure, Pa	$\ln p=18.92-1.4710^{-4}T(K)-25283/T+0.982\ln T$	< 0.1

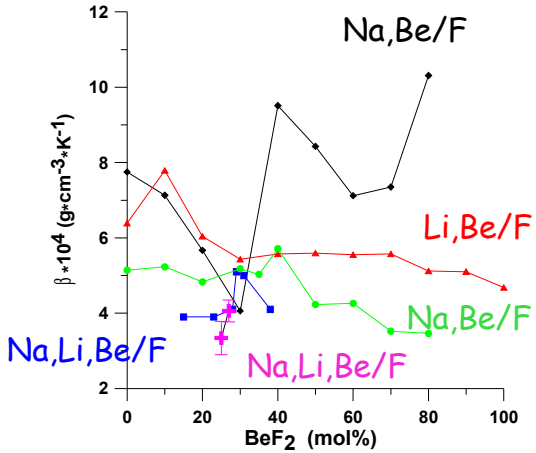


It is important from technical point of view that, as result of ISTC#1606 Phase 1 for molten Na,Li,Be/F system, was found quite wide range with minimal of LiF (17-15 mole%) and of BeF_2 (25-27mole%) content in the ternary composition, which provide fuel salt able to get solubility of PuF_3 till to 2 in mole% or even more at 600^0C, to keep adequate melting point (<500°C) and very low vapour pressure, to have good nuclear properties (neutron transparent salt), low activation, suitable transport properties and to be moderately expensive (about 25$ per kg).

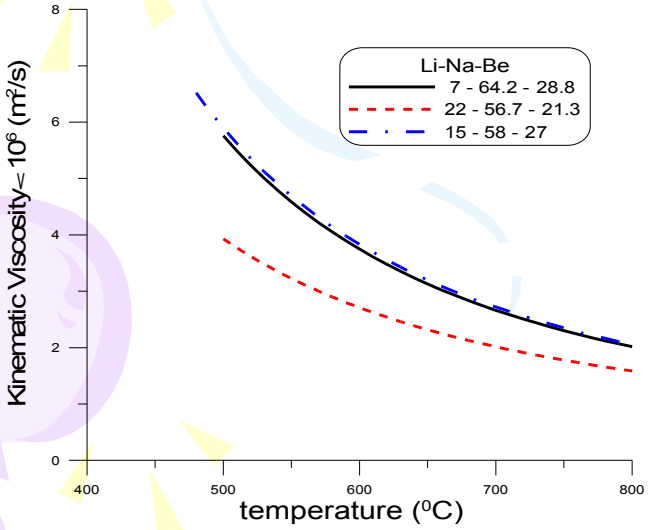
Phase 2: WP3

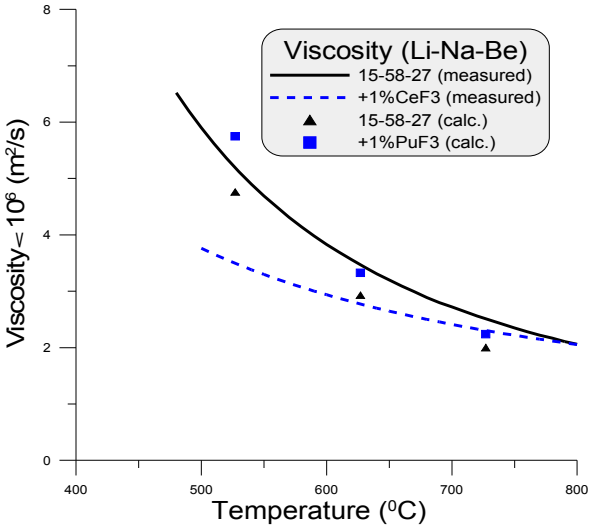
Measurement of density for molten 17LiF-58NaF-25BeF_2 and viscosity for 15LiF-58NaF-27BeF_2, effect of CeF_3





Density of the melts has been measured by hydrostatic weighing method in temperature range 482 - 770 C. The mistake of measurement is estimated as 0.9 %.





Viscosity of different molten Li,Be,Na/F mixtures have been measured by method of attenuation torsional oscillations of cylinder in a temperature range from freezing up to 800C. Accuracy of measurement is 4-6 % (dispersion).

Phase 2: Fuel salt clean up

- Reductive extraction: $Ln\ (in\ LM) + AnF_3 \rightarrow LnF_3 + An\ (in\ LM)$
- Electrodeposition:

- on liquid electrodes: $LM\ (Ln) + AnF_3 \rightarrow LnF_3 + LM\ (An)$

- on solid electrodes: $Ln + AnF_3 \rightarrow LnF_3 + An$

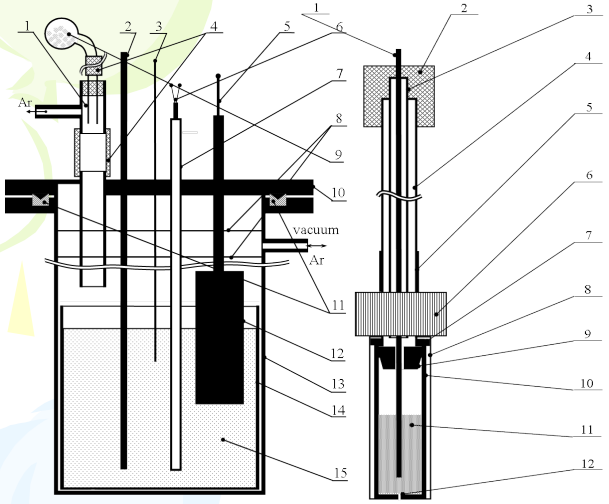
- Oxides precipitation: $Ln_2O_3 + AnF_3 \rightarrow LnF_3 + An_2O_3$

$Ln_2O_3 + AnF_4 \rightarrow LnF_3 + AnO_2$

$LnO_2 + AnF_4 \rightarrow LnF_4 + AnO_2$

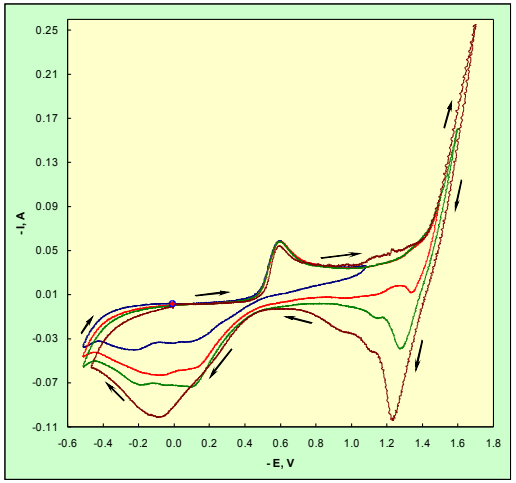
- Development of $Ni\text{-}NiF_2$ reference electrode based on boron nitride container for electrochemical studies in Na,Li,Be/F
- Studies on electrochemical properties of Pu and RE's and their trifluorides in molten Li,Na/F and Na,Li,Be/F
- Investigation of the influence of metal solvent nature on An/Ln separation in case of liquid Bi. Measurements of RE's and Pu distribution in system (liquid metal-Li) – (Na,Li,Be/F)
- Determination of PuF_3 free energy of formation on base of standard potential measurements solid-state galvanic in cells

Development of Ni-NiF_2 reference electrode based on boron nitride container for electrochemical studies in Na,Li,Be/F



$Ni|NiF_2$ reference electrode

1–electrode made of metallic nickel; 2 – vacuum-rubber plug; 3 – porcelain straw; 4 – stainless-steel tube; 5 – threaded bushing with flange welded to tube (4); 6 – clamping nut; 7 – bushing (5) flange; 8 – cup made of nickel foil; 9 – cover made of PBN-coated graphite; 10 – PBN-coated container; 11 – standard melt; 12 – capillary



Cyclic voltammograms measured at nickel working electrode in $(LiF\text{-}NaF\text{-}BeF_2)_{eutec}+NiF_2$

T = 856 K;

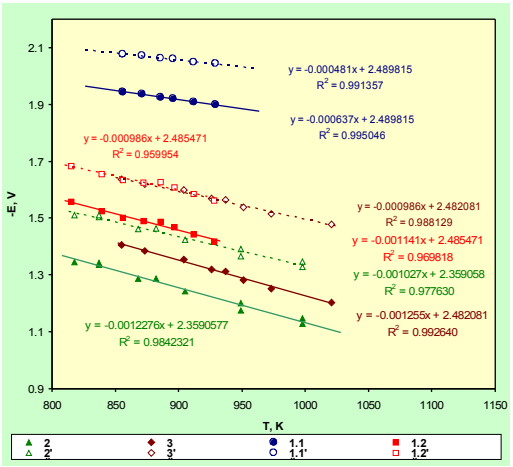
$C_{NiF2} \rightarrow 0$;

scan rate = 0.2 V/s;

reference electrode:

Ni wire in $(LiF\text{-}NaF\text{-}BeF_2)_{eutec.}$ $+NiF_2$ ($C_{NiF2} \leq 0.5$mole %);

auxiliary electrode: Ni crucible



Polytherms of Be and its alloys measured relative to nickel reference electrode in $(LiF\text{-}NaF\text{-}BeF_2)_{eutec}+NiF_2$

(1.1) εNa/Ni

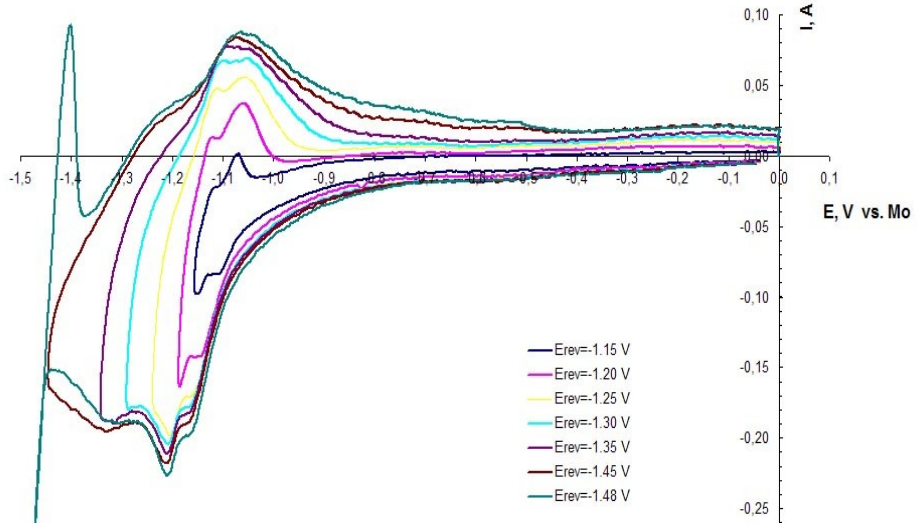
(1.2) εBe/Ni CNiF2=1.353×10^{-2}mole%

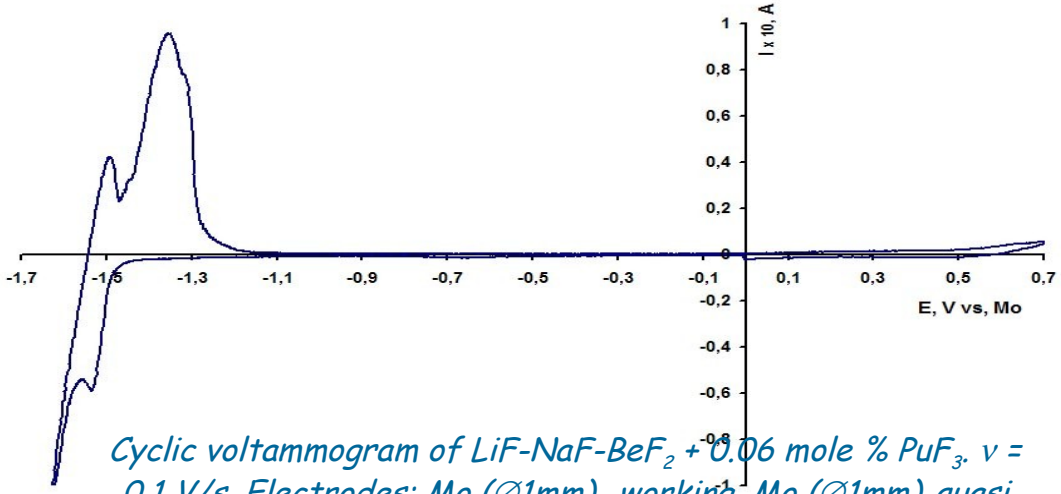
(2) εBeNi/Ni; Ni and dynamic beryllium electrodes were immersed in one and the same melt; CNiF2=4.774×10^{-3}mole%

(3) εBeMo/Ni; BeO container, protected with a nickel foil, served as the diaphragm separating the half-elements; CNiF2 in the RE half-element was 9.683×10^{-4} mol%

Electrochemical properties

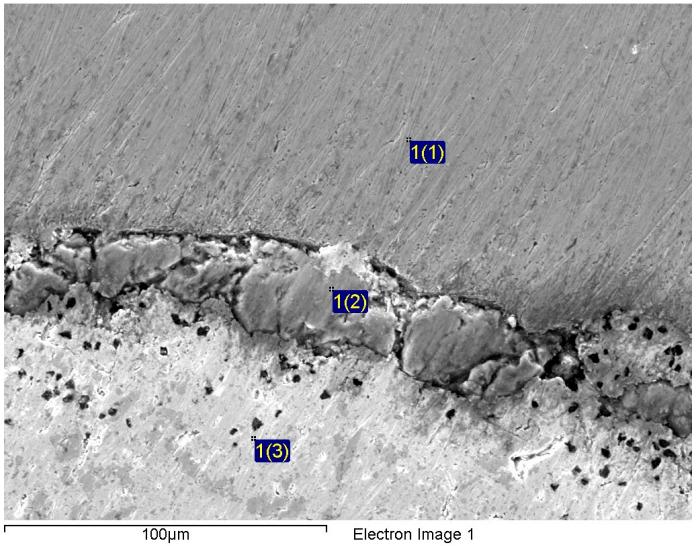
CV of PuF_3 (0.2 mole %) in LiF-NaF melt for various reversal potential values. v = 0.2 V/s. WE - Mo, RE – Mo. T=1023K.



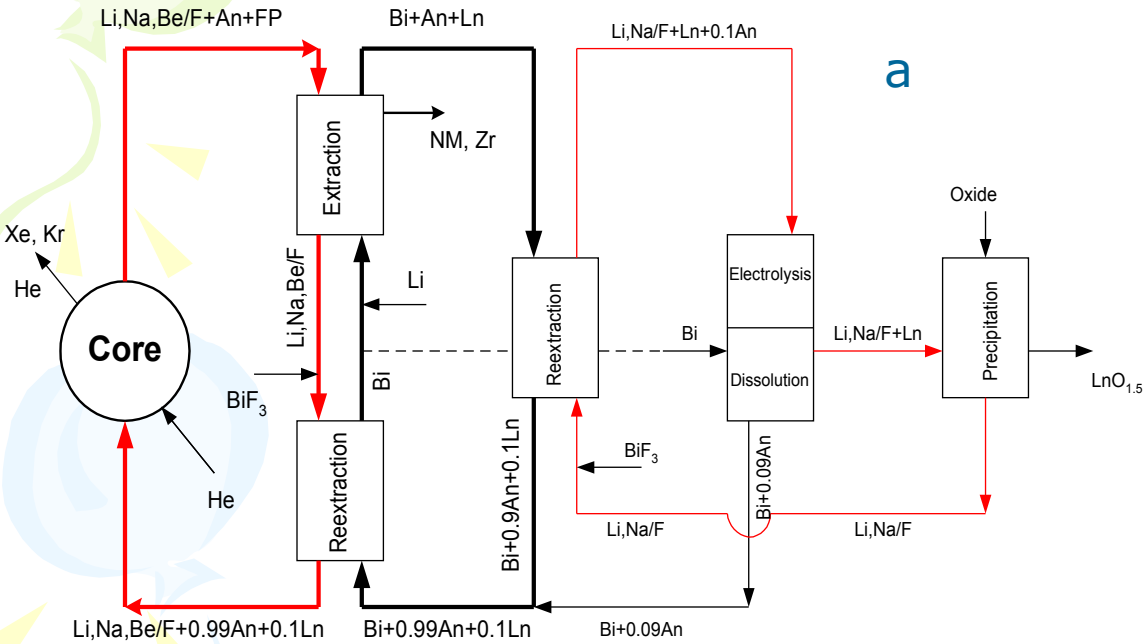


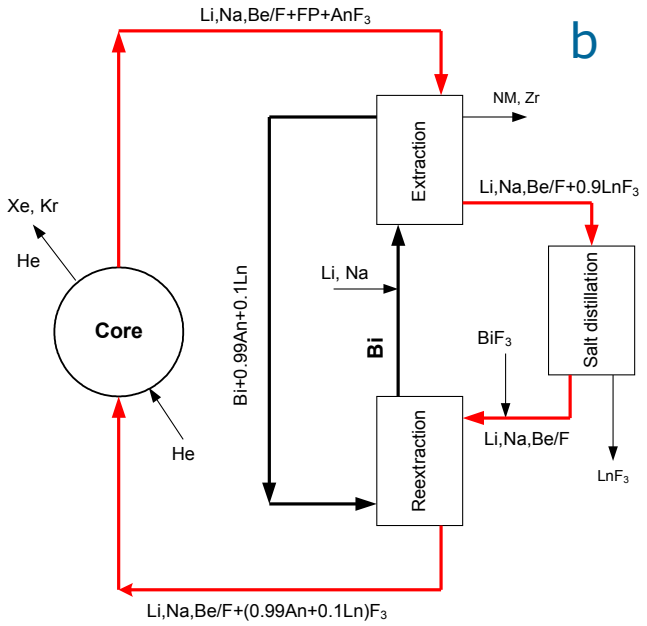
Cyclic voltammogram of LiF-NaF-BeF_2 + 0.06 mole % PuF_3. v = 0.1 V/s. Electrodes: Mo (∅1mm)- working, Mo (∅1mm)-quasi ref., graphite – auxiliary. T=853K, argon.

Equilibrium 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;
Diffusion coefficients of zirconium and plutonium ions in 15LiF-58NaF-27BeF_2 and 60LiF-40NaF melts were estimated;
Assumption was made concerning the alloying of plutonium with Be, Na and Ni in the process of Pu ions electrochemical reduction in fluoride melts.



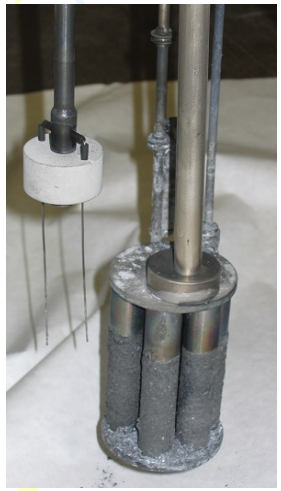
Na,Li,Be/F MOSART fuel clean up





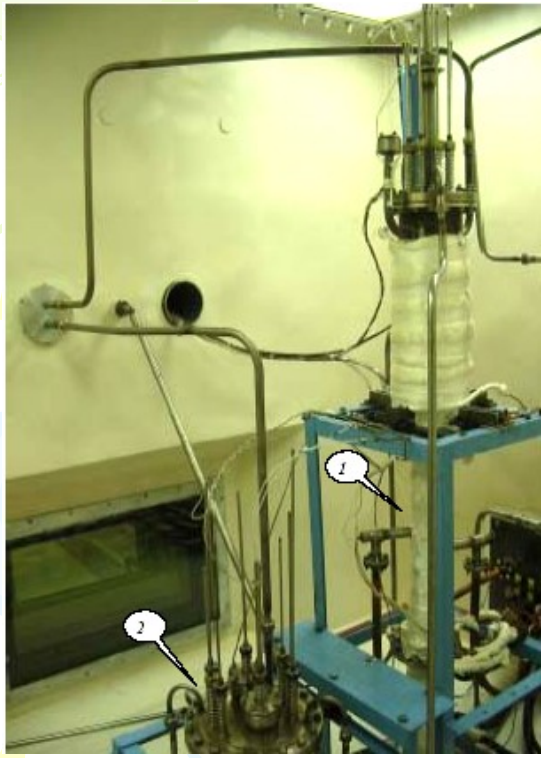
- Partial extraction of Na from molten salt into liquid Bi was found, however, Na flow could be sufficiently decreased by its preliminary addition into liquid Bi stream.
- At the first stage all An's and some of FP's (noble metals, Zr) are co-extracted into liquid Bi. At the second stage the fuel salt could be purified from Ln's by distillation. Further, An's are recycled to the core for transmutation.
- The lanthanides are separated from solute components in the MOSART processing flowsheet by method of salt distillation. This method of Ln's removal is under consideration, because in case Ln's extraction process from Li,Be,Na/F system the large amount of Na would be removed into liquid Bi.

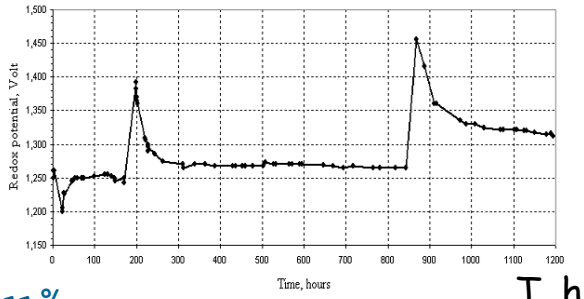
Phase 2: Corrosion studies



- Compatibility test (>1500h) between molten 15LiF-58NaF-27BeF_2 salt (mole %) and Ni - based alloys in thermal convection loop with measurement of Redox potential
- Effect of PuF_3 addition in molten 15LiF-58NaF-27BeF_2 salt (mole %) system on compatibility with Ni - based alloys
- Te corrosion study between molten 15LiF-58NaF-27BeF_2 salt and Ni-Mo alloys
- Detailed examination of Ni - based alloys specimens properties data after exposition

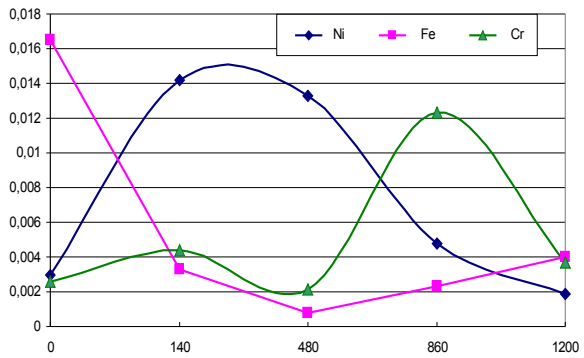
Loop corrosion studies





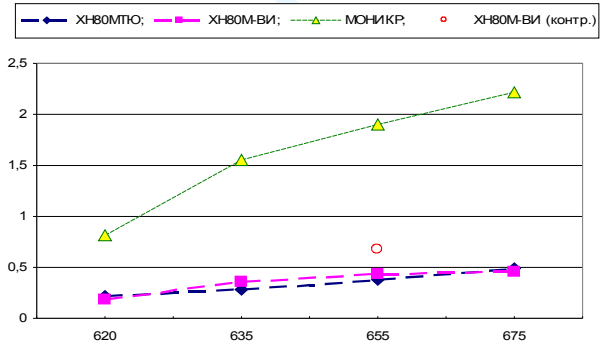
T, hr

Mass.%



58NaF-15LiF-27BeF$_2$

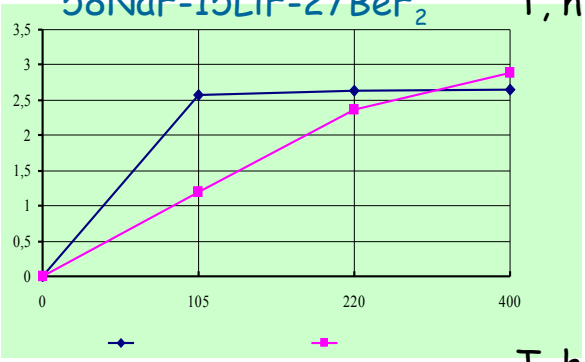
T, hr



58NaF-15LiF-27BeF$_2$

T, C

- mg/cm2



T, hr

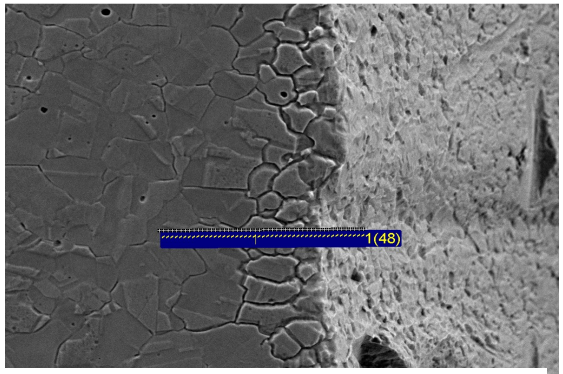
(58NaF-15LiF-27BeF$_2$) + 0,5PuF$_3$

- Corrosion rate of HH80M-VI and H80MTY alloys in the melt was 2-5μm/yr and 9-19 μm/yr for the MONICR alloy.
- It was not found any significant change in corrosion behavior of materials samples in melt due to the presence of 0.5 mole % PuF3 addition in Li,Be,Na/F salt.
- Specimens of HN80M-VI from the loop exposed during 400 hrs at temperature 650C showed uniform corrosion rate of about 6 μm/yr.
- No intergranular corrosion of alloys is observed in the loop .

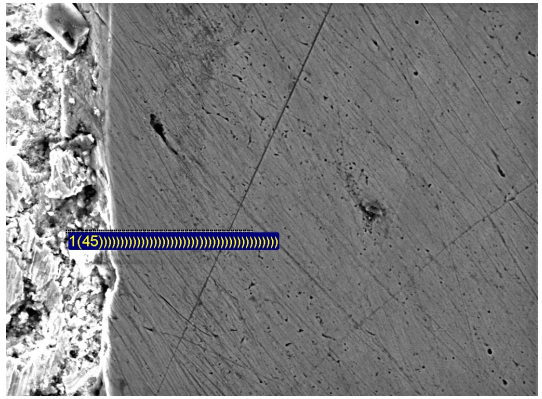
Corrosion studies

Surface microstructure of specimens after 1200 hrs in Li,Na,Be/F loop at temperature 690C

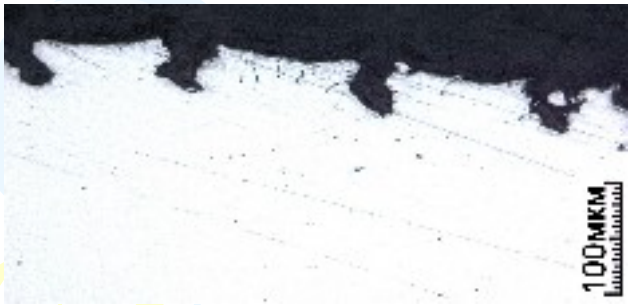
HN80M-VI



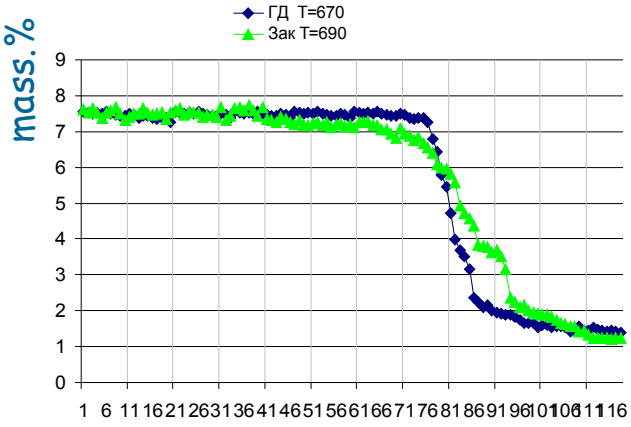
HN80MTY

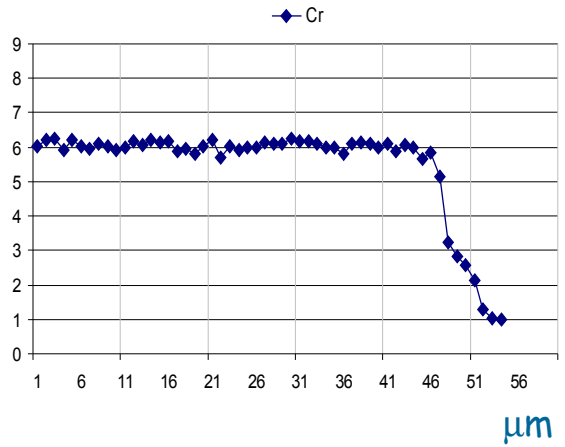


MONICR



15LiF-58NaF-27BeF2+ Cr3Te4
K = 30 pc·μm/cm
after 100h at 650C





In first set of ampoule Te corrosion test after 100hr exposition at 650C in 15LiF-58NaF-27BeF2 melt containing (in mass.%) respectively: Te<0,01; Ni – 0.008; Fe –0.02; Cr< 0.001 traces of intergranular cracking were found only on MONICR samples (K = 30 pc·μm/cm).

Phase 2: Reactor physics, thermal hydraulics and fuel cycle

Scenario 1: TRU from UOX spent fuel of a commercial PWR (60 GWd/tU - 4.9% 235U/U; after 1 year cooling)

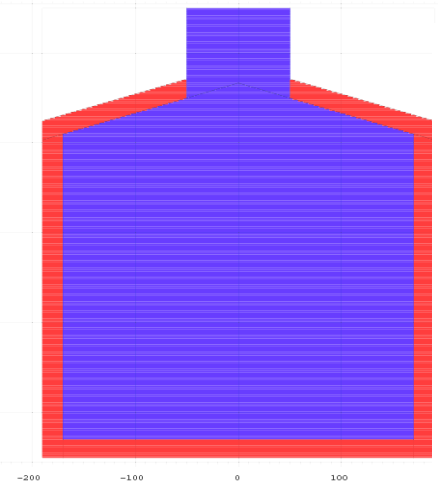
Scenario 2: 4,9% 235U/U irradiated in PWR up to 60 GWd/tHM (Stage 1); after 7 years cooling the Pu from the spent fuel it is used for MOX fuel production with natural uranium and 7% Pu; after 3 additional years of MOX fuel production, that fuel also irradiated in PWR up to 60 GWd/tHM (Stage 2). Remaining TRU, including MA from Stage 1, after 10 yrs cooling are MOSART fuel

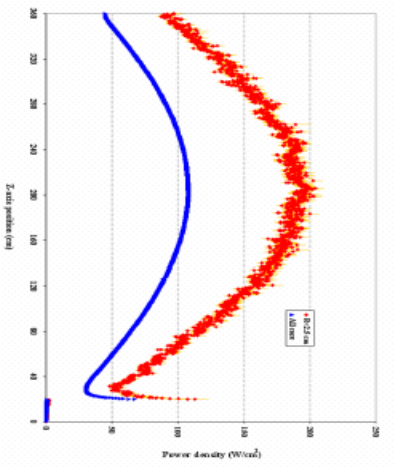
Component	Cycle times	Removal operation
Kr, Xe	50 sec	He Sparging
Zn,Ga,Ge,As,Se,Nb Mo,Cd,InSn,Sb,Te, Ru,Rh,Tc	2.4 hr	Plating out on surfaces + To off gas ystem
Zr	1-3 yrs	Reductive extraction, Oxide precipitation, Electrodeposition Fluoride Na,Li,Be/F distillation
Ni, Fe, Cr		
Np, Pu, Am, Cm		
Y,La,Ce,Pr,Nd, Pm,Gd,Tb,Dy, Ho Er Sm Eu		
Sr, Ba, Rb, Cs	>30 yr	
Li, Be, Na		Salt discard

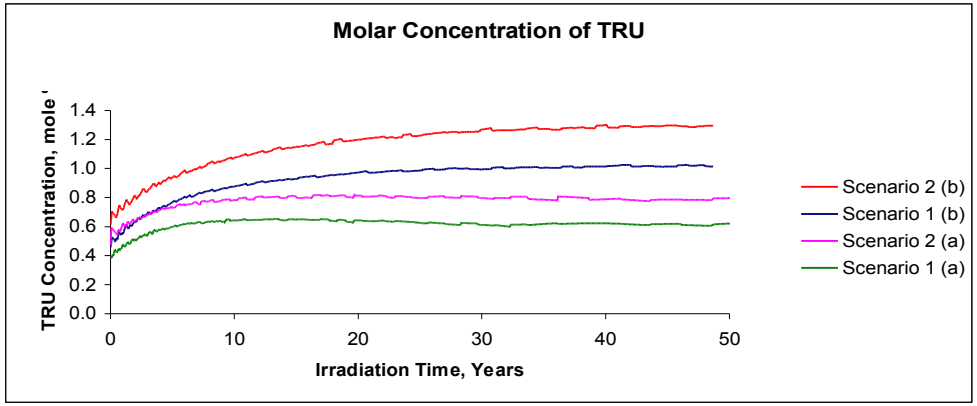
- Coupled thermal hydraulic and neutronic 3D calculations of selected MOSART system operated in forced convection
- Studies on main neutronic, fission products behavior and safety related parameters
- Design optimization using new data on fuel salt / core properties received

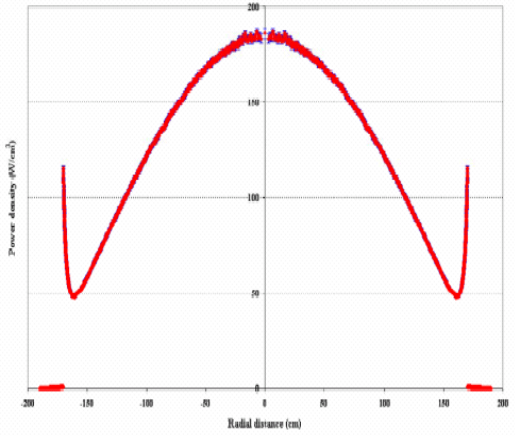
2400MWt MOSART core calculations

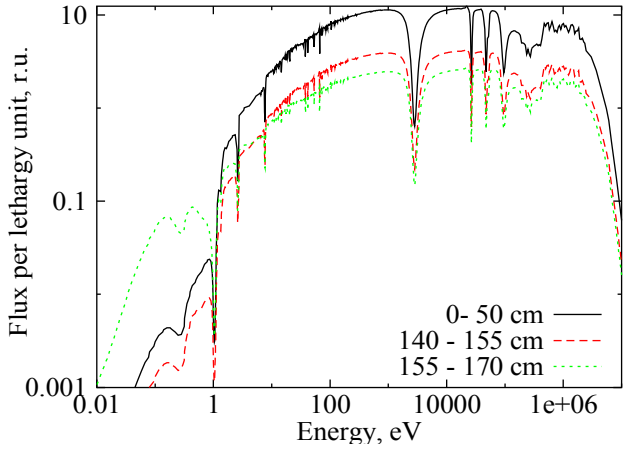
20cm graphite reflector; Scenario 1: α_{Total} =-4.125 pcm/K; Scenario 2: α_{Total} =-6.625 pcm/K

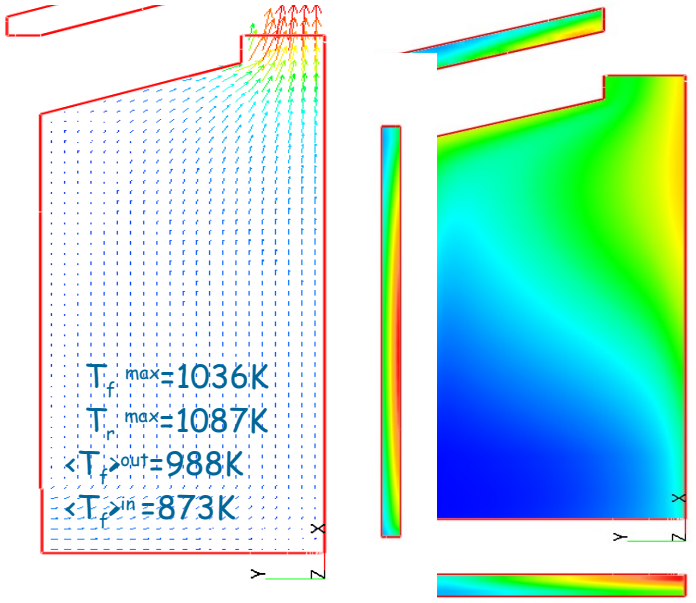












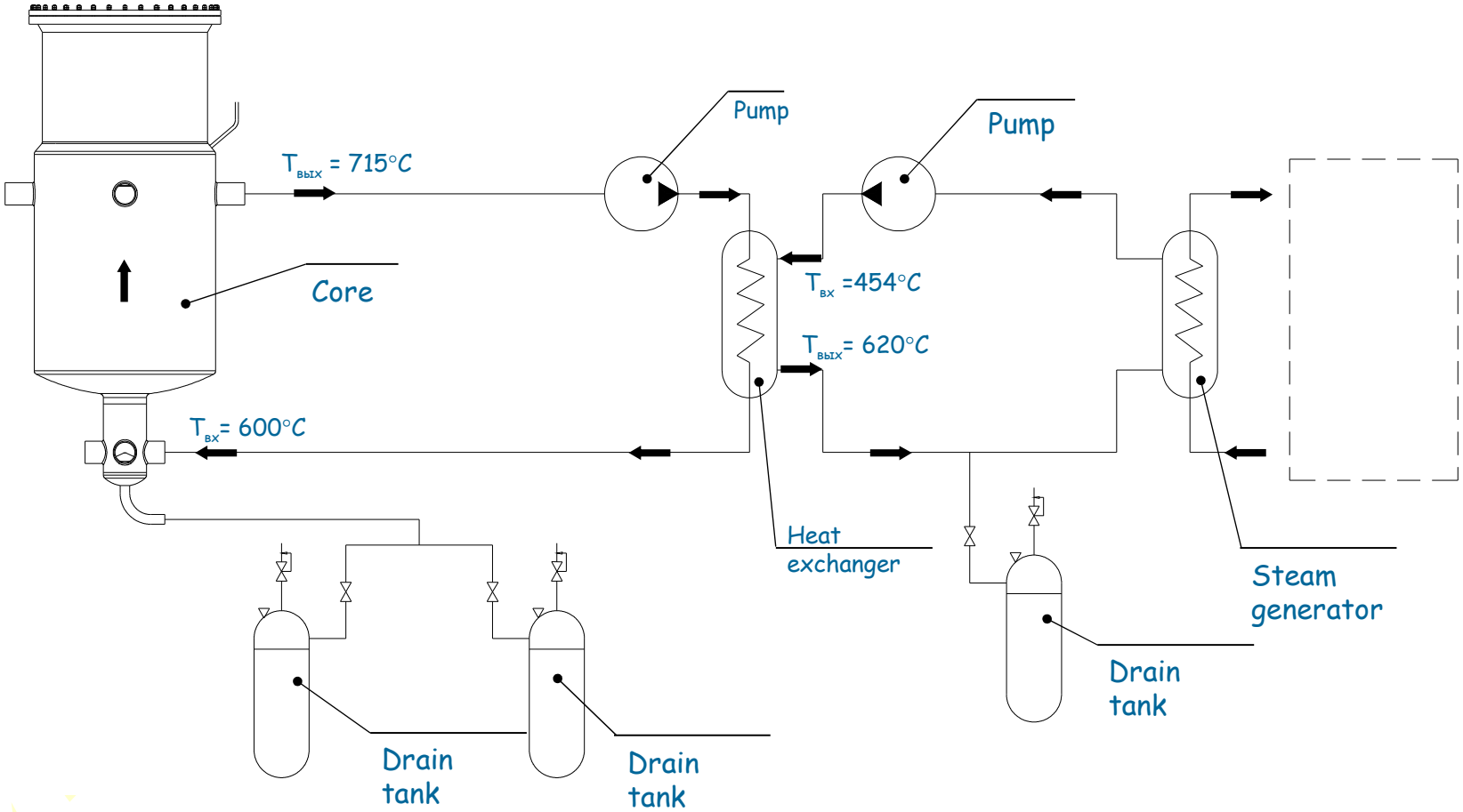
MOSART parameters comparison

Results have confirmed essential distinction in cross-sections of MA used in different libraries. At the same time the results obtained with the help of different codes, but with the use of the same nuclear data coincide rather well. The disorder in values of K-eff is near 2.5 %.

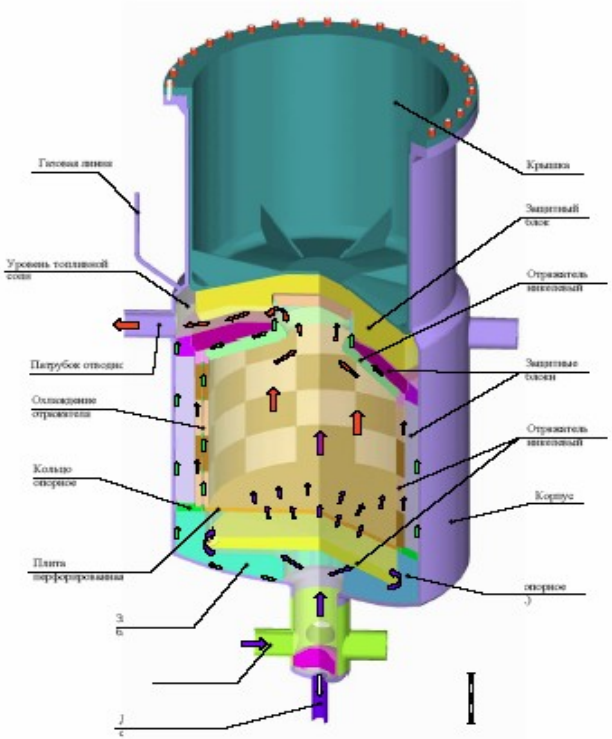
	BME MCNP4C +JEFF 3.1/ 1D 172 gr. +JEFF 3.1/ MCNP4C +JEF 2.2	FZK 2D 560 gr. JEFF 3.0/ JENDL 3.3/ ENDF 6.8/ JEF 2.2	NRG MCNP4C JEFF 3.1/ JEFF 3.0	Polito Un. 2D 4 gr. JEFF 3.1	RRC-KI MCNP4B +ENDF 5,6/ MCU +MCUDAT	SCK-CEN MCNPX JEFF 3.1
k-eff	1.00905/ 1.01984/ 0.96462	0.99285/ 1.01023/ 0.98474/ 0.96498	1.00887/ 0.99335	0.99595	0.99791/ 0.98930	1.00904/ 0,96581
a-total, pcm/K	/-3.86	-3.86/ -3.82/ -3.86	-3.75	-3.78	-3.71/ -3.41	-3.66
a-Doppler, pcm/K	/-1.67	-1.52/ -1.53/ -1.46	-1.42	-1.73	-1.62/ -1.09	-1.69
a-reflector, pcm/K	/-0.05	-0.05		-0.04		
Generation time, μs		8.3/ 8.2	11.6	8.8		8.7
β-eff, pcm		/340	323			369

MOSART flow diagram

The primary salt is circulated outside the reactor vessel through 4 loops. Each circuit contains a 1 m^3/s single stage centrifugal pump and a shell-and-tube heat exchanger. Heat is transferred from the primary salt to secondary fluid. Each of the four secondary circuits has a 1.3 m^3/s centrifugal pump with variable-speed drive.



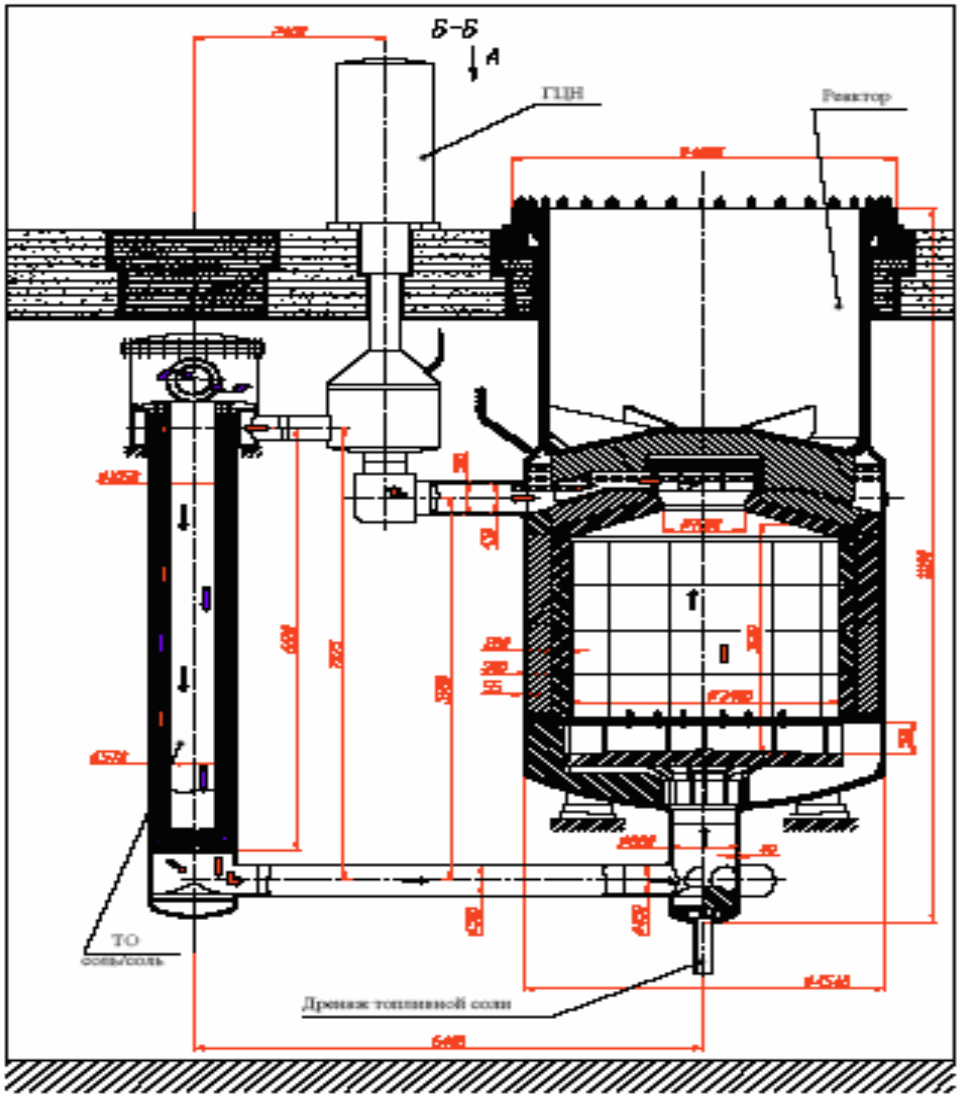
MOSART Design data



Cooling was provided for the reactor vessel and other parts of design to keep temperatures within the tolerances imposed by neutron fluence and stress conditions

	MOSART	MSBR
Thermal capacity, MWt	2400	2250
Reactor vessel ID,m	4.43	6.77
Vessel wall thickness, cm	5.5	5.1
Vessel design pressure, N/m^2	$5.2 \cdot 10^5$	$5.2 \cdot 10^5$
Core height, m	3.6	3.96
Radial thickness of reflector, cm	20	76.2
Volume fraction of salt in core	1	0.13/0.37
Average core power density, MW/m^3	75.0	22,2
Peak core power density, MW/m^3	163	70,4
Average neutron flux, $n \cdot cm^{-2} \cdot s^{-1}$	10^{15}	$2.6 \cdot 10^{14}$
Max graphite damage flux, $n \cdot cm^{-2} \cdot s^{-1}$	$1.5 \cdot 10^{14}$(>180keV)	$3.3 \cdot 10^{14}$(>50keV
Graphite temperature at max graphite damage region, K	1084	982
Estimated useful life of graphite, yrs	3-4	4
Total weight of graphite in reactor, t	20	304
Flow velocity of salt in core, m/s	0.5	1.3
Total fuel salt in reactor vessel, m^3	40.4	30.4
Total fuel salt in primary system, m^3	56.2	48.7
Cycle time for salt inventory, efpd	300	10 - 30

Materials inventory



Fuel salt, m3	MOSART	MSBR
Reactor vessel		
Core	32	19
Plenums	7.5	6.2
Annulus	1.8	3.8
Reflectors	0.1	1.4
Heat exchanger		
Tubes	6.2	7.6
Inlets /Outlets	0.6	0.8
Pump+Piping	7.4	9.3
Off gas bypass loop	0.3	0.3
Others	0.3	0.3
Total	56.2	48.7

Inventory, t	MOSART	MSBR
Primary / secondary salt	120 / 456	162 / 456
Graphite	20	304
Hastelloy NM	1280	1377
SS	3028	2840

MOSART SPECIAL MATERIALS

Fuel salt: ^{7}LiF – NaF - BeF_2 – AnF_3 (14.8 - 57.4 - 26,5 - 1.3)

- Lithium is enriched to 99,99% ^{7}Li;
- Cost: NaF-1,4 $/kg; BeF2-35 $/kg; LiF -17 $/kg (without ^{7}Li enrichment);
- V=56m3; 71,4 т NaF, 11,4т ^{7}LiF (3,1 т ^{7}Li) и 37,2т BeF_2. Total cost = 3 000 000 $

Coolant salt: NaF-$NaBF_4$ (8 –92);

- V=240m^3; cost - 5 $/kg; Total cost 2 300 000 $

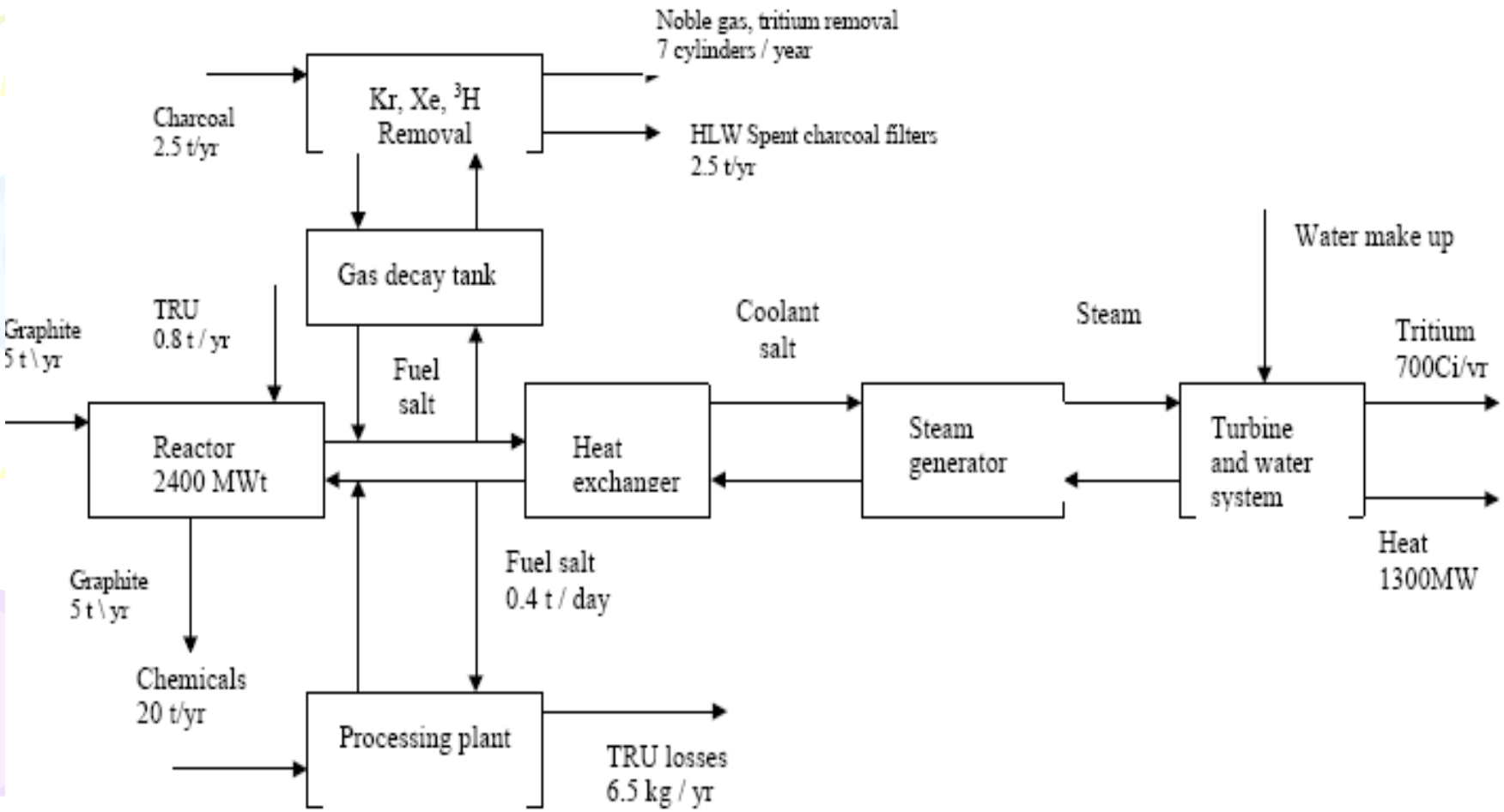
Graphite (reflector):

- molten salt should be excluded from the open pore volume (pore structure < 10^{-6}m)
- No requirement on gas permeability 10^{-8} cm^2/sec
- to be replaced at 4 yr intervals
- M=20t;

Hastelloy NM / H80MTY:

- Primary system, Secondary system, except of chemical processing system and salt storage tank
- M= 1280t; cost - 50 $/kg; Total cost = 64 000 000 $

MOSART flow diagram



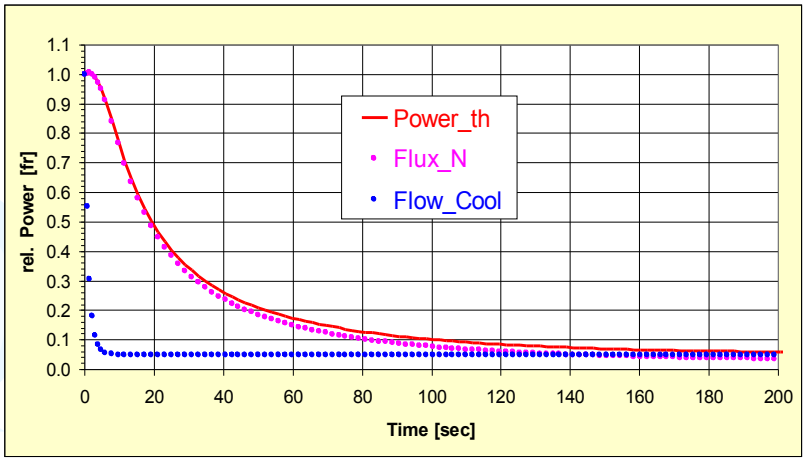
List of Transients Analyzed

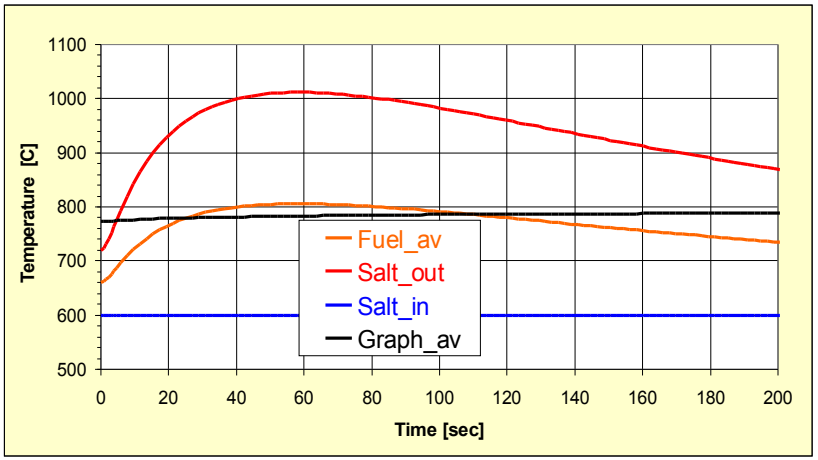
Number	Transient	Description
	MOSART	
U - 1	ULOF	loss of forced circulations in primary and secondary system, core inlet temperature is assumed to remain constant
U - 2	ULOH	loss of heat sinks (HX failure)
U - 3	over-cooling of primary side	core temperature inlet drops by 100 C in 60 sec
U - 4a	UTOP	200 pcm jump in reactivity at HFP due to an acclomerated fissile particle that is assumed to remain lodged inside the core region, coolant inlet temperature will remain constant
U - 4b	UTOP	200 pcm jump in reactivity at HFP due to an acclomerated fissile particle that is assumed to repeatedly transit the core region, coolant inlet temperature will remain constant
U - 5a	UTOP	500 pcm jump in reactivity at HFP due to an acclomerated fissile particle that is assumed to remain lodged inside the core region, coolant temperature inlet will remain constant
U - 5b	UTOP	500 pcm jump in reactivity at HFP due to an acclomerated fissile particle that is assumed to repeatedly transit the core region, coolant inlet temperature will remain constant

Unprotected Loss of Flow (ULOF)

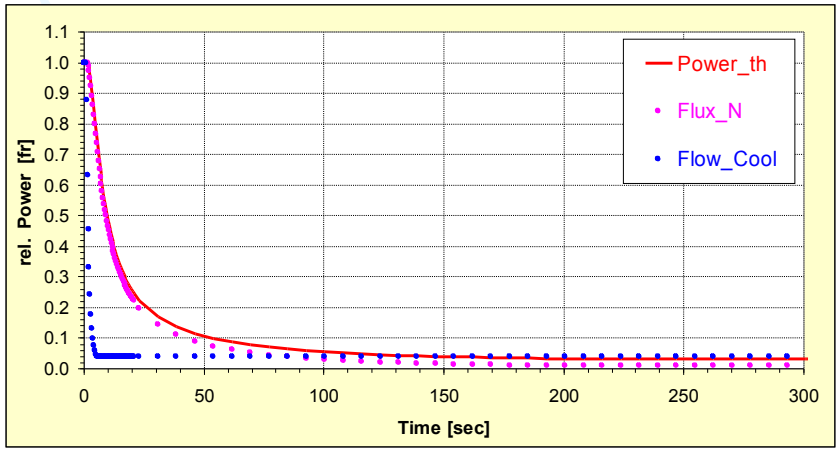
Scenario 1. Code: SIM-ADS (FZK)

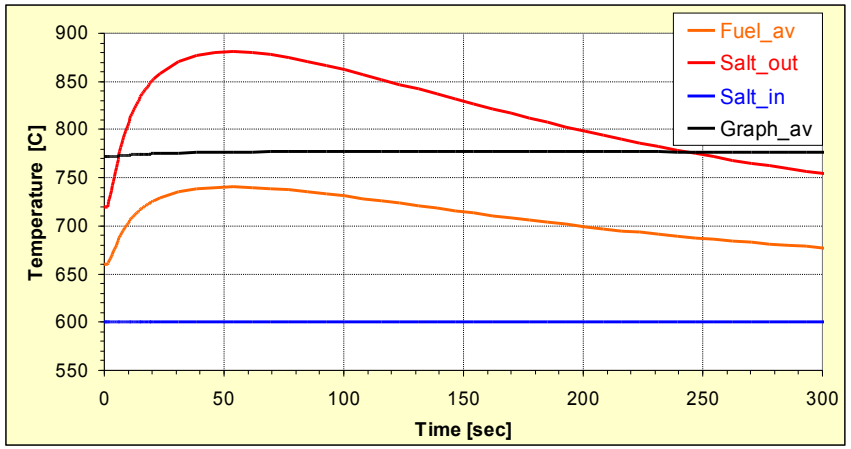
-3.41 pcm/K





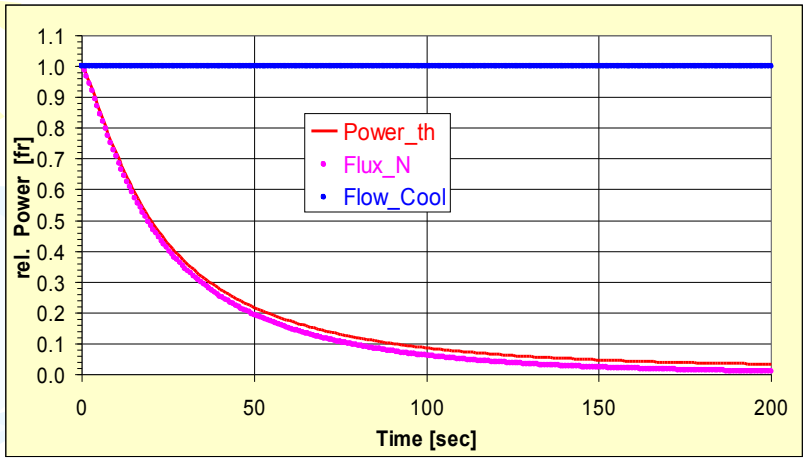
-4.125 pcm/K

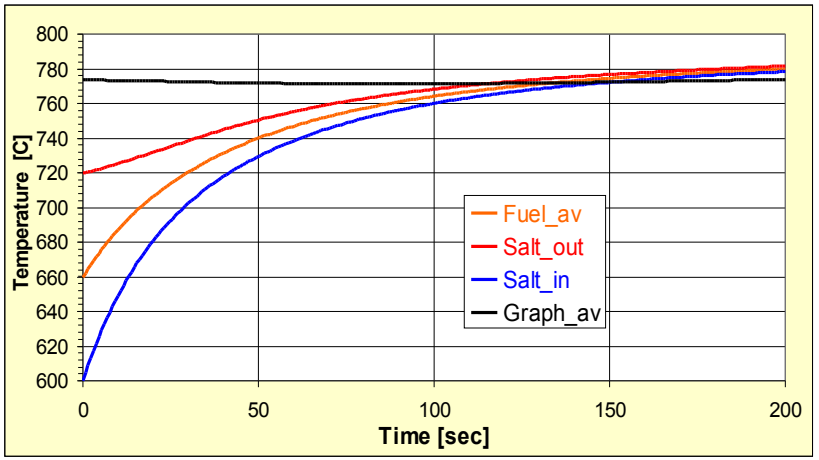




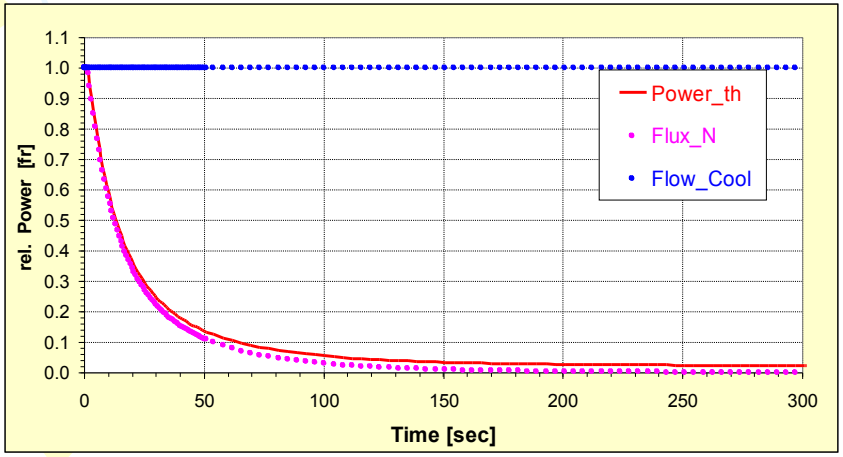
Unprotected Loss of Heat Sink (ULOH) Scenario 1. Code: SIM-ADS (FZK)

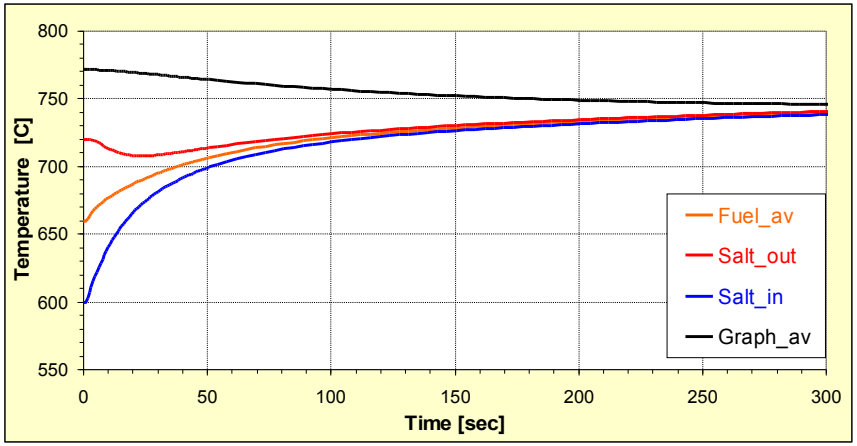
-3.41 pcm/K





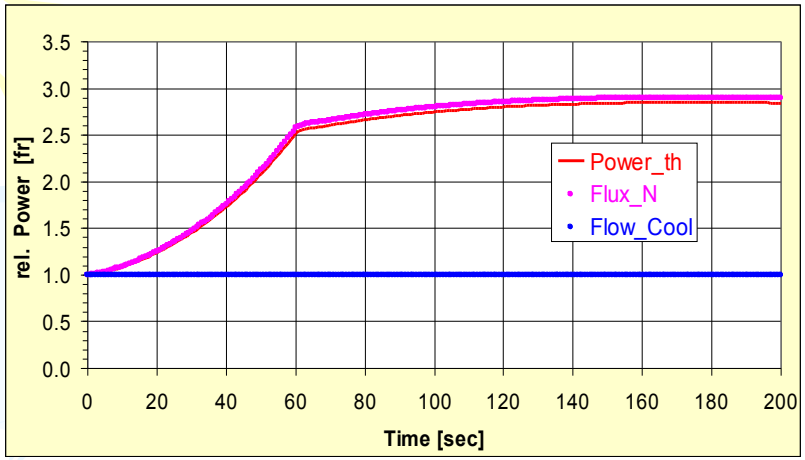
-4.125 pcm/K

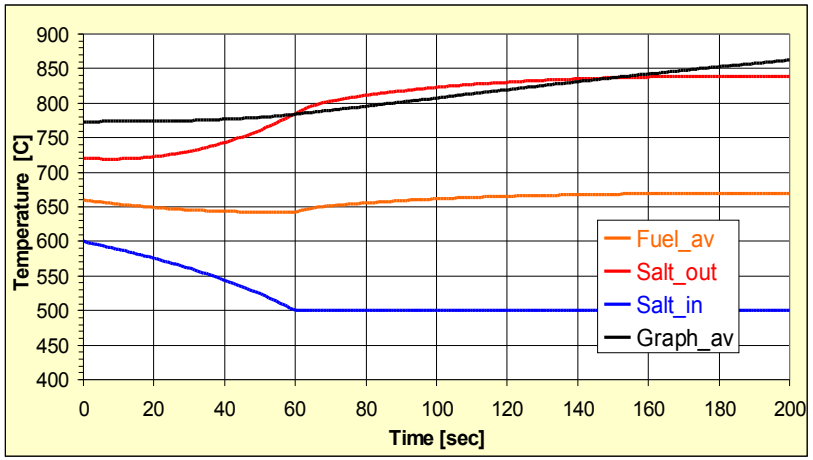




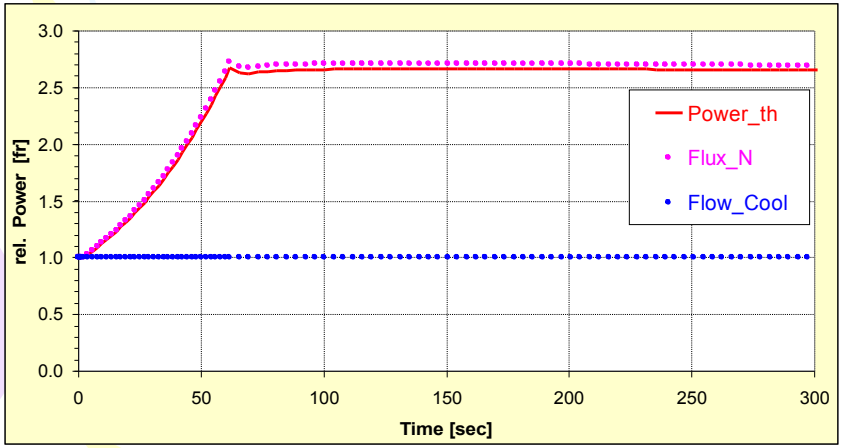
Unprotected Overcooling (UOC) Scenario 1. Code: SIM-ADS (FZK)

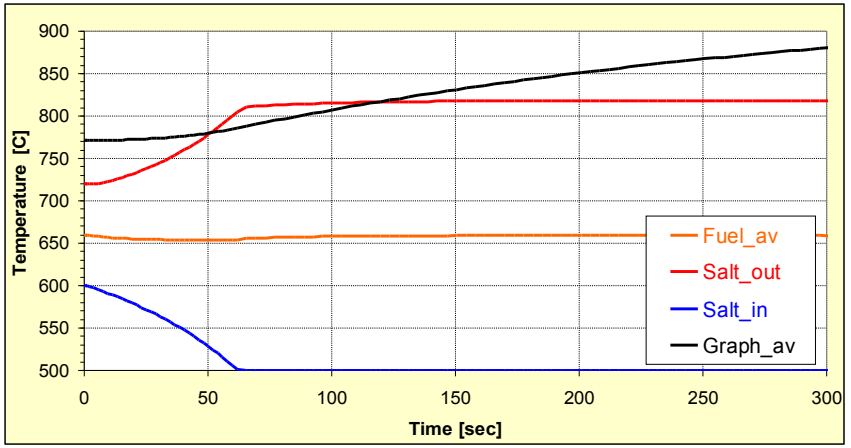
-3.41 pcm/K





-4.125 pcm/K

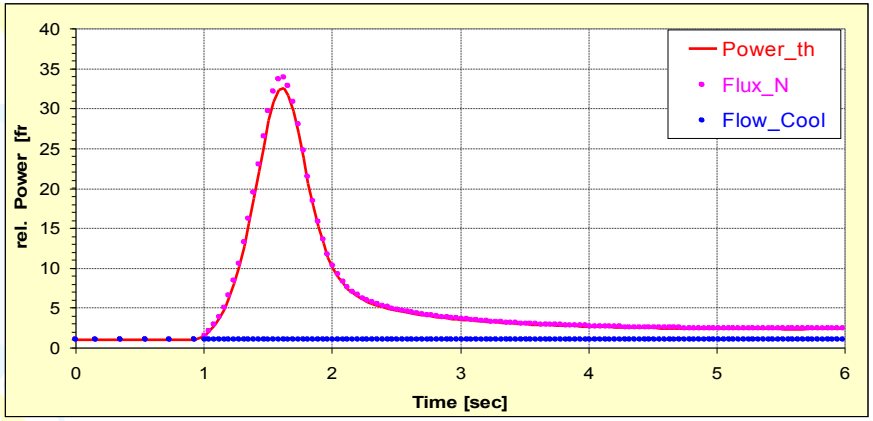


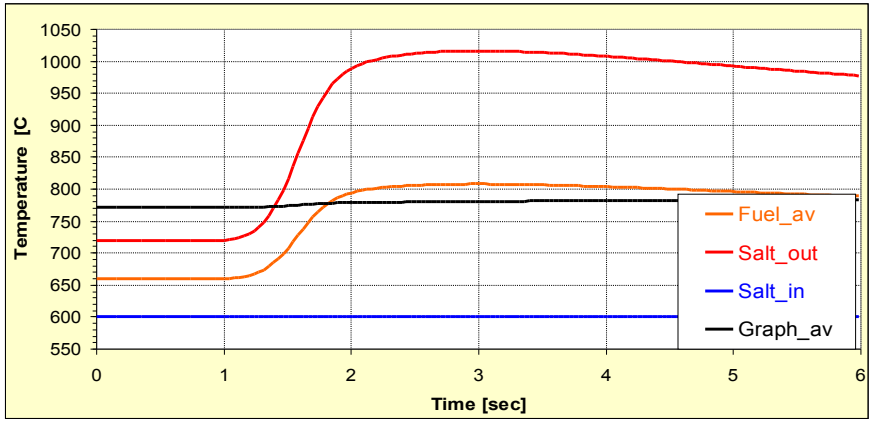


Unprotected Reactivity Insertion (UTOP) Scenario 1. Code: SIM-ADS (FZK)

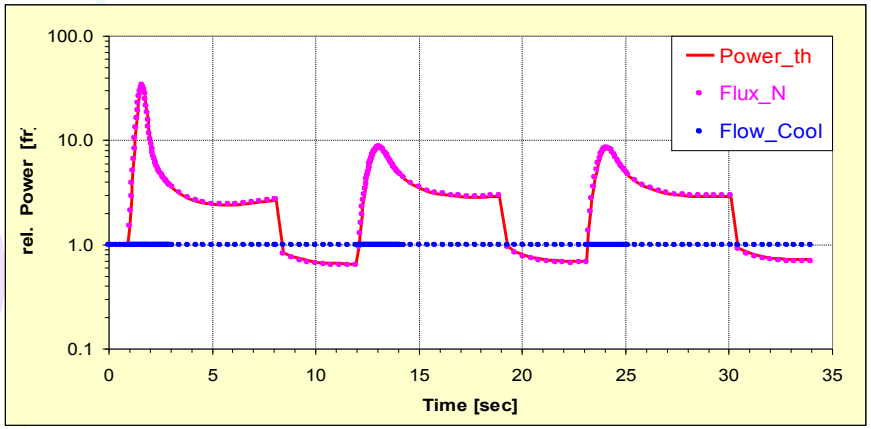
Assumed acclomoration of fuel particle : ~ + 500 pcm; this assumption is of course debateable

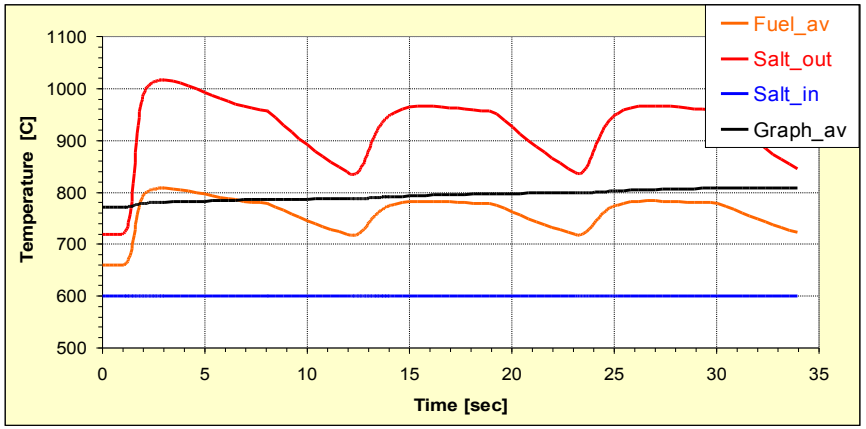
-4.125 pcm/K





-4.125 pcm/K





Meetings & Reports

- The detailed annual technical report for year 2006 is completed and now it is for approval in Rosatom. It will be circulated by the end of January 2007
- The work on the project Final report is underway now.
- In 2006 two project development meetings with EC partners from CEA, EDF, CNRS, FZK and NRI in Cadarache and Snezinsk were executed
- At last meeting in Snezinsk (November, 2006) it was decided: in order to provide opportunity to finalize some additional experiments on Te corrosion and fuel salt clean up, as well as to prepare and agree with foreign partners from EU final report "no-cost" extension of # 1606 task for 3-6 months is required
- To discuss draft of the project final report it is planned additional meeting with EC partners at RRC-KI on September, 2007
- The new proposed date to end the task will be October 31, 2007

Training in modern methods for study of An-containing MS properties

- *The objective of the training was to gain experience and practical skills in the novel experimental methods for MSR technology development*
- *Training providers: CEA, FZK, EDF, CNRS*
- *Session 1:* CEA-Cadarache, France, 2006, March, 06-10
- *Session 2 :* VNIITF, Snezinsk, RF, 2006, November, 13-17
- **Training allowed project key experts to become in detail of MSR reactor physics and fuel cycles considerations; measurement of fundamental properties of the prospect fuel and coolant salt compositions and development of the key structural materials for the fuel/coolant circuit, as well as to gain deepened practical skills**
- **The experience and practical skills in the novel experimental methods for MSR technology development received is applied to the Work plan of the Project 1606-2 and new ISTC LIST project proposal**

Papers prepared within 2006

ICAPP' 06

- Experience with Alloys Compatibility with Fuel and Coolant Salts and their Application to MOlten Salt Actinide Recycler & Transmuter, *Ignatiev V., A. Surenkov, I. Gnidoi, V. Fedulov, V. Afonichkin, A. Bovet, V. Subbotin, A. Toropov, Paper 6002*

PHYSOR 2006

- Safety-related neutronics parameters of a molten salt actinide recycler & transmuter, *A. Rineiski, V. Ignatiev, D. Da Cruz, S. Dulla, O. Feinberg, E. Malambu, W. Maschek, A. Stanculescu, M. Szieberth, S. Wang*

АТОМНАЯ ЭНЕРГИЯ 2006

- Experimental study on physical properties of molten salt fluorides as applied to molten salt actinide recycler and transmuter, *V. Ignatiev, A. Merzlyakov, V. Subbotin, A. Panov, Yu. Golovatov,*
- Experimental study on compatibility of nickel base alloys with molten salt fluorides, *V. Ignatiev, A. Surenkov, I. Gnidoi, V. Fedulov, V. Uglov A. Panov V. Subbotin, A. Toropov, V. Afonichkin, A. Bovet, v.101, #4, October, 2006, p.278-285*

Papers prepared within 2006

EUCHEM 2006

- Dynamic reference electrode for investigation of fluoride melts containing beryllium difluoride, *V. Afonichkin, A. Bovet, A. Zherebtsov, A. Panov, V. Subbotin, A. Toropov, V. Ignatiev, A. Surenkov*
- Experimental study of corrosion resistance of nickel-based alloys in fluoride melts, *A. Zherebtsov, A. Panov, V. Subbotin, A.Toropov, I. Gnidoy, V. Ignatiev, A.Surenkov, V. Uglov*
- *Na,Li,Be/F Melt Preparation for Experiments in Thermal Convection Loop, A. Panov, V. Subbotin, A. Toropov, V. Afonichkin, V. Ignatyev , A. Surenkov*

9th OECD/NEA IEM on P&T

- Progress in Integrated Study of MOlten Salt Actinide Recycler & Transmuter System, *V. Ignatiev, V. Afonichkin, O. Feynberg, A. Lopatkin, A. Merzlyakov, A. Myasnikov, A. Panov, V. Smirnov, V. Subbotin, A. Surenkov, A. Toropov, I. Tretiakov, G. Vanukova, R. Zakirov, D. Da Cruz, S. Dulla, E. Malambu, W. Maschek, A. Rineiski, M. Schikorr, A. Stanculescu, S. Wang*

ICONE 14

- TRANSIENT ANALYSES FOR A MOLTEN SALT TRANSMUTATION REACTOR USING THE EXTENDED SIMMER-III CODE, *S. Wang, A. Rineiski, W. Maschek, V. Ignatiev*

CONCLUSIONS

- It is important that, for molten Li,Be,Na/F system, was found quite wide range with minimal of LiF (17-15 mole%) and of BeF2 (27-25mole%) content in the ternary composition, which provide fuel salt able to get PuF3 solubility of 2 and 3 mole%, respectively, at 600C; to keep adequate melting point (<500°C) and very low vapour pressure; to have good nuclear properties, low activation, suitable transport properties; to be well compatible with the materials in the system and moderately expensive (about 25$ per kg).

- MOSART system has homogeneous core with intermediate - to- fast spectrum of neutrons. The fuel salt specific power is about 43 W/cm3. The effective flux of such system is near 1x1015 n $cm^{-2}s^{-1}$. At equilibrium state for both scenarios of start up and feed plutonium and minor actinides compositions and the soluble fission product removal cycle 300 epdf for cores with different reflector parameters considered, the AnF3+LnF3 concentration in fuel salt is truly within the solubility limit of AnF3+LnF3 for molten 15LiF-27BeF2-58NaF (mole %) at minimum fuel salt temperature in primary circuit of 600C. Minimal AnF3+LnF3 critical concentrations at equilibrium were received for core with in 0.2m graphite reflector.

- MOSART core of homogeneous configuration can satisfy most important neutronic and thermal-hydraulic considerations: (1) the AnF3+LnF3 concentration in fuel salt is truly within the solubility limit of AnF3+LnF3 for molten 15LiF-27BeF2-58NaF (mole %) at minimum fuel salt temperature in primary circuit of 600C for both fuel cycle scenarios under consideration; (2) core with 0.2m graphite reflector for T= 900 – 1600K has strong negative temperature reactivity coefficients (– 4.125 pcm/K and – 6.625 pcm/K for scenarios 1 and 2 of equilibrium critical loading, respectively); (3) regions of reverse, stagnant or laminar flow are avoided and (4) the maximum temperature of solid reflectors is low enough to allow it use for suitable time.

CONCLUSIONS

- Transient study has demonstrated that the MOSART design is an inherently stable reactor design on account of its large, negative fuel temperature coefficient (-4.125 pcm/K) in combination with its negative graphite reflector reactivity coefficient (-0.04 pcm/K). The fast spectrum MOSART system is relatively fast in its thermal response since it does not contain a graphite matrix inside the core region as the thermal-spectrum MSBR design (aside of the graphite reflector). The MOSART reactor is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP.
- Preliminary consideration of environment effects indicate that MOSART system could have attracted performance and TRU transmutation efficiency features while providing lower total materials inventories and waste compared to prior MSR designs, including MSBR (e.g., it allows significantly reduce to the order mass flows of graphite and 7Li enriched of 99.99 % in the design).
- While a substantial R&D effort would be required to commercialize MOSART, there is no killing unresolved issues in the needed technology. The major technical uncertainties in the conceptual design are in the area of tritium confinement, fuel salt processing and behavior of some fission products.