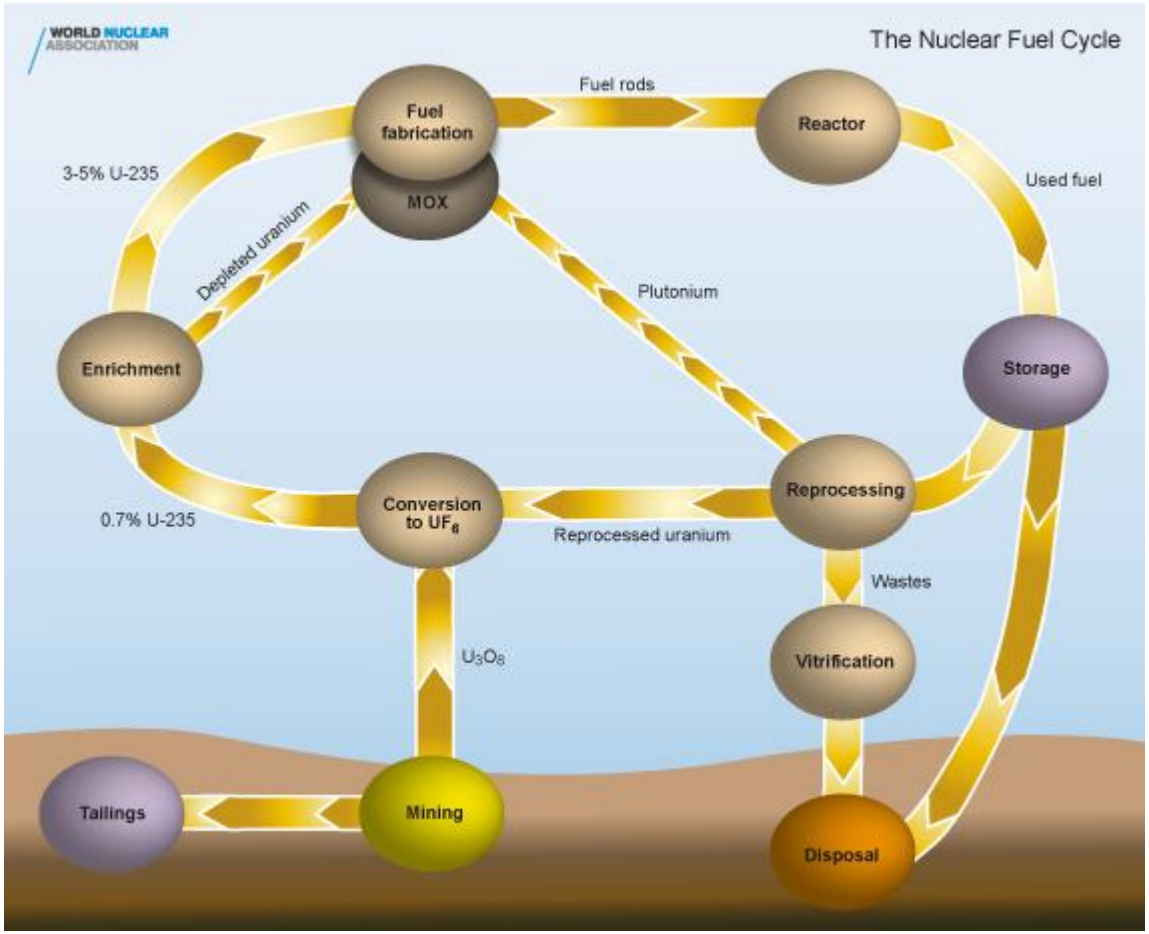
Dubieńskie badania możliwości wypalania odpadów jądrowych

Podsumowanie prac za ostatnie 3 lata

Spis tematów, którymi się zajmuję

1. Wstęp – cykl paliwowy
2. Pomiary wypalania aktynowców na zestawie KWINTA w ZIBJ Dubna
3. Badania pola neutronów na zestawie KWINTA przy pomocy Y-89
4. Dynamiczne modelowanie reaktora na stopionych solach (MSR) i ekstraktora aktynowców ze stopionych soli do stopionego bizmutu

Cykl paliwowy reaktora jądrowego



http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Introduction/Nuclear-Fuel-Cycle-Overview/

Material balance in the nuclear fuel cycle

Mining	Anything from 20,000 to 400,000 tonnes of uranium ore
Milling	230 tonnes of uranium oxide concentrate (which contains 195 tonnes of uranium)
Conversion	288 tonnes uranium hexafluoride, UF6 (with 195 tU)
Enrichment	35 tonnes enriched UF6 (containing 24 t enriched U) – balance is 'tails'
Fuel fabrication	27 tonnes UO2 (with 24 t enriched U)
Reactor operation	8760 million kWh (8.76 TWh) of electricity at full output, hence 22.3 tonnes of natural U per TWh
Used fuel	27 tonnes containing 240 kg transuranics (mainly plutonium), 23 t uranium (0.8% U-235), 1100 kg fission products.

The above figure may be regarded as typical for the annual operation of a 1000 MWe nuclear power reactor such as many operating today

Ponad 400 reaktorów na świecie * 27 ton /reaktor/rok ≈ 10 000 t/rok
= $1x10^{10}$g/rok/19.1g/cm^3 ≈ $5x10^8$cm^3 /rok≈ 500 m^3/rok

http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Introduction/Nuclear-Fuel-Cycle-Overview/

Co to są odpady jądrowe

Used fuel from light water reactors (at normal US burn-up levels) contains approximately:

•95.6% uranium, over 98.5% of which is U-238 (the remainder consists of: trace amounts of U-232 and U-233; less than 0.02% U-234; 0.5-1.0% U-235; around 0.5% U-236; and around 0.001% U-237 – which accounts for nearly all of the activity)

•2.9% stable fission products

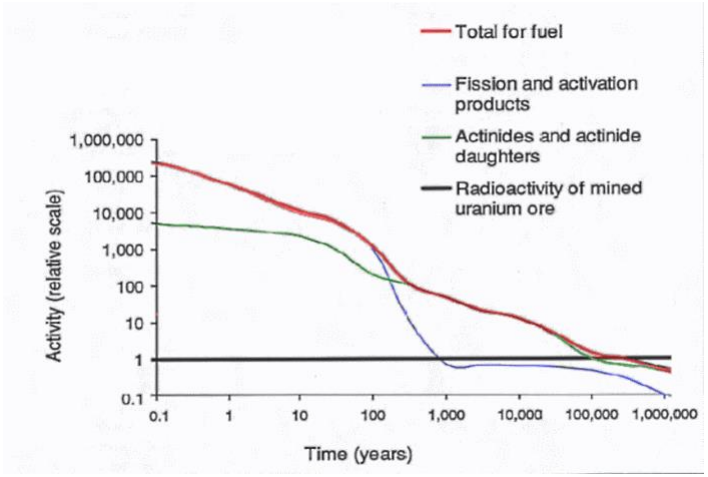
•0.9% plutonium

•0.3% caesium & strontium (fission products)

•0.1% iodine and technetium (fission products)

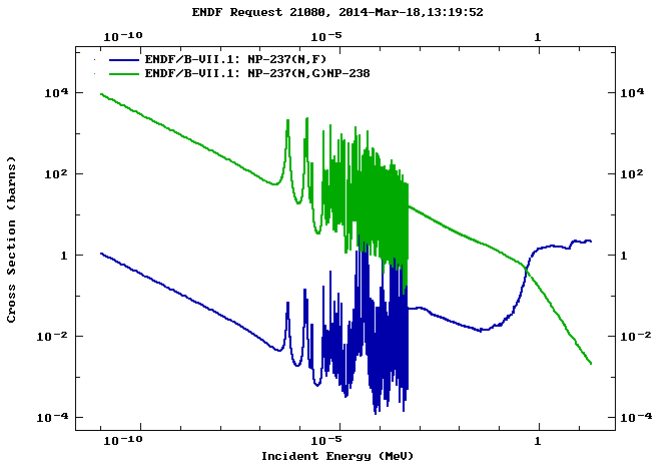
•0.1% other long-lived fission products

•0.1% minor actinides (americium, curium, neptunium)



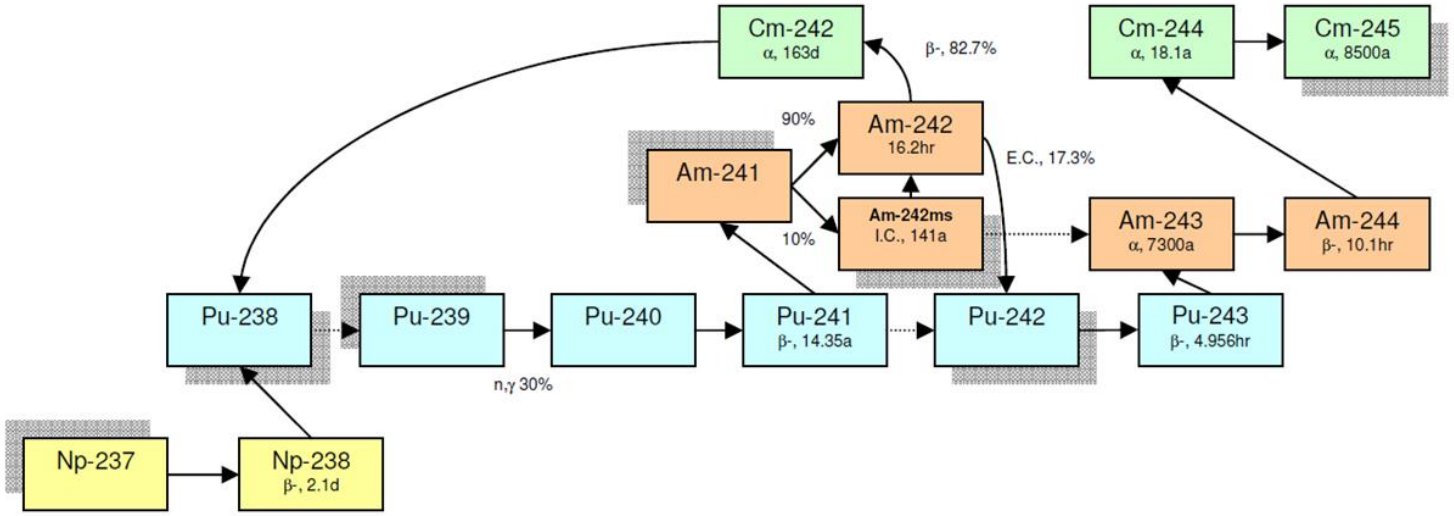
http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Nuclear-Wastes/Radioactive-Waste-Management/

What's the problem with actinides



Np-237 fission and neutron capture CS dependence on energy

and actinides accumulation chains



Samuel E. Bays; Reactor Physics Characterization of Transmutation Targeting Options in a Sodium Fast Reactor; INL/CON-07-12439

Neutron capture produces another actinide. Np-237 fission is in fact the only way to get rid of its long lived activity. High energy neutrons needed to make fission prevail over capture.

Transmutation probabilities (%)

Isotope	$T_{1/2}$	thermal spectrum	fast spectrum
Np-237	$2.144*10^6$y	3	27
Pu-238	87.7y	7	70
Pu-239	24110y	63	85
Pu-240	6563y	1	55
Pu-241	14.35y	75	87
Pu-242	$3.733*10^5$y	1	53
Am-241	432.2y	1	21
Am-242m	141y	75	94
Am-243	7370y	1	23
Cm-242	162.8d	1	10
Cm-243	29.1y	78	94
Cm-244	18.1y	4	33

http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Fuel-Recycling/Processing-of-Used-Nuclear-Fuel/

Metody pozbywania się odpadów jądrowych

- Przechowywać - tak, ale nie wykorzystuje się ponad 95% uranu
- Przestawić energetykę jądrową z dotychczasowych reaktorów termicznych na reaktory prędkie. Generation IV International Forum (GIF) wytypowało 6 typów reaktorów spełniających wymagania pod względem bezpieczeństwa, ekonomii i możliwości wypalania aktynowców.

Generation IV International Forum (GIF) reactors

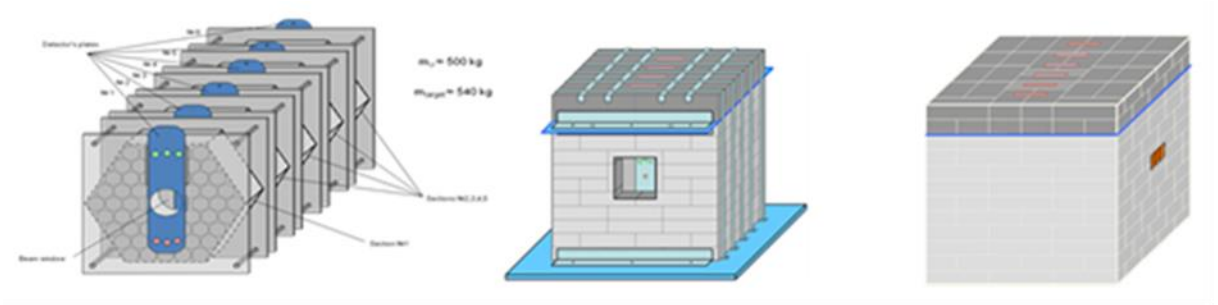
	neutron spectrum (fast/ thermal)	coolant	temperature (°C)	pressure*	fuel	fuel cycle	size(s) (MWe)	uses
Gas-cooled fast reactors	fast	helium	850	high	U-238 +	closed, on site	1200	electricity & hydrogen
Lead-cooled fast reactors	fast	lead or Pb-Bi	480-570	low	U-238 +	closed, regional	20-180** 300-1200 600-1000	electricity & hydrogen
Molten salt fast reactors	fast	fluoride salts	700-800	low	UF in salt	closed	1000	electricity & hydrogen
Molten salt reactor - Advanced High-temperature reactors	thermal	fluoride salts	750-1000		UO_2 particles in prism	open	1000-1500	hydrogen
Sodium-cooled fast reactors	fast	sodium	500-550	low	U-238 & MOX	closed	50-150 600-1500	electricity
Supercritical water-cooled reactors	thermal or fast	water	510-625	very high	UO_2	open (thermal) closed (fast)	300-700 1000-1500	electricity
Very high temperature gas reactors	thermal	helium	900-1000	high	UO_2 prism or pebbles	open	250-300	hydrogen & electricity

* high = 7-15 MPa
+ = with some U-235 or Pu-239
** 'battery' model with long cassette core life (15-20 yr) or replaceable reactor module.

http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Power-Reactors/Generation-IV-Nuclear-Reactors/

Eksperymenty w Dubnej na głęboko podkrytycznym zestawie KWINTA

Prace w ZIBJ Dubna są realizowane w ramach protokołu Nr. 4382-1-14/16 o współpracy. Badania te są całkowicie finansowane z grantów dubieńskich przyznawanych rokrocznie od wielu lat – koszty wyjazdów i prac badawczych.

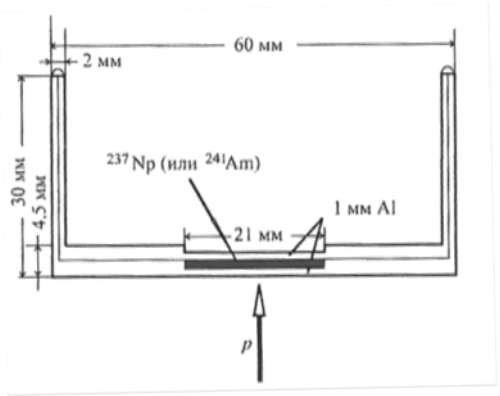


Zestaw KWINTA symuluje konfigurację reaktora prędkiego sterowanego akceleratorem (ADS) – materiał – uran naturalny

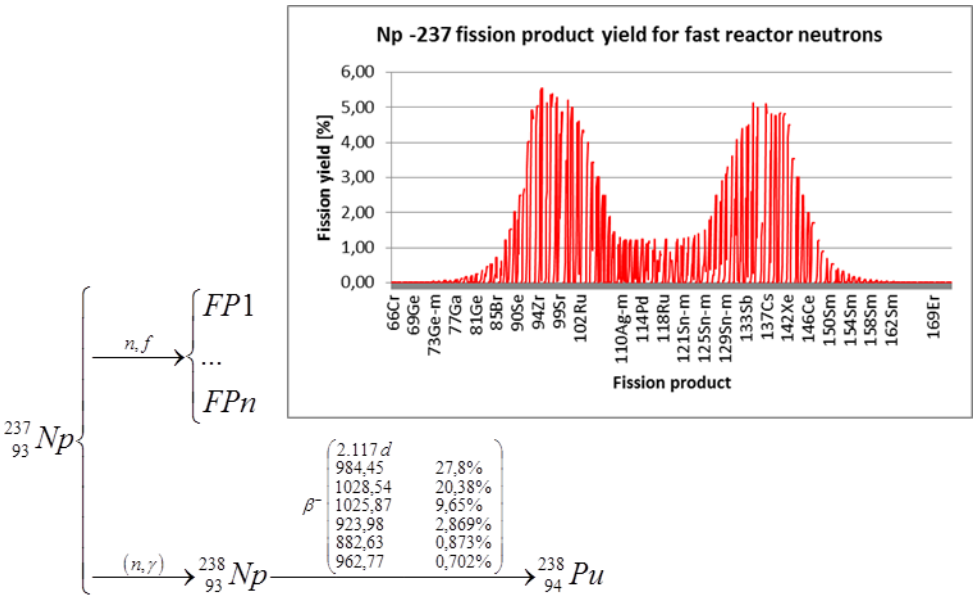
Cel eksperymentów – określić efektywność wypalania aktynowców mniejszych - Am-241 i Np-237

Metoda określania efektywności wypalania aktynowców

- Naświetlić próbkę aktynowca
- Zarejestrować widmo gamma próbki
- Wyznaczyć parametry pików i zidentykować ich źródło – które z produktów rozszczepienia a które z wychwytu neutronów
- Wyznaczyć liczbę rozszczepień i liczbę wychwytów



Schemat ogólny procesów w próbce Np-237 w trakcie naświetlania



Basic gamma lines identified in Np-237 sample

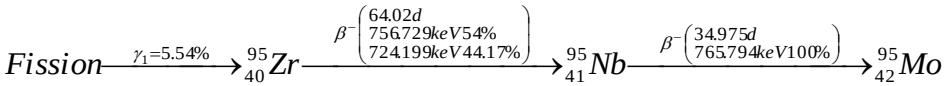
E-gamma	Isotope	Source	T1/2	Fission yield [%] [4]	I-gamma [%] [3]
529.87	133I	FP	20.87h	4,45	87
657.94	Zr-97->97Nb*	FP	16.744h	5,38	98,23
667.71	Te-132->I-132**	FP	3.26d	4,39	98,7
743.36	Zr-97	FP	16.744h	5,35	93,6
772.6	Te-132->I-132**	FP	3.26d	4,39	75,6
1131.51	I-135	FP	6.57h	4,16	22,6
1260.41	I-135	FP	6.57h	4,16	28,7
923.98	Np-238	CP	2.117d	N/A	2,869
962.77	Np-238	CP	2.117d	N/A	0,702
984.45	Np-238	CP	2.117d	N/A	27,8
1025.87	Np-238	CP	2.117d	N/A	9,65
1028.54	Np-238	CP	2.117d	N/A	20,38

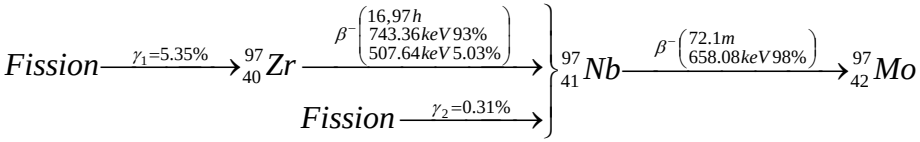
FP – fission product. CP – neutron capture product.

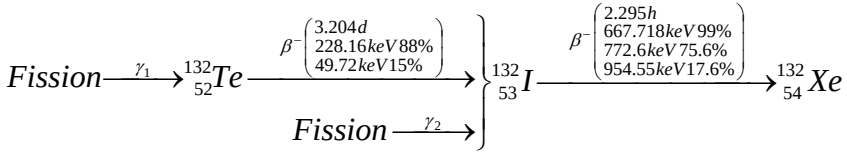
*Line 657.94 keV stems in fact from Nb-97 beta decay ($T_{1/2}$ = 72.1 min), but its quantity is modified by Zr-97 decay rate ($T_{1/2}$ = 16.744h) [3,4]. Therefore Zr-97 decay constant (16.744h) approximates the line 657.94 activity decreasing.

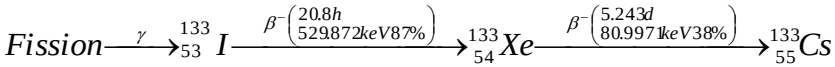
**Lines 667.71 and 772.6 keV stem from I-132 ($T_{1/2}$ = 2.295h) but their activities are modified by Te-132 decay rate ($T_{1/2}$ = 3.26d) [3,4]. Therefore Te-132 decay constant (3.26d) approximates the lines activity decreasing.

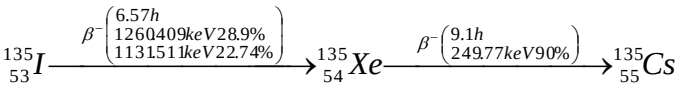
Processes in Np-237 sample during irradiation – decay chains











Wzory na wyznaczanie liczby reakcji z danych doświadczalnych

$$I_{f1g} = \frac{A_{1g}}{m \cdot \phi \cdot \gamma \cdot \varepsilon_{pg} \cdot I_g} \cdot \frac{\lambda_1 \cdot t_{irr}}{\left(1 - e^{-\lambda_1 t_{irr}}\right)} \cdot \frac{e^{\lambda_1 t_+}}{\left(1 - e^{-\lambda_1 t_{real}}\right)} \cdot \frac{t_{real}}{t_{live}}$$

$$I_{f2g} = \frac{A_{2g}}{m \phi \varepsilon_g I_g} \frac{t_{real}}{t_{live}} \frac{t_{irr}}{\left[c_2 \left(1 - e^{-\lambda_2 t_{real}}\right) e^{-\lambda_2 t_+} + c_1 \left(1 - e^{-\lambda_1 t_{real}}\right) e^{-\lambda_1 t_+}\right]}$$

$$c_2 = \frac{\gamma_2}{\lambda_2}\left(1 - e^{-\lambda_2 t_{irr}}\right)$$

$$c_1 = \frac{\gamma_1}{\lambda_1}\left(1 - e^{-\lambda_1 t_{irr}}\right)$$

$I_{f2g} = \frac{N_A \sigma_f}{A}$ - fission rate (number of fissions per gram of sample) based on activity of the 2-nd isotope in decay chain

A_{1g}, A_{2g} - gamma peak area

γ_1, γ_2 - Fission product 1 and 2 yield, [%]

λ_1, λ_2 - fission product decay constant, [s^{-1}]

t_{irr}, t_+, t_{real} - irradiation, cooling and measurement time, [s]

m - sample mass, [g]

ϕ - integral number of beam protons

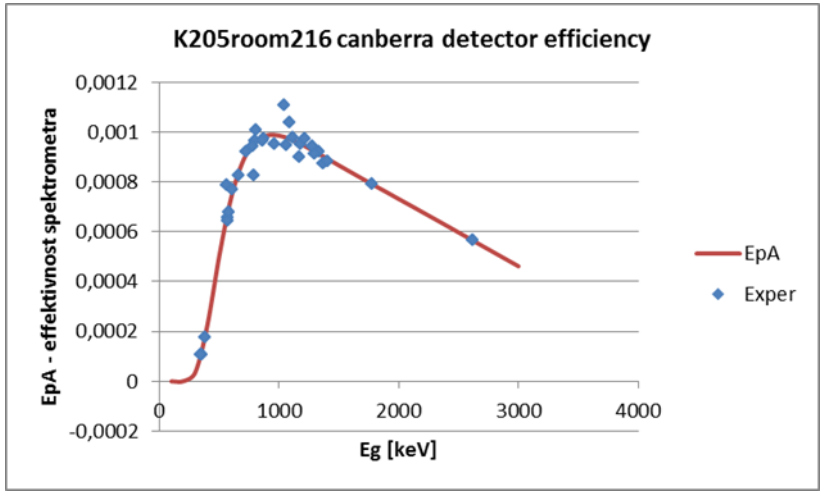
ε_{pg} - detector efficiency.

I_g - gamma line intensity, [%]

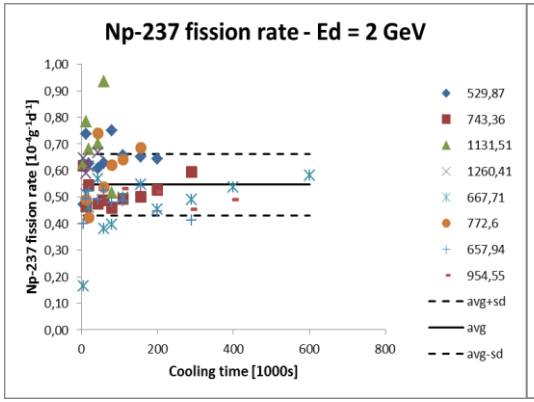
Wyznaczanie efektywności detektora germanowego
Źródła kalibracyjne, formuła i krzywa wydajności

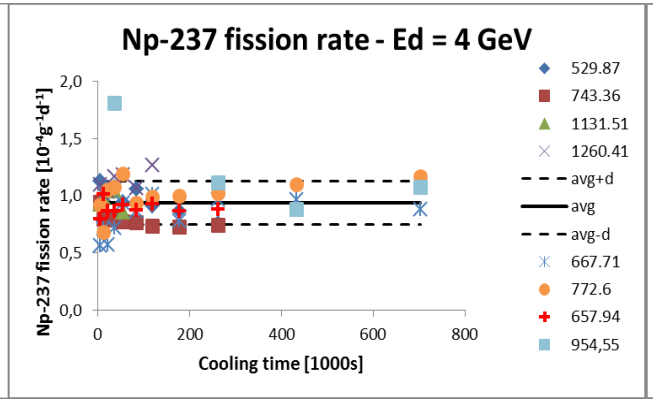
Nuklide	Aktivity [kBq]	Error[%]	T1/2	Eg [keV]	Ig
Mn54	116	3	312.3d	834.848	99.976
Co57	123	3	271.79d	122.0614	85.60
				136.4743	10.68
Co60	107	3	5.2714y	1173.237	99.9736
				1332.501	99.9856
Y88	210	3	106.65d	898.042	93.7
				1836.063	99.2
Sn113	225	3	115.09d	391.690	64
Ba133	54.6	3	10.51y	53.161	2.199
				80.9971	34.06
				223.234	0.450
				276.398	7.164
				302.853	18.33
				356.017	62.05
				383.851	8.94
Cs137	114	3	30.07y	661.657	85.1
Ce139	153	3	137.640d	165.864	80
Eu152	54.3	3	13.537y	121.7817	28.58
				244.6975	7.583
				411.1163	2.234
				443.965	2.821
				778.9040	12.942
				964.079	14.605
				1112.074	13.644
				1408.006	21.005
Th228	40.5	3	1.931y	84.373	1.22
				238.632	43.3
				240.986	4.10
				277.371	6.1
				300.087	3.28
				510.77	22.6
				583.187	84.5
				727.33	6.58
				785.37	1.102
				860.557	12.42
				1620.5	1.49
				2614.511	99
Am241	**110**	**3**	**432.2y**	**59.5412**	35.9

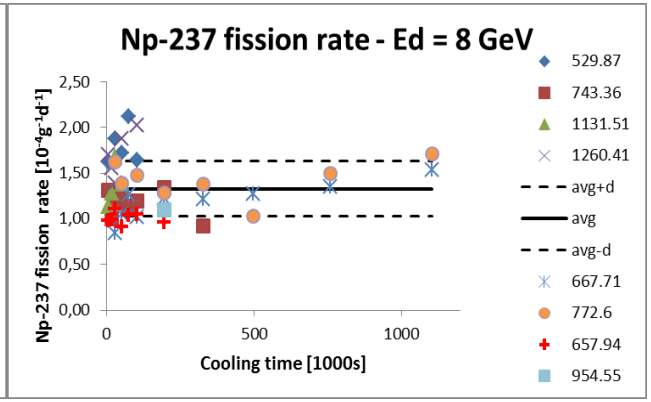
$$\varepsilon_{pA}(E) = \frac{S_E}{B \cdot t_{live}} \cdot \frac{100}{I_\gamma} \cdot e^{\lambda t_+}$$

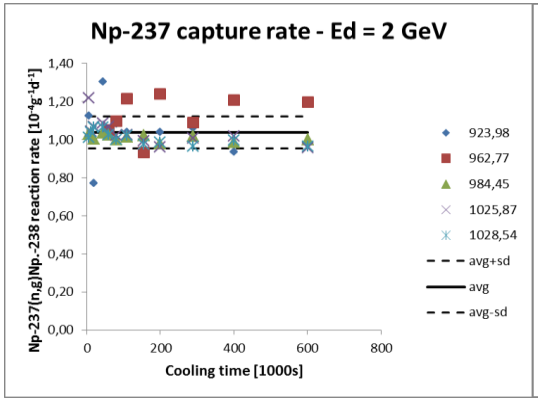


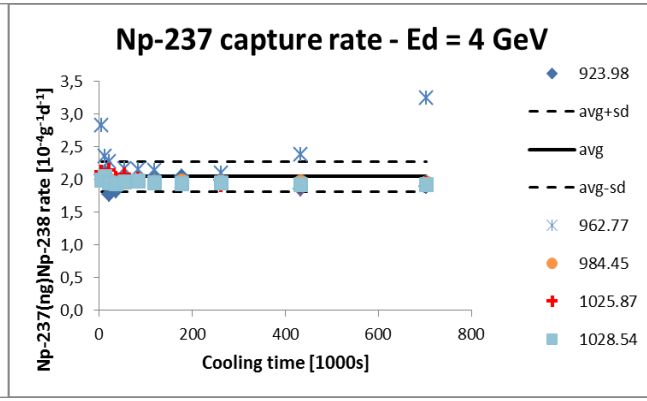
Np-237 fission rate and capture rate example results

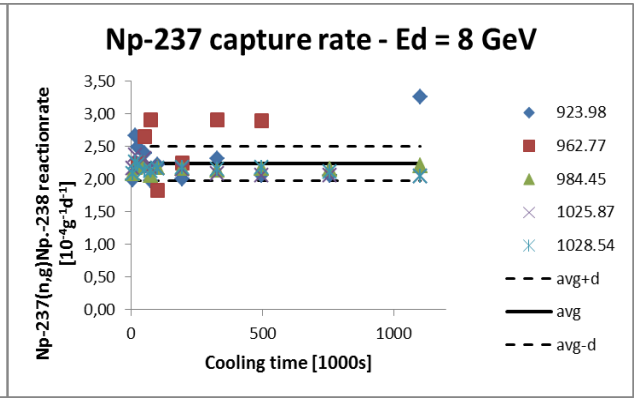




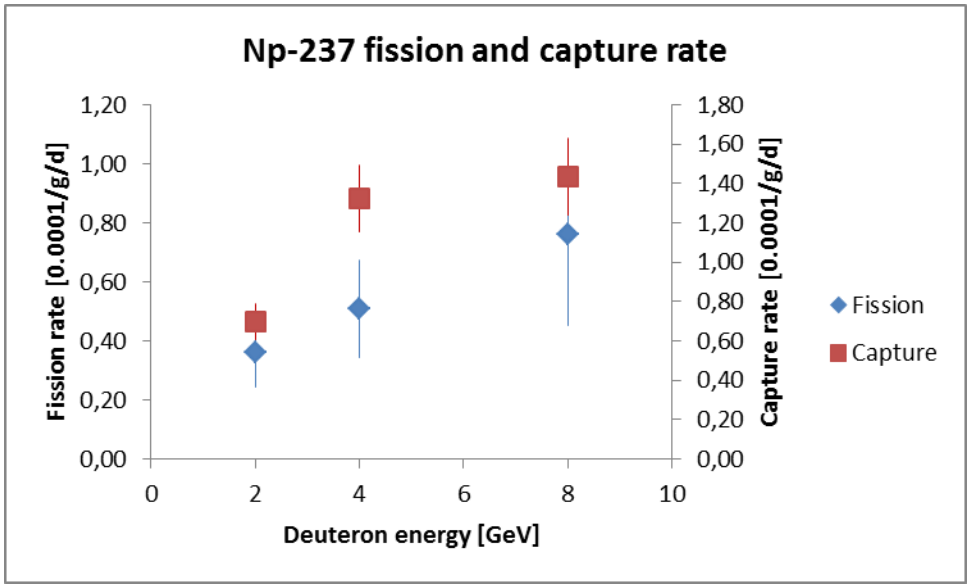






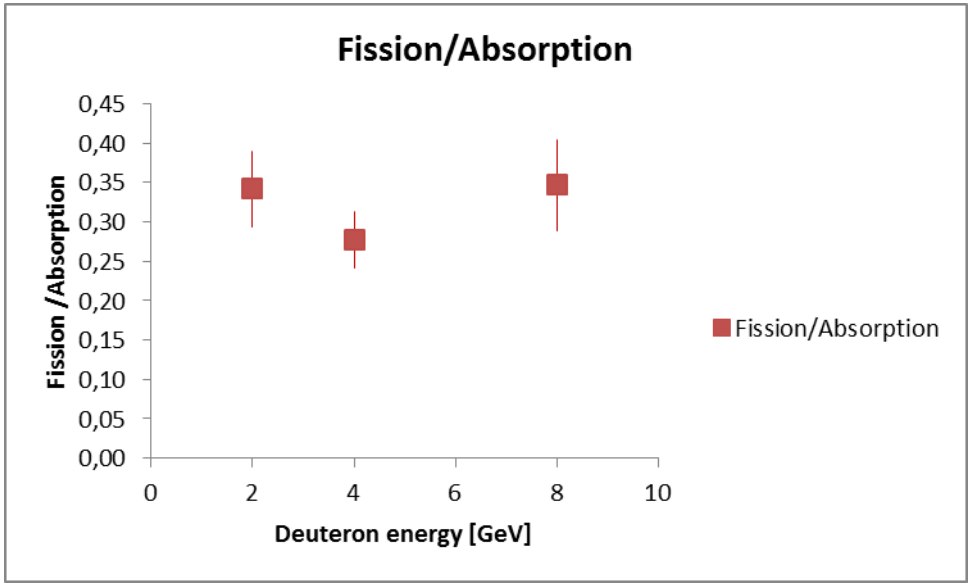


Np-237 fission and capture dependence on deuteron energy



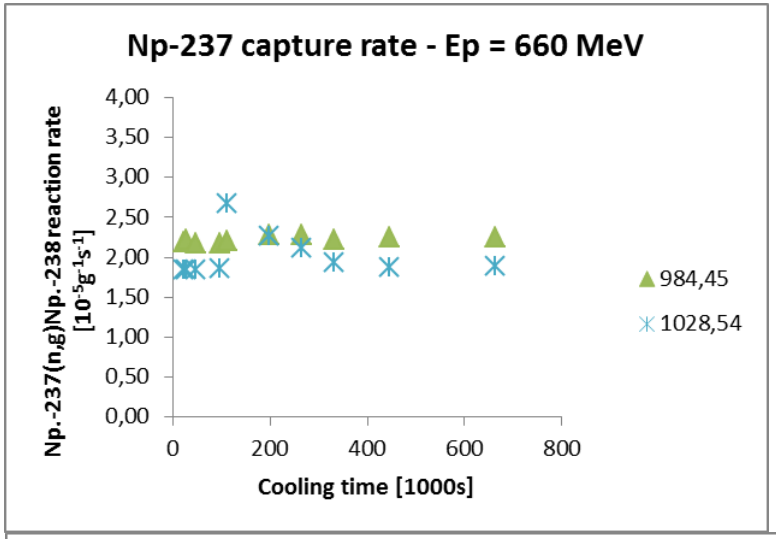
Beam deuteron energy [GeV]	Fission rate [$10^{-4}g^{-1}d^{-1}$]	Standard deviation [$10^{-4}g^{-1}d^{-1}$]	Standard deviation [%]	Capture rate [$10^{-4}g^{-1}d^{-1}$]	Standard deviation [$10^{-4}g^{-1}d^{-1}$]	Standard deviation [%]
2	0.363	0.12	33.04	0.699	0.0887	12.69
4	0.509	0.164	32.26	1.33	0.171	12.88
8	0.76	0.306	40.3	1.43	0.196	13.67

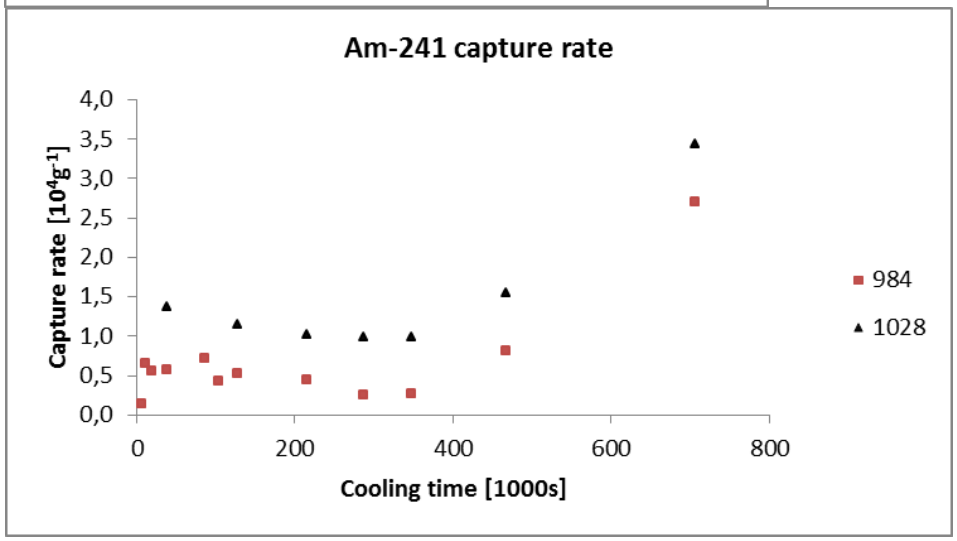
Np-237 fission/absorption ratio dependence on deuteron energy



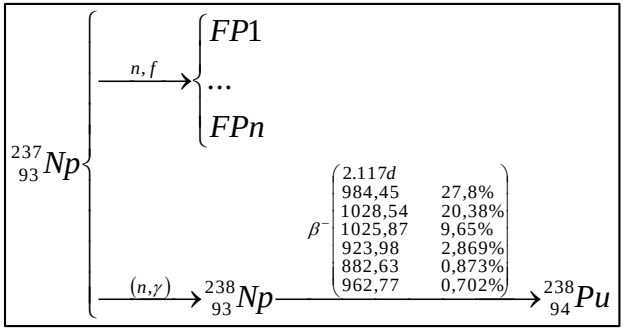
Beam deuteron energy [GeV]	Fission to absorption ratio $I_f/(I_f+I_c)$	Error	%Error
2	0.34	0.05	14.05%
4	0.28	0.04	12.91%
8	0.35	0.06	16.58%

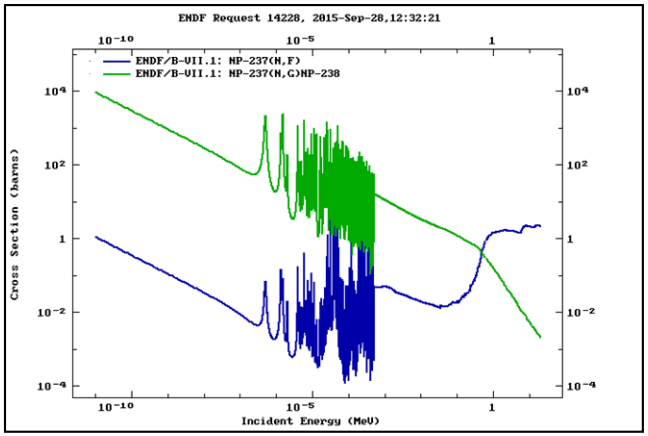
Eksperyment 2014-11 - Różnice między Am-241 i Np -237

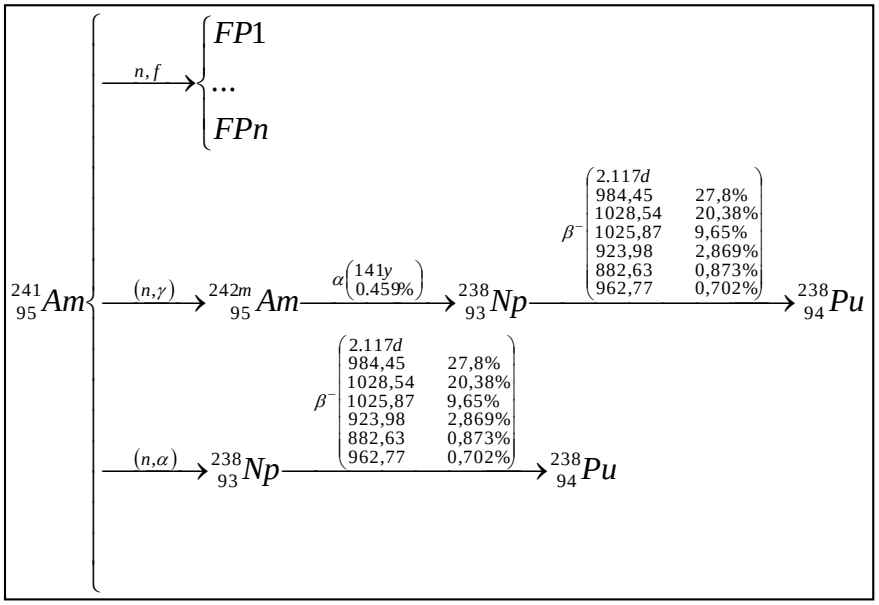


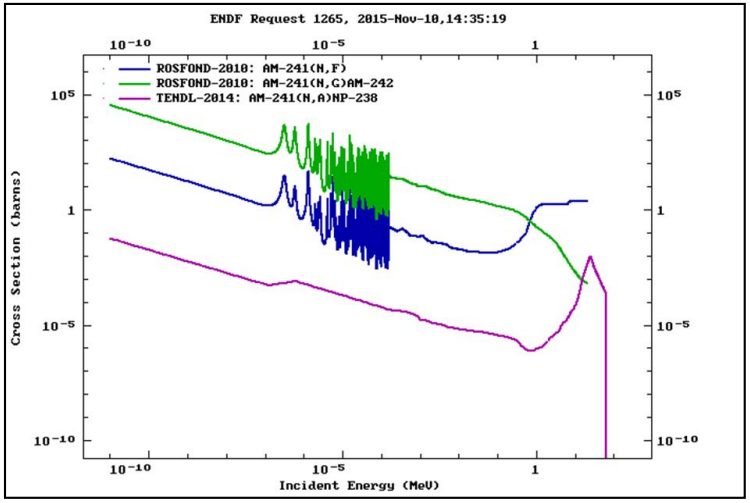


Różnice między Am-241 i Np -237 - cd









Proposed method of actinide sample application as neutron fluency detector

- Having the average fission and capture cross section values (spectral index) we can evaluate the average neutron flux in the location of the actinide sample using the measured amount of fissioned and captured actinide isotopes.
- Two different equations for fissioned and captured actinide isotopes should give the same average neutron flux value what is a proof (check, test) for correct measurement.
- The number of selected fission products or neutron captured actinide isotopes (N_y) in the actinide sample of volume V_p can be expressed:
- $$N_y = V_p \, \bar{\phi} \, N \, \bar{\sigma} \, t$$

  where

  $\bar{\Phi}$ - average neutron flux in the place of actinide sample location [n/cm^2·s]

  N – number of actinide nuclei in volume unit [cm^{-3}]

  $\bar{\sigma}$ -average microscopic cross section for the reactions (n, f) or (n, γ) [barns],

  t - irradiation time.

Stosunek liczby rozszczepień do wychwytów a średnia energia neutronów

Neutron energy [MeV]	Fission cross section σ(n,f) [barn]	Capture cross section σ(n,γ) [barn]	σ(n,f)/ σ(n,γ)
0.3	0.073	0.681	0.107
0.37	0.142	0.594	0.239
0.38	0.152	0.579	0.262
0.39	0.169	0.564	0.299
0.4	0.183	0.548	0.333
0.41	0.211	0.533	0.395
0.42	0.239	0.519	0.461
0.43	0.268	0.504	0.531
0.44	0.297	0.491	0.605
0.45	0.325	0.478	0.679
0.46	0.348	0.464	0.750
0.50	0.447	0,411	1.087

Compilation of incineration rate and capture rate of Np -237 results.

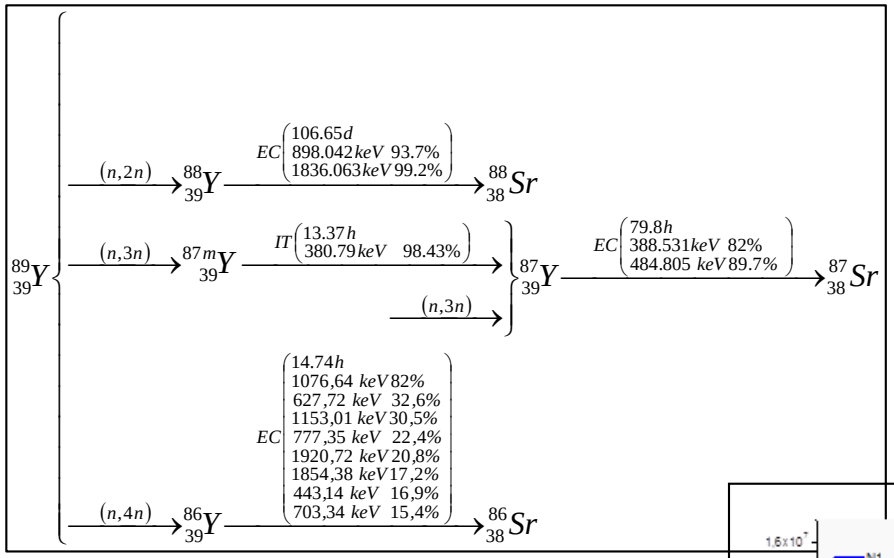
Deuteron beam energy	Ed = 2 GeV	Ed = 4 GeV	Ed = 8 GeV
Deuteron fluency	$3.02(10)\times 10^{13}$	$2.73(10) \times 10^{13}$	$0.91(4) \times 10^{13}$
Mass of Np.-237 sample [g]	0.987	1.115	1.115
Number of Np-237 fissions per gram sample	$(1.12\pm0.393)\times10^{9}$	$(1.69\pm0.698)\times10^{9}$	$(1,03\pm0.543) \times10^{9}$
Number of neutron captured Np-237 atoms per gram sample	$(2.13\pm0.\ 271) \times10^{9}$	$(4.73\pm0.609) \times10^{9}$	$(1.99\pm0.272) \times10^{9}$
Fission rate per gram sample and deuteron	$(3,72\pm1,3)\times10^{-5}$	$(6.20\pm2,56)\times10^{-5}$	$(1,13\pm0.596) \times10^{-4}$
Capture rate per gram sample and deuteron.	$(7.064\pm0.897) \times10^{-5}$	$(1.73\pm0.223 \times10^{-4}$	$(2.19\pm0.299) \times10^{-4}$
Fission/Capture rate	0.53 ± 0.195	0.36 ± 0.154	0.52 ± 0.281
Fission rate per gram sample, deuteron and its energy	$(1,86 \pm 0.650) \times10^{-5}$	$(1.55 \pm 0.639) \times10^{-5}$	$1.41 \pm 0.745) \times10^{-5}$
Capture rate per gram sample, deuteron and its energy	$3.53 \pm 0.448) \times10^{-5}$	$4.33 \pm 0.558) \times10^{-5}$	$2.73 \pm 0.374) \times10^{-5}$

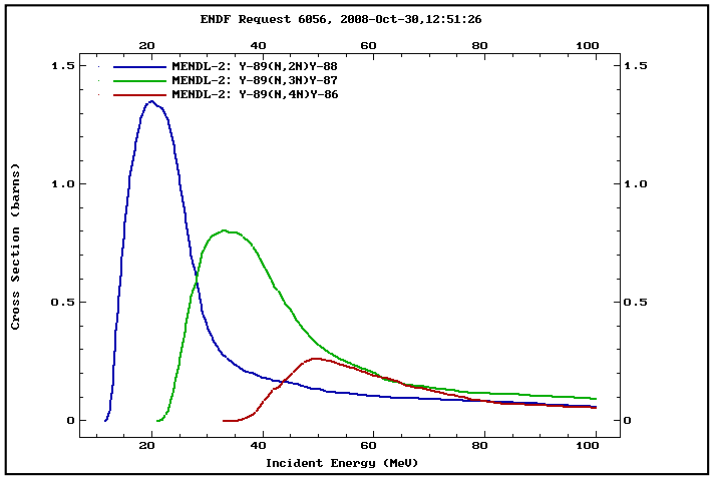
Compilation of incineration rate and capture rate of Np -237 results – cont.

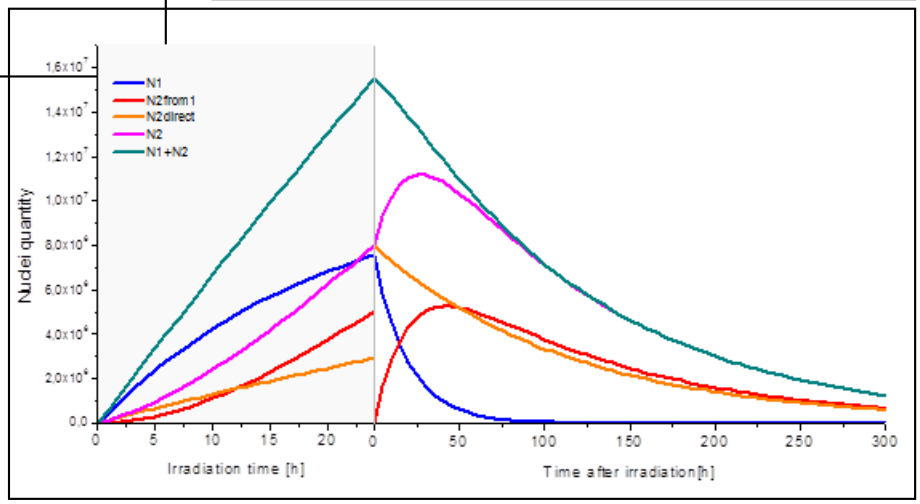
Table 6. Compilation of average neutron flux and neutron fluency in the location of the actinide neptunium 237 in the QUINTA assembly using for evaluation the measured number of fissions and captures in the samples for three experiments

Deuteron beam energy	2 GeV	4 GeV	8 GeV
Average neutron flux using for evaluation the number of fissions in the sample [n/cm^2·s] x10^7	7.29	10.08	5.21
Average neutron flux using for evaluation the number of captures in the sample [n/cm^2·s] x10^7	7.37	10.09	2.58
Neutron fluency using for evaluation the number of fissions in the sample [n/cm^2] x10^{12}	1.64	3.63	3.13
Neutron fluency using for evaluation the number of captures in the sample [n/cm^2] x10^{12}	1.66	3.39	1.55
Average neutron energy [MeV]	**0.43**	**0.4**	**0.43**

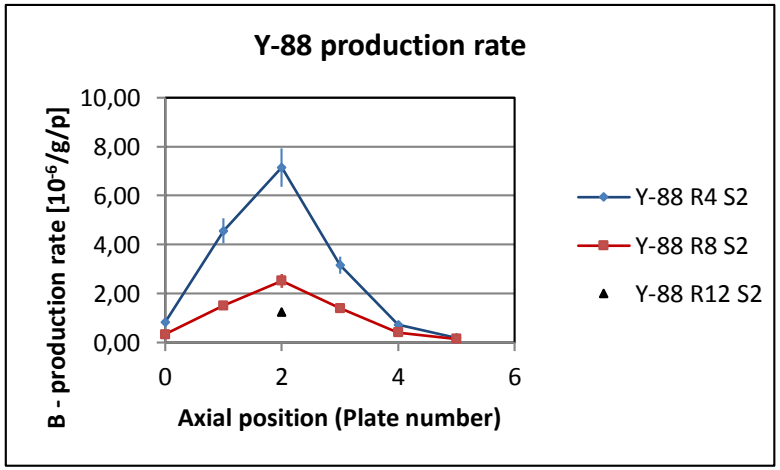
Pomiary przy pomocy Y-89
- kanały reakcji (n,xn) i odpowiadające im linie gamma
- przekroje reakcji (n,xn)
- symulacja przebiegu reakcji Y-89(n,3n)Y-87

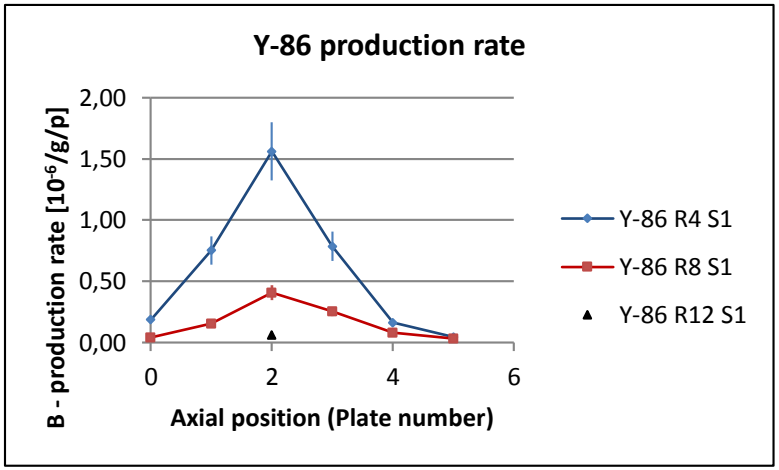


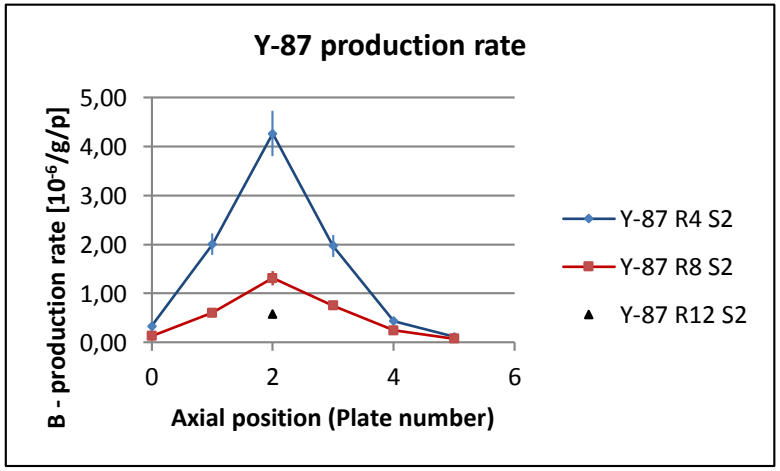




Przykładowe wyniki - eksperyment 2014-11
Osiowe rozkłady szybkości reakcji Y-89(n,xn)







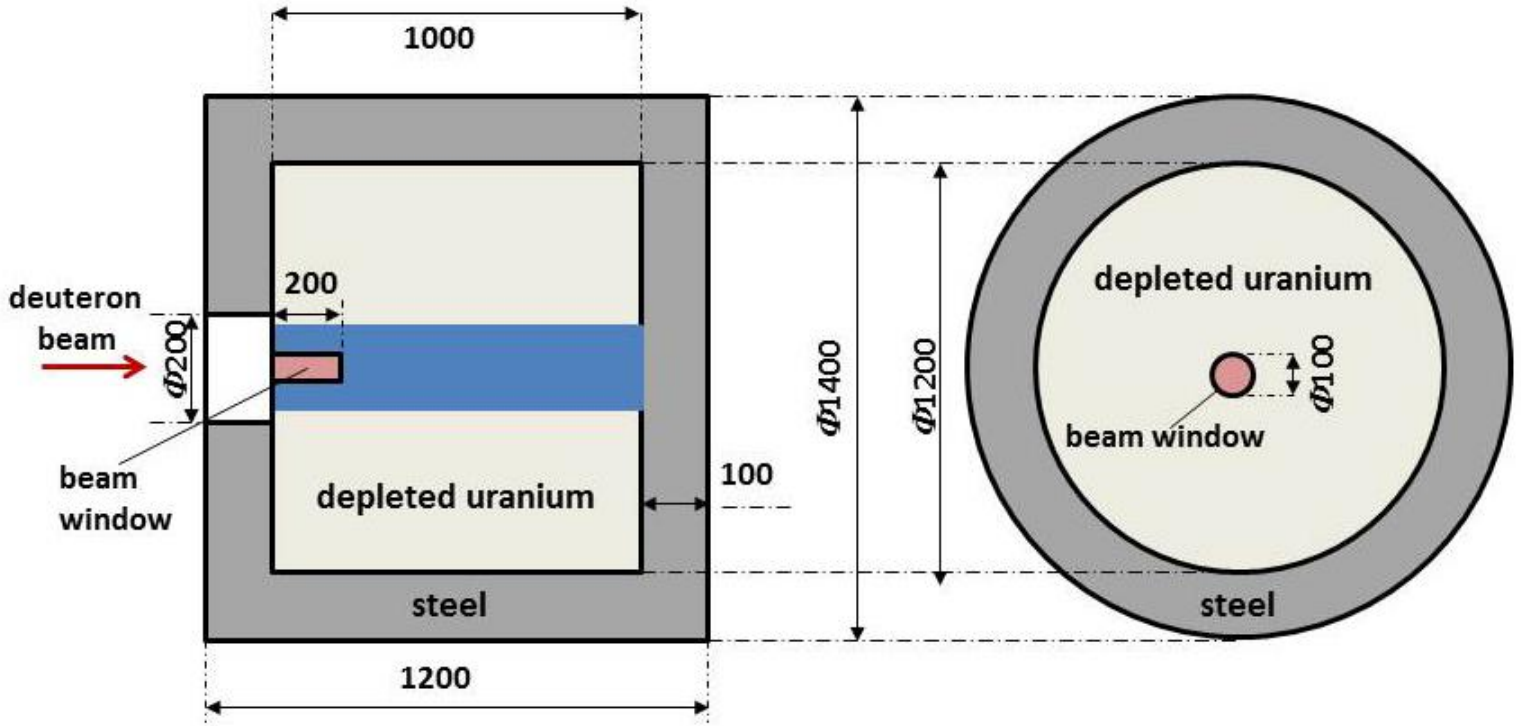
Co zrobiono ostatnio (2015-12)

Udział w eksperymencie 2015-12-04 na zestawie KWINTA w wiązce protonów 660 MeV z FAZOTRONU

- Naświetlono próbki Np-237, Am-241, Pu-239, po jednej każdego rodzaju
- Naświetlono próbki Y-89

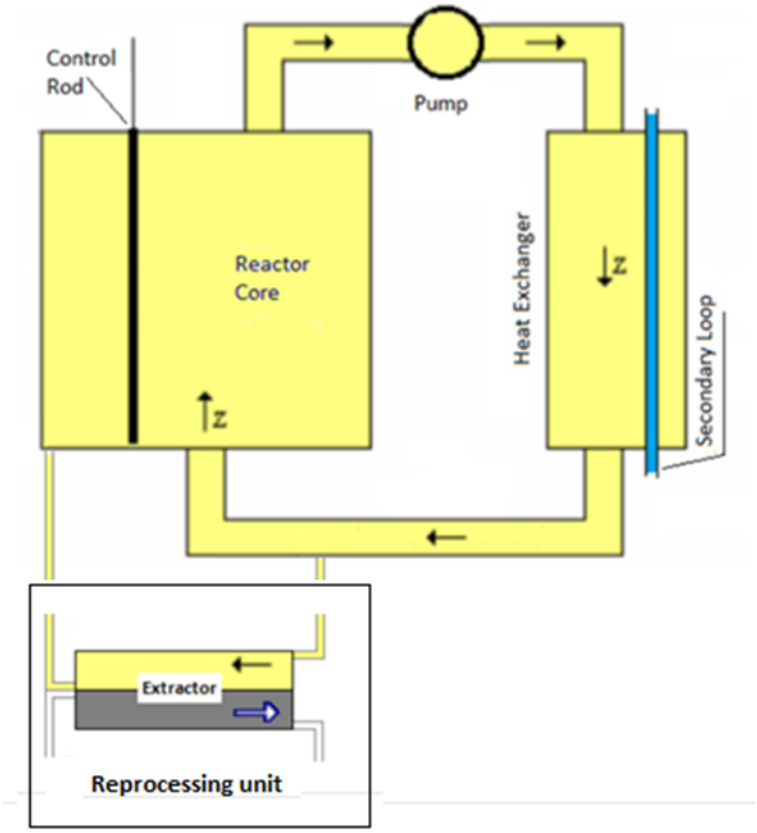
Pomiary Am-241 (prawdopodobnie) jeszcze trwają. Wstępna obróbka wyników pomiarów aktynowców w toku.

BURAN - następny projekt w ZIBJ DUBNA
Przekroje poprzeczne zestawu



P. Zhivkov - XXII Baldin ISHEPP, September 15-20, 2014, Dubna, Russia

Dynamic modeling of MSR
Molten Salt Reactor seen by model



Point reactor kinetics equations for MSR

$$\frac{dP_R}{dt} = \frac{\rho + \rho_{rod} - \beta_{eff} + \alpha_T\left(T - T_0\right)}{\Lambda} P_R + E_f \cdot k_{eff} \sum_{j=1}^{6} \left(\lambda_j - \frac{v_p}{V_R}\left(1 - e^{-\frac{\lambda_j V_O}{v_p}}\right)\right) C_j - \frac{hOLv_p}{V_R}\left(T - T_2\right)$$

$$\frac{dC_j}{dt} = \frac{\beta_j}{\Lambda \cdot E_f k_{eff}} P_R - \left(\lambda_j + \frac{v_p}{V_R}\left(1 - e^{-\frac{\lambda_j V_O}{v_p}}\right)\right) C_j \qquad \mathrm{j} = 1...6$$

$$\frac{dT}{dt} = \frac{1}{\Lambda V_R C_p d}\left(P_R - hOL\left(T - T_2\right)\frac{v_p}{V_R}\right)$$

P_R – reactor power; T – primary loop fuel temperature; T_2 – secondary loop temperature;

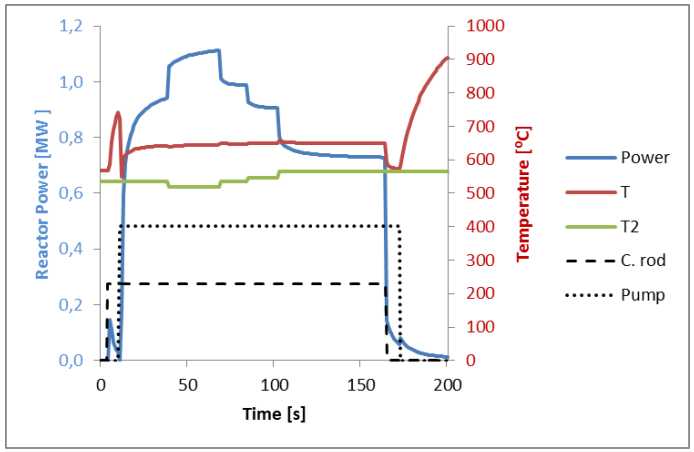
ρ - core reactivity; ρ_{rod} – control rod weight; α_T – temperature reactivity coefficient;

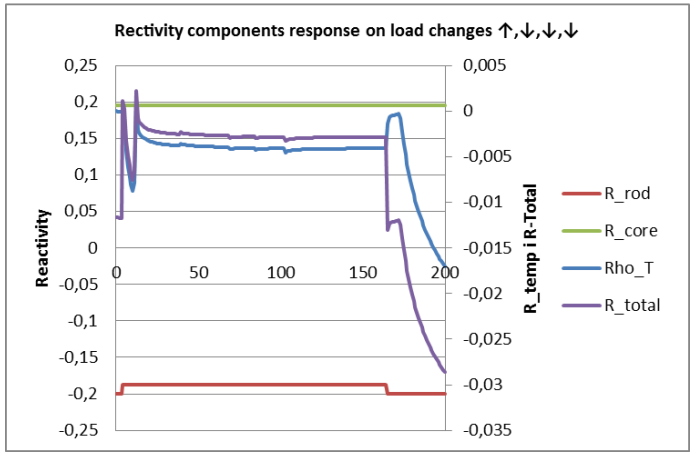
β_i – i-th group of delayed neutrons contribution to neutron balance;

Λ -neutron generation life time; V_p – coolant flow/pumping rate; V_R – reactor volume;

V_O – outer volume; $C_i(t)$ – number of nuclei of i-th group precursor; O - heat exchanger circumference; L - heat exchanger length; C_p – specific heat; d – fuel salt density

MSR behavior simulation – rod up, pump on, then power load modification





Simulation run:

- Rod ↑
- Pump on
- Load ↑,↓,↓,↓
- Rod ↓
- Pump off

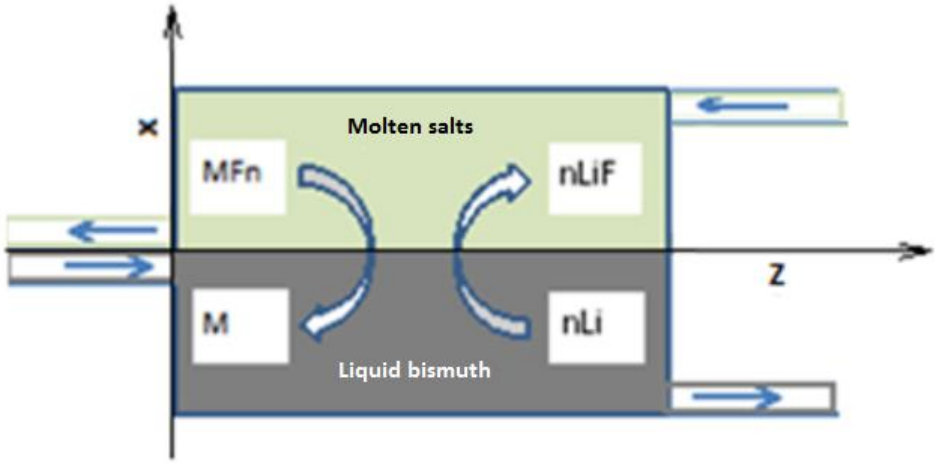
Power load modification by T_2 modification.

Note!

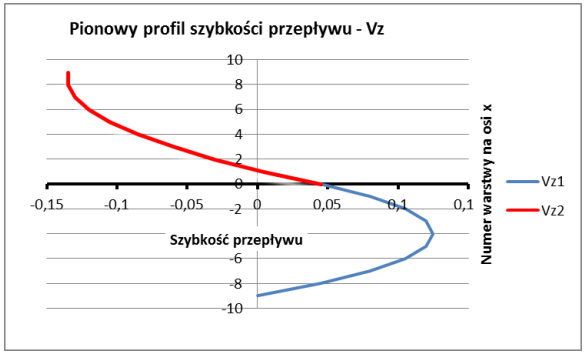
ρ_{core} i ρ_{rod} on left axis while ρ_{Temp} and ρ_{Total} on right. Pump speed and rod position not in scale.

Conclusion – Reactor power automatically follows power load changes. Fuel temperature weakly reacts on power load changes.

MSR reprocessing unit
Extractor scheme



$$MF_n(s) + nLi(b) - - > nLiF(s) + M(b)$$



Extraction/fluid dynamics laws, assumptions and methods

$$\frac{\partial c}{\partial t} = \mp\lambda c - V_z \frac{\partial c}{\partial z} + D\frac{\partial^2 c}{\partial x^2} + D\frac{\partial^2 c}{\partial z^2}$$

- Conservation of mass (mass continuity)

$$\frac{\partial p}{\partial z} = \mu \frac{\partial^2 V_z}{\partial x^2}$$

- Conservation of momentum (Newton's second law of motion or Navier-Stokes equations)

$$c(x,z)\big|_{t=0} = c_0(x,z)$$

- Initial conditions

$$V_z(x)\big|_{x=x_0} = 0$$

- Boundary conditions

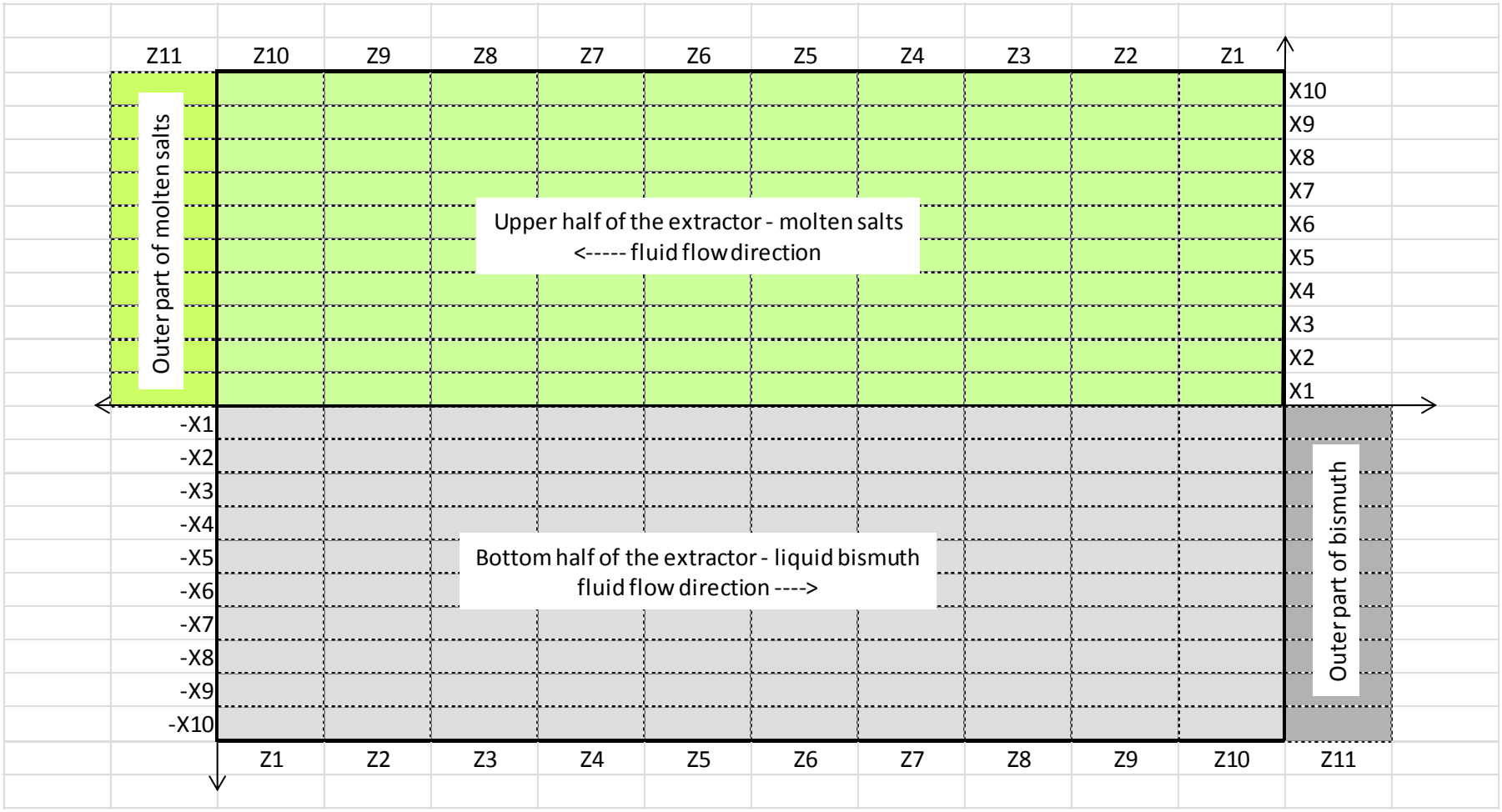
c – concentration
λ – reaction rate
μ – viscosity coefficient
D – diffusion coefficient
V – fluid flow velocity

Apart from that:
- Extractor geometry unchangeable
- Fluid flow profile – stable, not dependent on substance concentration
- Outer forces (*ρF*) neglected
- Energy conservation law omitted – extractor fluid temperatures constant

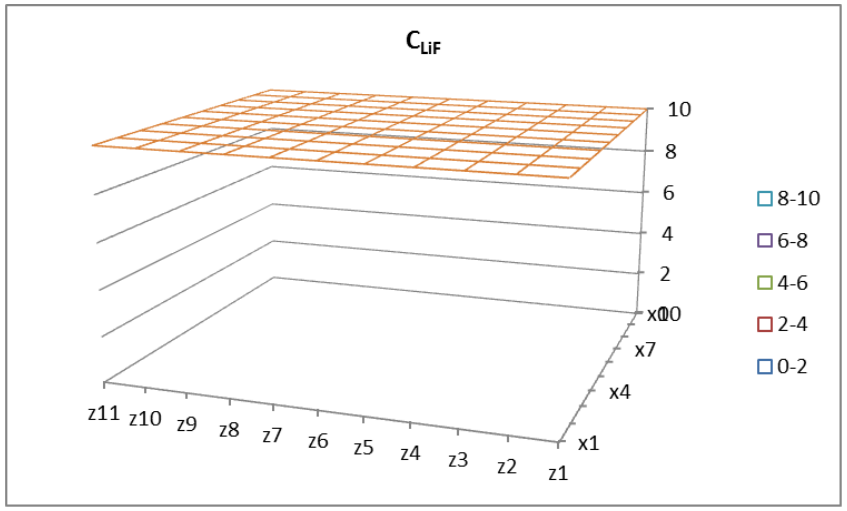
- Central finite difference approximation applied for space part of equations.
- Explicit RK4 method for time part solution then

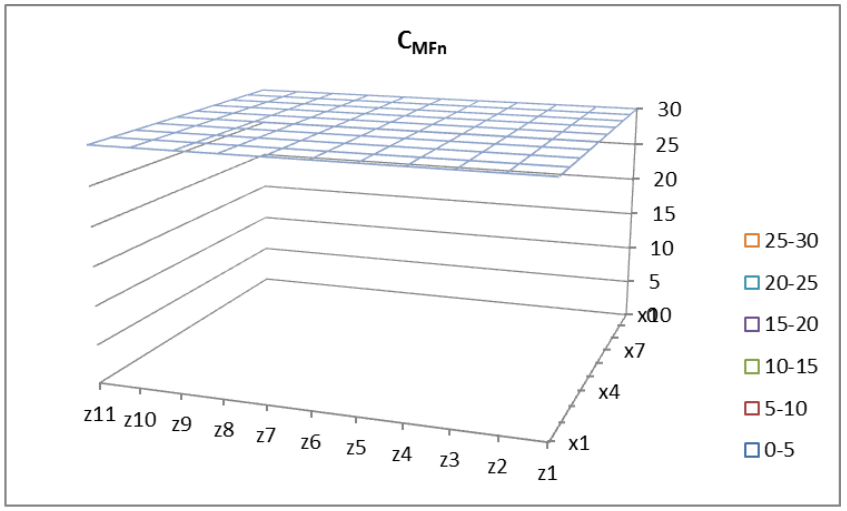
The extractor model tested with dummy parameters for while.

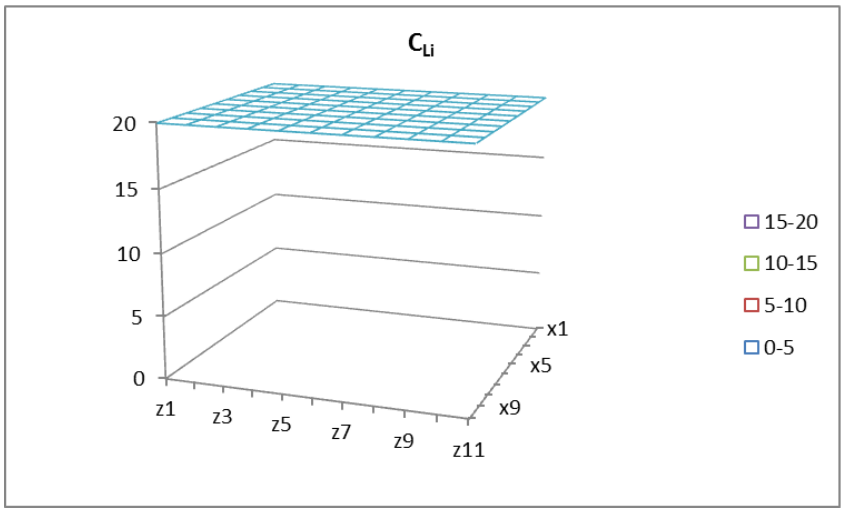
Extractor modeling mesh

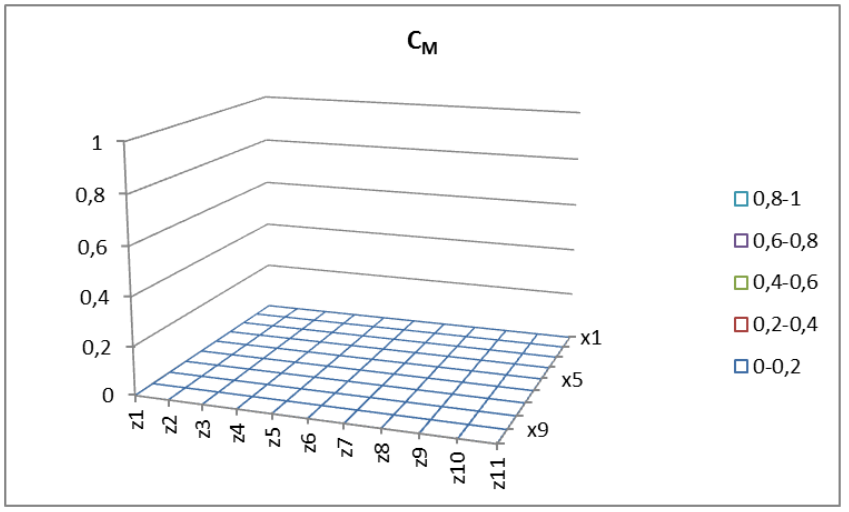


Extraction process substances initial concentration distribution (t=0)

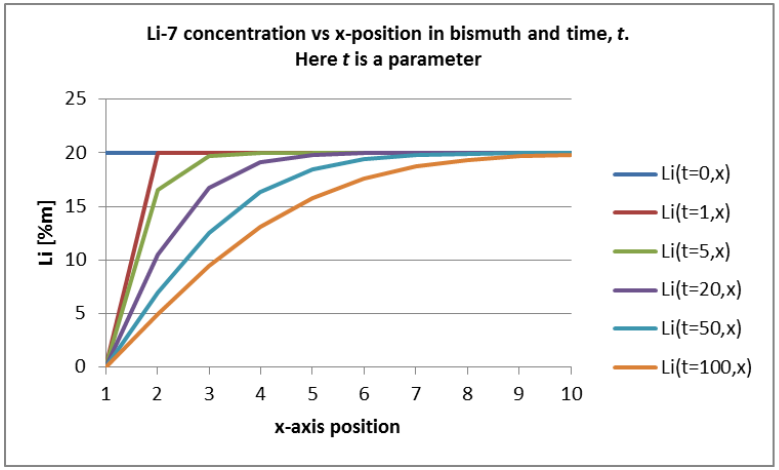


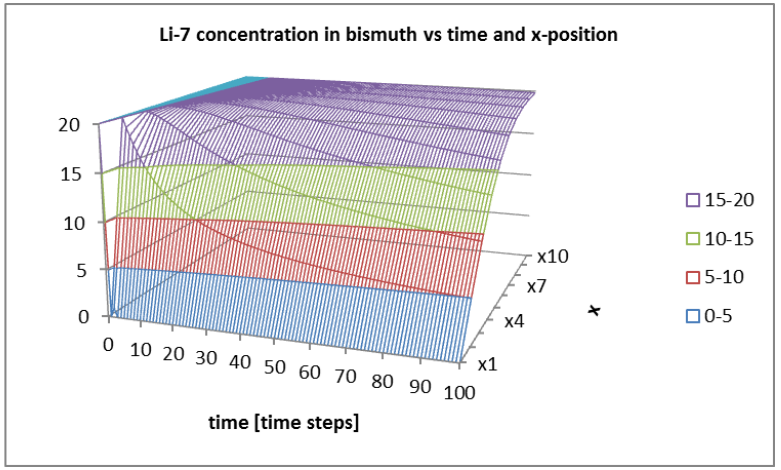


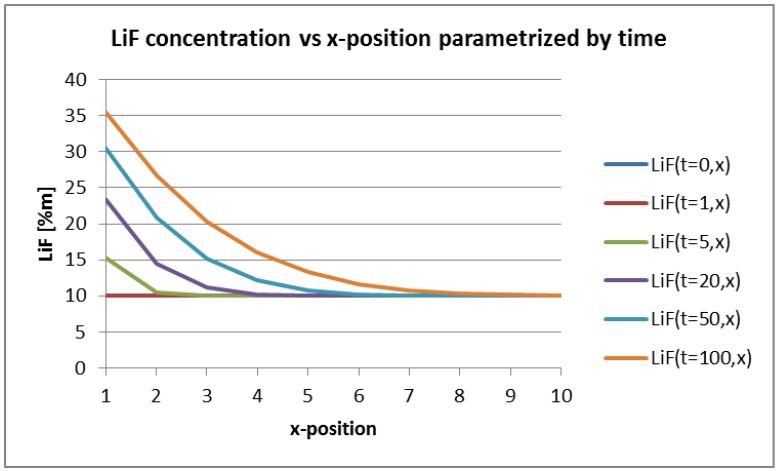


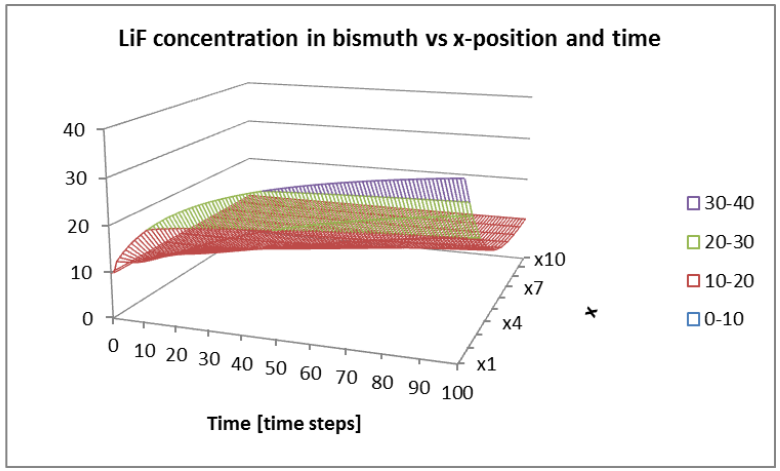


1D view of Li and LiF concentration dependence on x-position and time

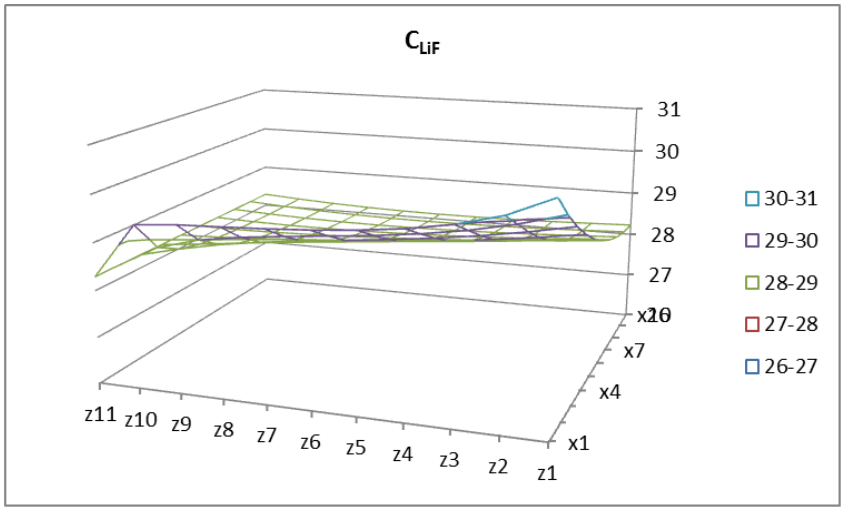


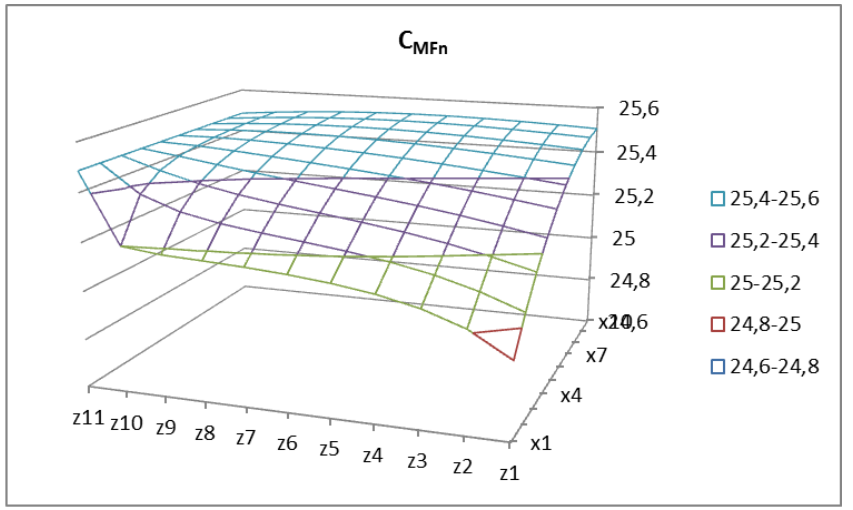


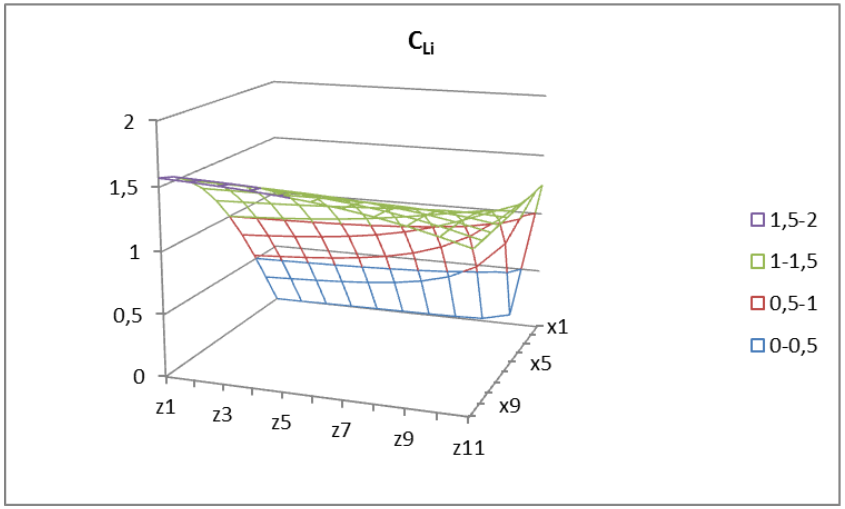


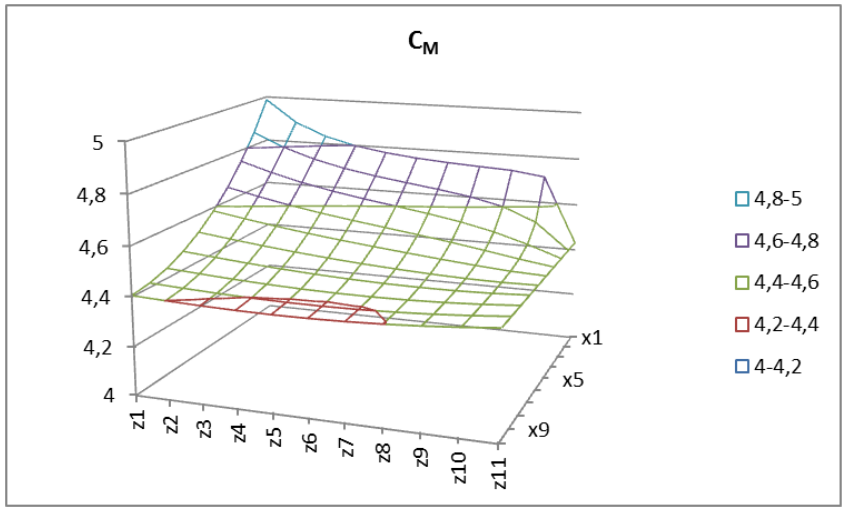


Extraction process substances concentration distribution after some more time









To be done yet

In extractor
- Energy conservation law to be included in extractor simulation
- More components to be taken into account

Back-extraction process to be simulated

Online reprocessing simulation completed

Testing on real reactor data and compare with DYN1D and DYN3D

Podsumowanie

Prace w ZIBJ warto kontynuować gdyż dotyczą szczególnie interesujących obecnie zagadnień fizyki reaktorowej – wypalanie odpadów jądrowych, zamknięcie cyklu paliwowego.

Prace w ZIBJ warto kontynuować gdyż jest tam odpowiednia aparatura – akceleratory, zestaw KWINTA, zestaw BURAN, detektory pomiarowe.

Prace są wykonywane w wielonarodowym zespole badawczym, co daje gwarancję ich kontynuowania i ukończenia.

Koniec, dziękuję za uwagę

Slajdy rezerwowe

Wzory na wyznaczanie liczby reakcji z danych doświadczalnych - cd

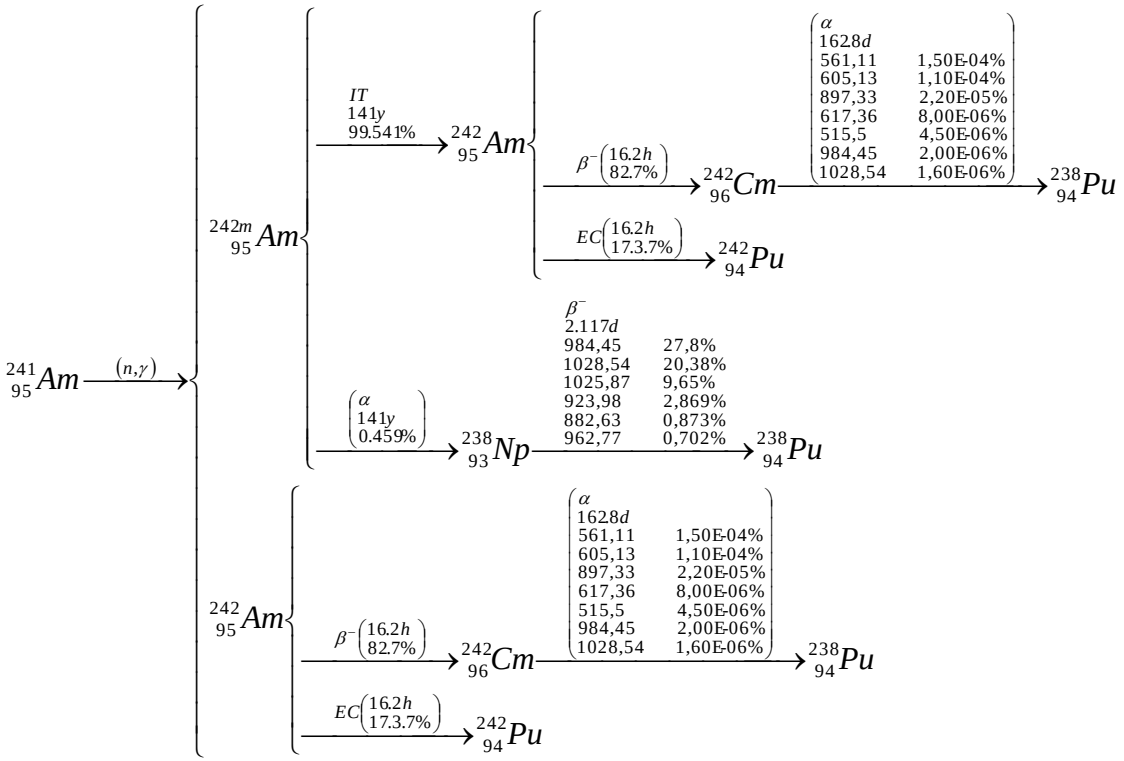
$$I_{f1\gamma} = \frac{A_{1\gamma}}{m\phi\varepsilon_\gamma I_\gamma}\frac{\lambda_1 t_{irr}}{\left(1-e^{-\lambda_1 t_{irr}}\right)}\frac{e^{\lambda_1 t_+}}{\left(1-e^{-\lambda_1 t_m}\right)}\frac{t_{real}}{t_{live}}$$

$$I_{f2\gamma} = \frac{A_{2\gamma}}{m\phi\varepsilon_\gamma I_\gamma}\frac{t_{real}}{t_{live}}\frac{t_{irr}}{\left[c_2\left(1-e^{-\lambda_2 t_{real}}\right)e^{-\lambda_2 t_+} - c_1\left(1-e^{-\lambda_1 t_{real}}\right)e^{-\lambda_1 t_+}\right]}$$

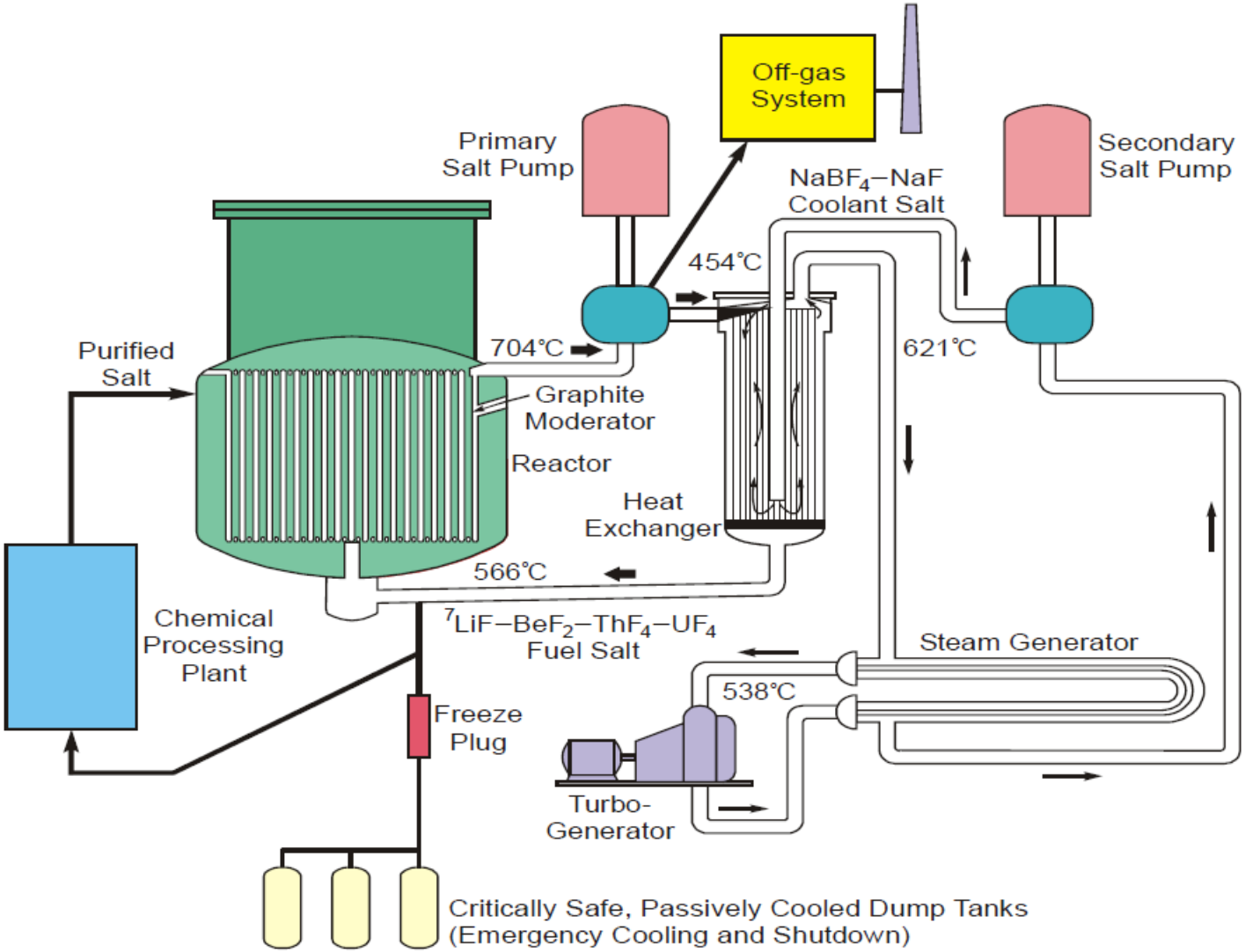
$$c_2 = \left(\frac{\gamma_1+\gamma_2}{\lambda_2} + \frac{\gamma_1}{\left(\lambda_1-\lambda_2\right)}\right)\left(1-e^{-\lambda_2 t_{irr}}\right)$$

$$c_1 = \frac{\gamma_1}{\lambda_1-\lambda_2}\left(1-e^{-\lambda_1 t_{irr}}\right)$$

Łańcuchy reakcji po wychwycie neutronu Am-241(n,γ)



Molten Salt Reactor scheme



Molten Salt Reactors (MSRs) for Electricity and Waste Destruction; Dr. Charles Forsberg; Presentation of Generation IV Nuclear Energy System Conception of Nuclear Energy (DOE/NE-1);June 7, 2002

Intermediate and long term changes
Bateman equations for isotopic changes

$$\frac{dN_i}{dt} = \sum_j \gamma_{ij}\sigma_{f,j}N_j\phi + \sigma_{c,i-1}N_{i-1}\phi + \lambda_i^{'}N_i^{'} - \sigma_{f,i}N_i\phi - \sigma_{c,i}N_i\phi - \lambda_i N_i$$

$\sum_j \gamma_{ij}\sigma_{f,j}N_j\phi$ - isotope *j* production from fission

γ_{ij} – isotope *i* production efficiency during *j-th* isotope fission

$\sigma_{c,i-1}N_{i-1}\phi$ - isotope *i* production by neutron capture in *i-1-th* isotope

$\lambda_i^{'}N_i^{'}$ - production by decay

$\sigma_{f,i}N_i\phi$ - loss because of fission

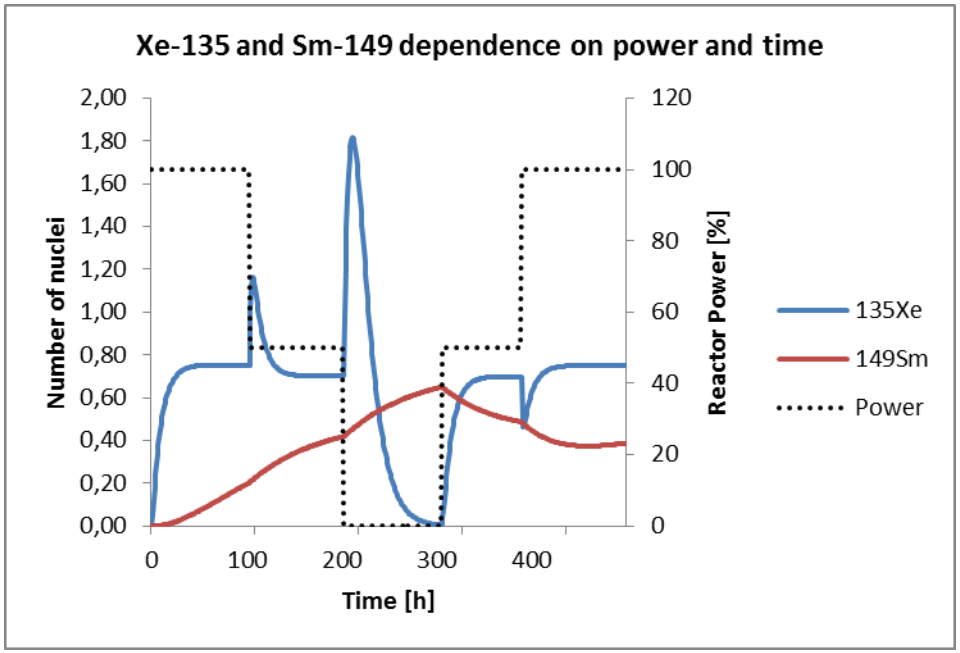
$\sigma_{c,i}N_i\phi$ - isotope *i* loss because of neutron capture

$\lambda_i N_i$ - isotope *i* loss by decay

Set of 65 equations. Coefficients from [1]. Solved with RK4 method

Calculation step = 1 hour

Xe-135 and Sm-149 concentration response to power change - 2



1. Reactor power decrease (from 100% to 50%) makes Xe-135 concentration grow and then decrease.
2. Rector power growth from 0 to 50% makes Xe-135 grow but next 50% growth makes Xe-135 decrease and then increase.