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Void reactivity coefficient for liquid metal
cooled reactor

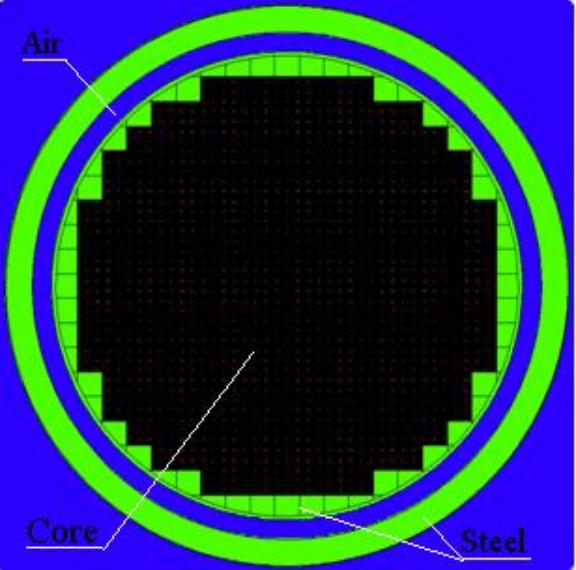
February 2023

Outline

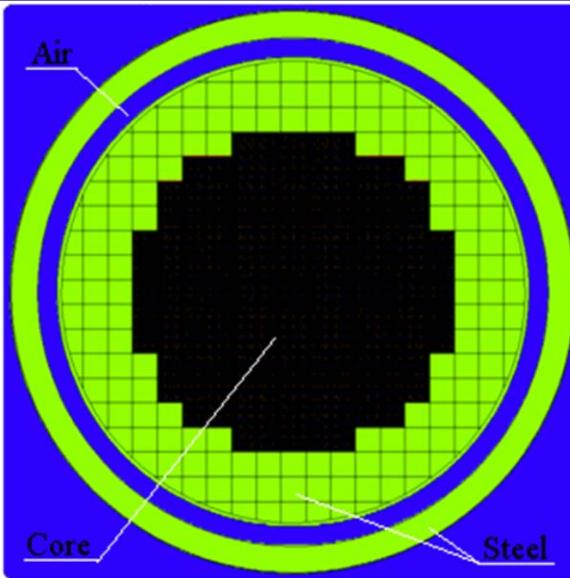
- 1. Introduction
- 2. Geometry and materials
- 3. Definitions
- 4. Calculation results of void reactivity coefficient
- 5. Loss of neutrons
- 6. Analytical approach
- 7. Conclusion

Introduction

- The liquid metal cooled reactors can be considered as a future of nuclear plants because of their features:
 - - breeding and, as consequence, high value of burnup and economic rationale
 - - can achieve high temperature of coolant
 - - metal coolants remove heat from reactor core more rapidly and allow much higher power density.
- There are only about 20 experimental and commercial fast neutron reactors in operation worldwide.
- The reasons is that fast reactors are usually cooled by liquid metal and have positive void reactivity coefficient (α_v) and it is difficult to decrease it.



Large core, 241 assemblies

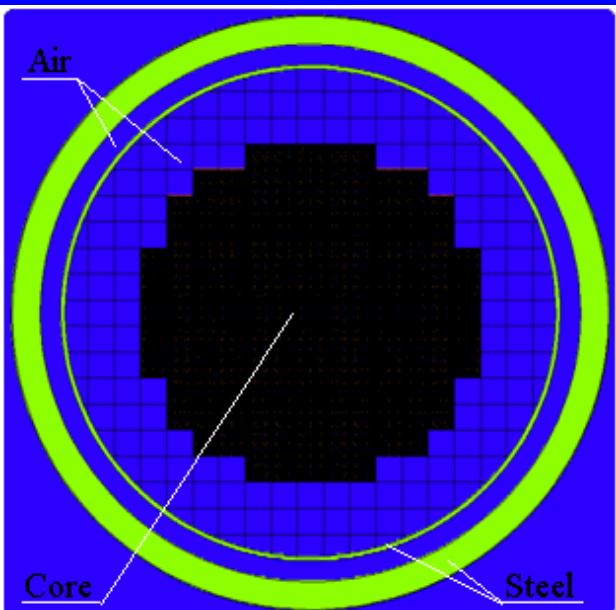


Small core + reflector,
137 assemblies

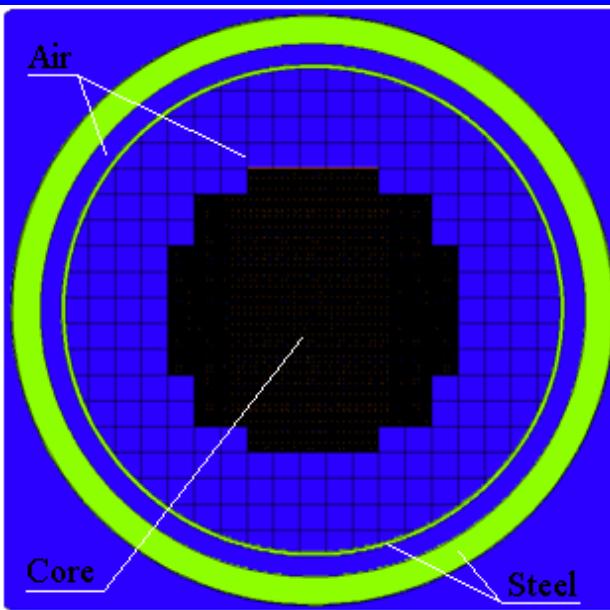
Geometry

The colors mean: black - core, blue - air, light green – steel.

The computer simulation model is based on the geometry and materials of European Pressurized Reactor (EPR)

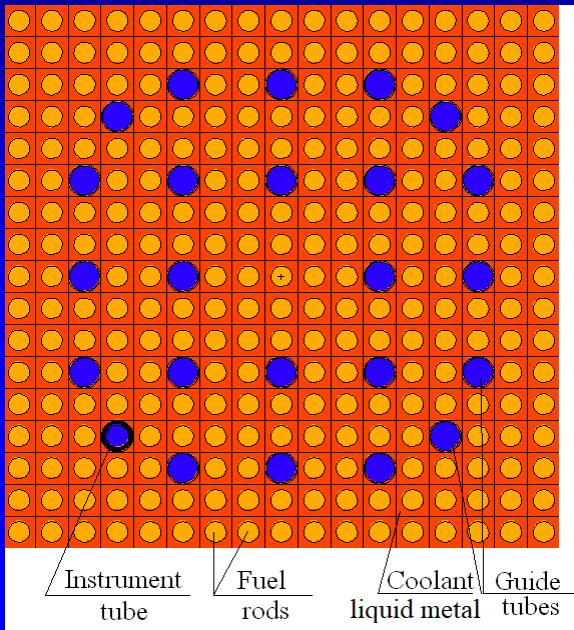


Small core, 137 assemblies



Very small core, 101 assemblies

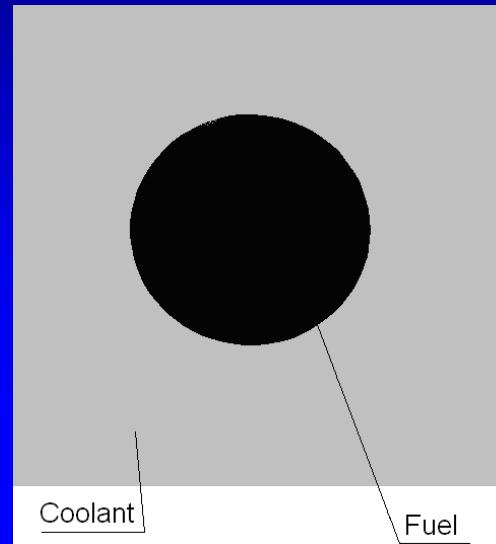
Geometry and materials



The modified A1 assembly
based on EPR

Fuels:
U-238+Pu-239
 $(U+Pu-239)O_2$
Th-232+U-233
 $(Th+U-233)O_2$

Colants:
Na
Pb



Fuel cell configuration. Dimension of the cell 1.26x1.26cm. Radius of fuel rod and coolant and fuel volume are changeable

VCR=Coolant volume/Cell volume

Definitions

Herein is used the following definition of void reactivity coefficient (α_V)

$$\alpha_V(\text{void}) = \frac{k_{\text{void}} - k_{\text{norm}}}{k_{\text{norm}}}$$

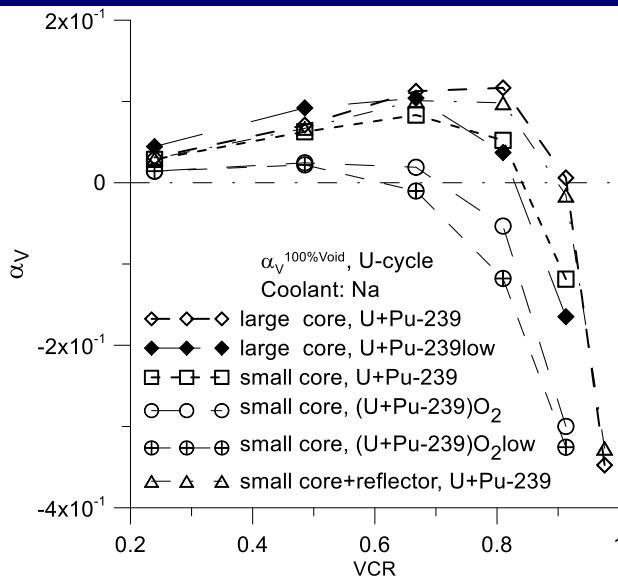
Here,
void [%] is an average value of void fraction in the fuel cells ,
 k_{norm} , k_{void} – effective multiplication factors at normally work and voided reactor core,
respectively.

$\alpha_V^{\text{void}}(VCR)$ α_V as a function of VCR parameter and constant value of void

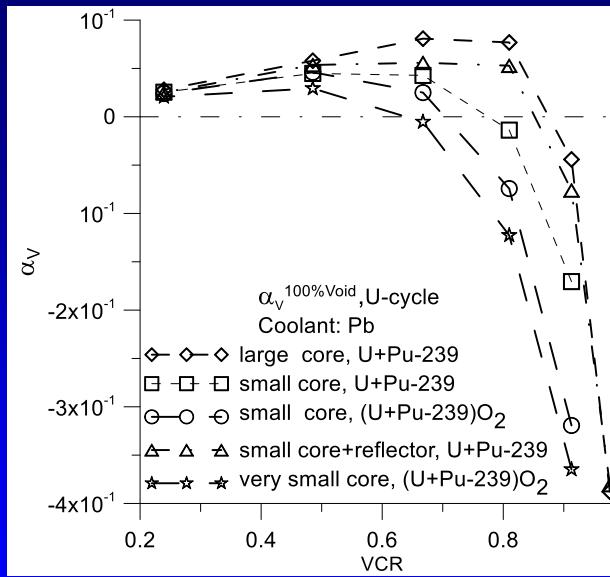
$\alpha_V^{VCR}(\text{void})$ α_V as a function of void parameter and constant value of
VCR parameter

$\alpha_V(100\%void)$ means 0% of coolant density. This means that volume of coolant is filled by air at atmospheric pressure. This case simulates the Loss of coolant accident (LOCA).

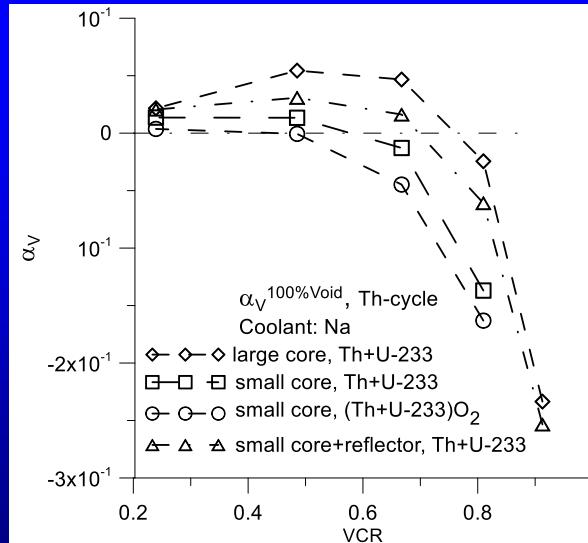
Calculation results



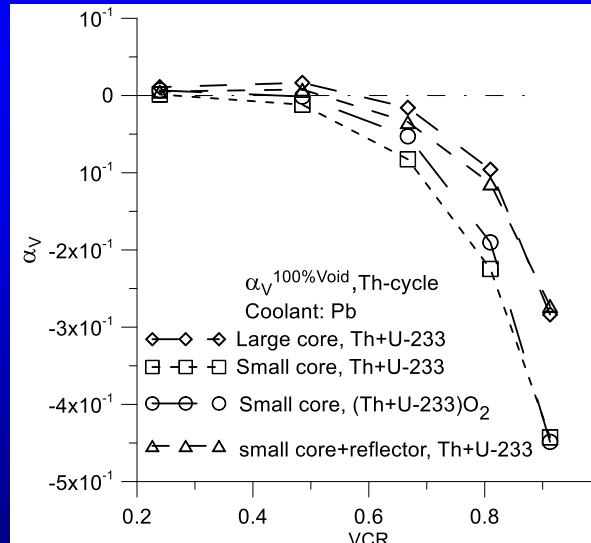
U-cycle for coolant Na



U-cycle for coolant Pb



Th-cycle for coolant Na



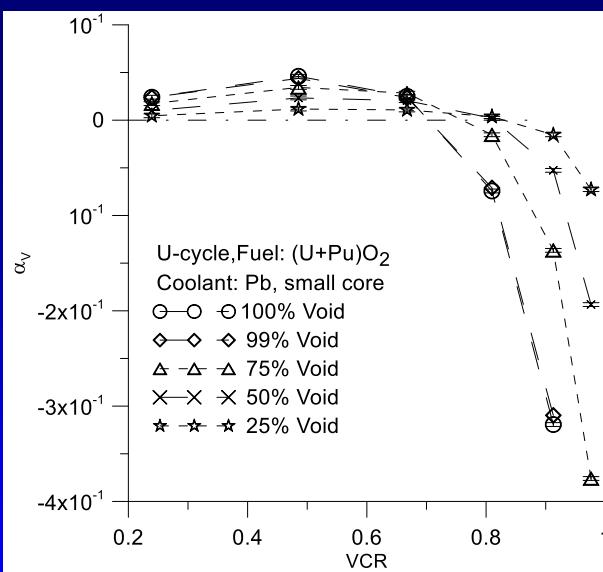
Th-cycle for coolant Pb

$\alpha_V^{100\% \text{void}}(VCR)$ as a function of VCR parameter

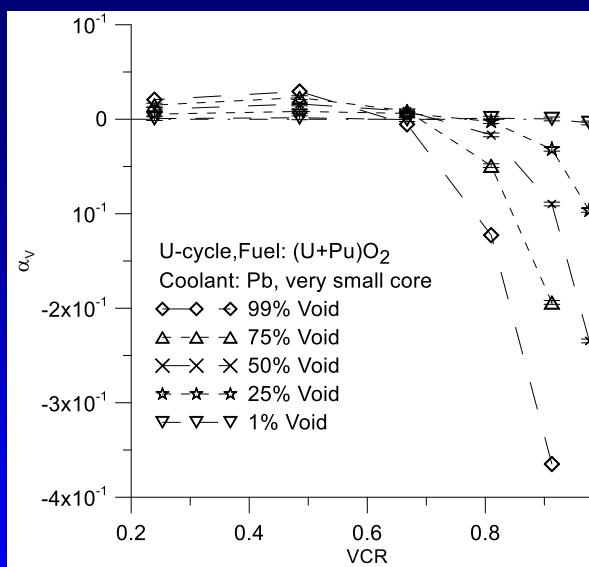
$VCR_0^{100\% \text{void}}$ Is a very important parameter of reactor

'low' means low value of average density of fuel.
Statistical error is equal to 0.002

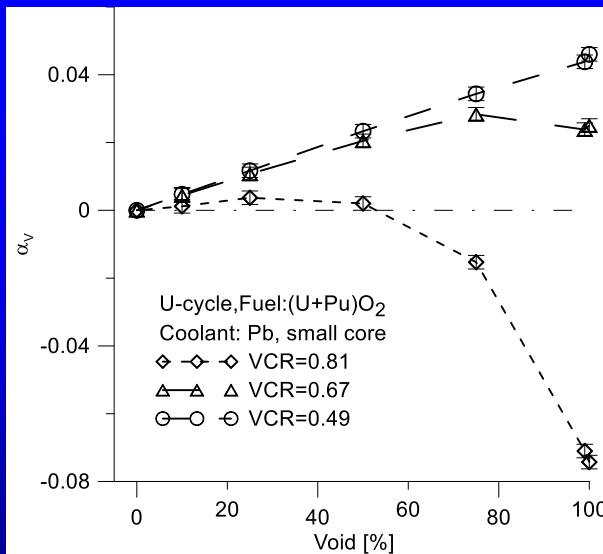
Calculation results



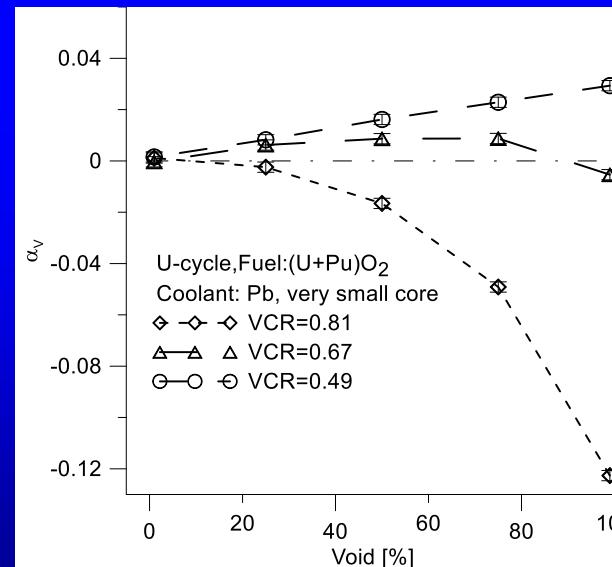
$\alpha_V^{n\%void}(VCR)$ for n equal to 25%, 50%, 75%, 99%, 100% for small core.



$\alpha_V^{n\%void}(VCR)$ for n equal to 1%, 25%, 50%, 75%, 99%, 100% for very small core.



$\alpha_V^{VCR}(void)$ as a function of void for small core for VCR=0.49, 0.67, 0.81



$\alpha_V^{VCR}(void)$ Same as on the left figure but for very small core

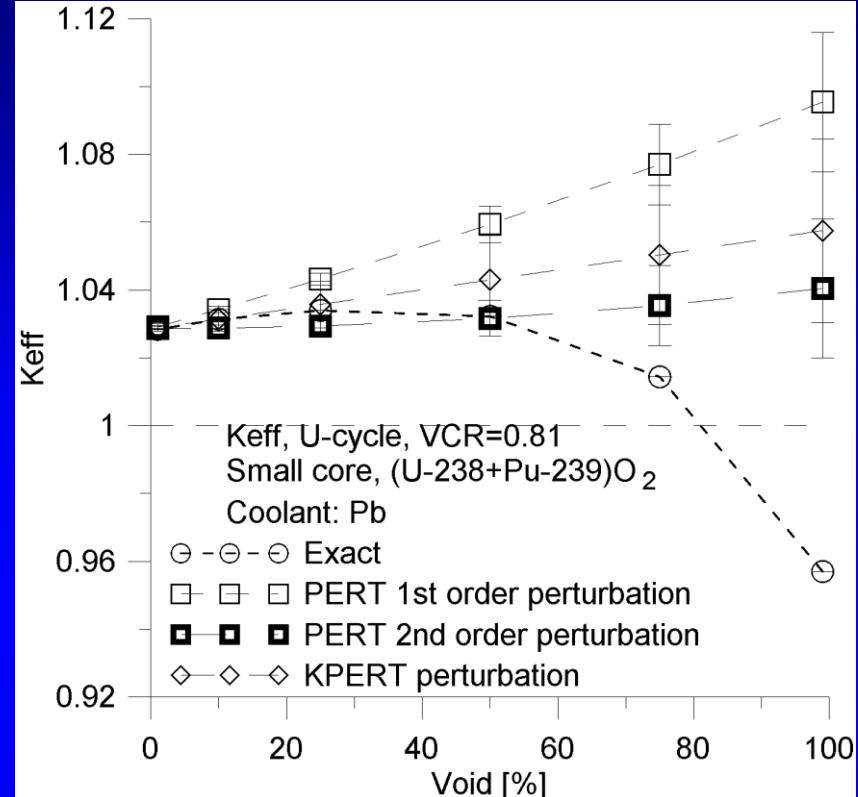
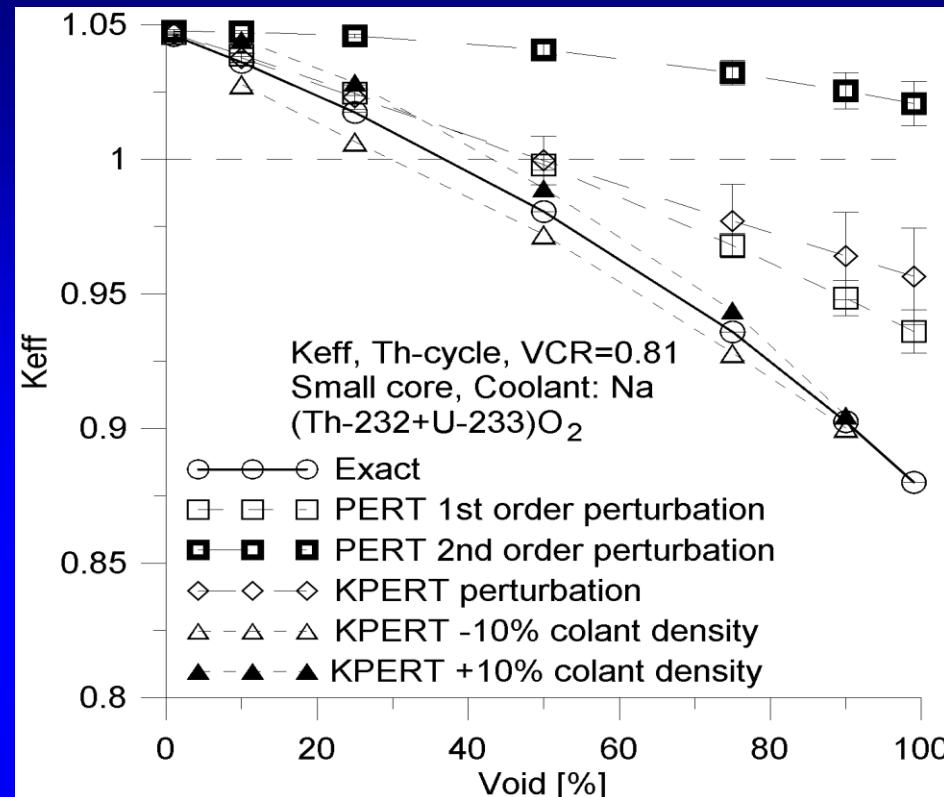
$\alpha_V^{n\%void}(VCR)$ as a function of VCR parameter

U-cycle for coolant Pb

$\alpha_V^{VCR}(void)$ as a function of void parameter

Statistical error is equal to 0.002

Perturbation method



The PERT perturbation method is limited to the 1st and 2nd order terms of a Taylor series expansion.

The KPERT card gives a first-order *estimate* of the reactivity worth of the density perturbation

Jeffrey A. Favorite, "On the Accuracy of the Differential Operator Monte Carlo Perturbation Method for Eigenvalue Problems," *Transactions of the American Nuclear Society*, **101**, 460–462 (2009).

Jeffrey A. Favorite, "An Alternative Implementation of the Differential Operator (Taylor Series) Perturbation Method for Monte Carlo Criticality Problems," *Nuclear Science and Engineering*, **142**, 3, 327–341 (2002); <https://doi.org/10.13182/NSE02-A2311>

Loss of neutrons - definitions

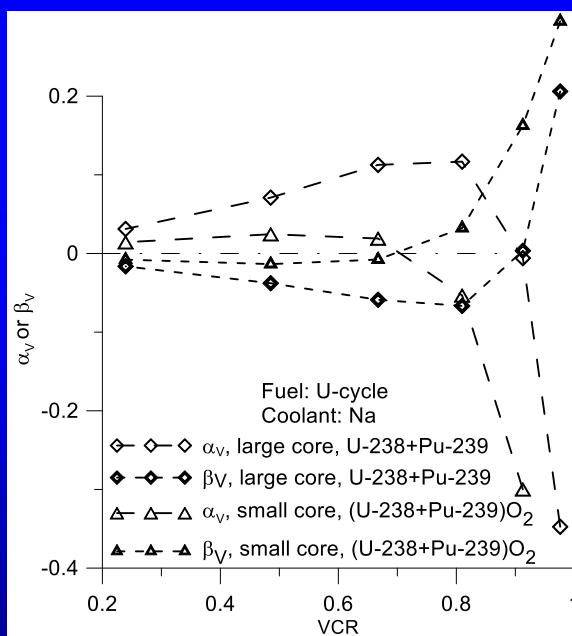
$$Loss(VCR) = Absorption(VCR) + Escape(VCR)$$

Absorption- means absorption fraction without absorption neutron induce fissions i.e. (n,γ)

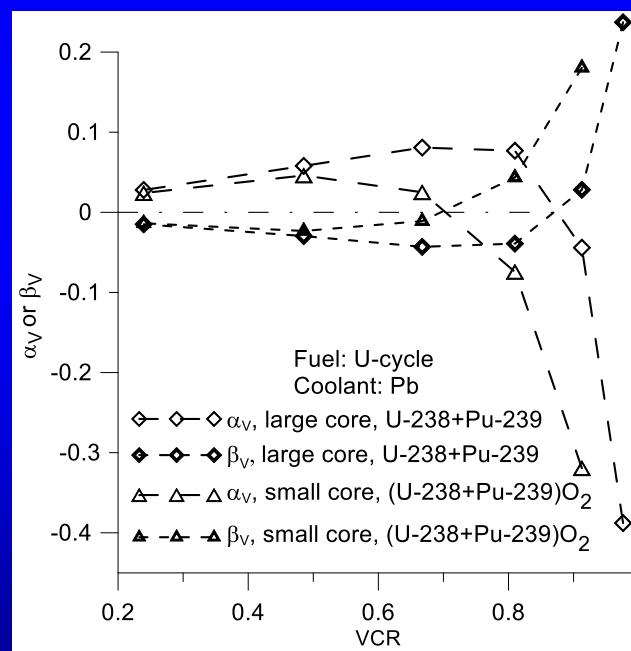
Escape - means total escaped fraction of neutrons. Loss is a function of VCR and void parameter.

$$\beta_V^{n\%void} = \frac{Loss_{void} - Loss_{norm}}{Loss_{norm}}$$

$Loss_{norm}$, $Loss_{void}$ – means loss neutrons fraction at 0% void and n% void respectively.



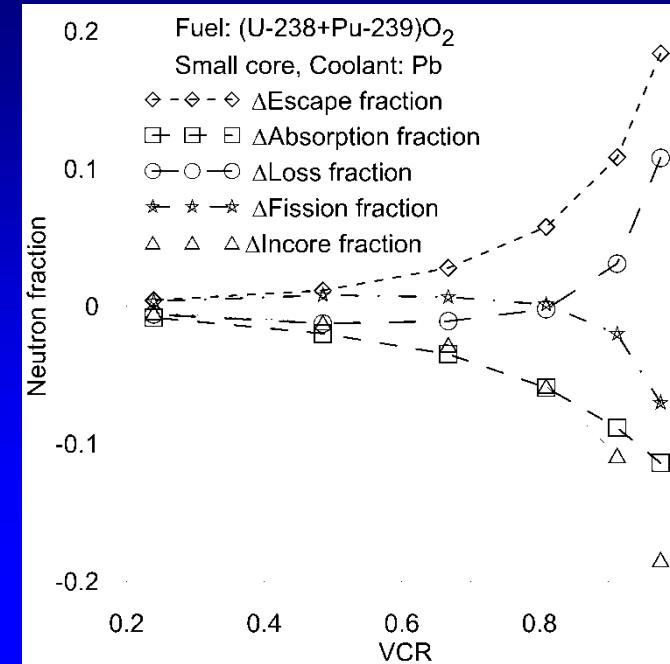
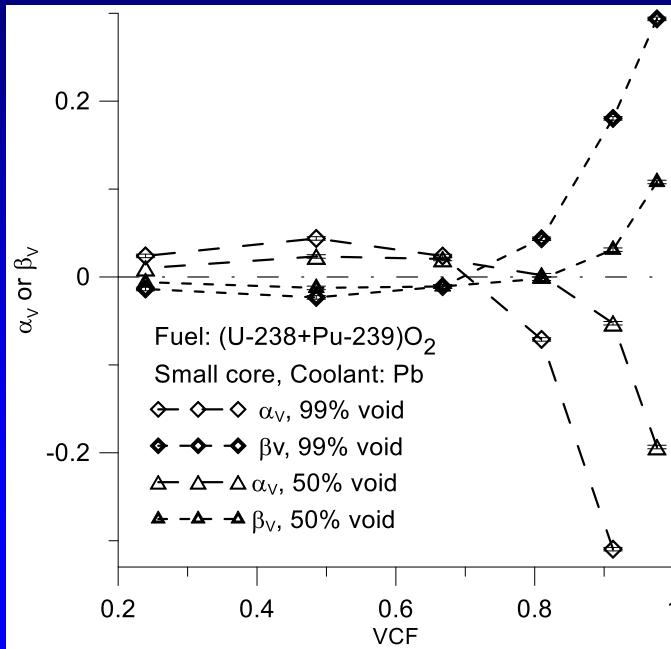
Coolant: Na



Coolant: Pb

The $\alpha_V^{100\%void}(VCR)$ and
the $\beta_V^{100\%void}(VCR)$ for
U-cycle.

Loss of neutrons



Change of the Escape , Absorption, Loss and Fission fraction during 99% void. Small core

Please note, that ΔLoss fraction has a negative sign of $\Delta\text{fission}$ fraction

The main reason of positive value ΔLoss fraction is fast increasing of ΔEscape fraction

Discussion

Size of core	Fuel	Coo-lant	$VCR_0^{100\%void}$ Error <1%	Enrichment [%] Error <0.5%	Average neutron energy [MeV] Error<1%	Pu-239 mass [g] Error<1%	Total actinides mass [g] Error<1%
large	U+Pu	Na	0.914	10.6	0.69	3.90E+05	3.67E+06
large	U+Pu low density	Na	0.836	10.6	0.72	7.37E+05	6.96E+06
small	U+Pu	Na	0.866	9.6	0.87	3.12E+05	3.25E+06
small	(U+Pu)O ₂	Na	0.705	9.8	0.81	6.13E+05	6.29E+06
small	(U+Pu)O ₂ Low density	Na	0.619	10.3	0.81	8.37E+05	8.14E+06
very small	(U+Pu)O ₂	Na	0.681	9.8	0.84	4.94E+05	5.02E+06
very small	(U+Pu)O ₂ Low density	Na	0.537	9.2	1.12	6.70E+05	7.28E+06
large	U+Pu	Pb	0.887	11.1	0.68	5.36E+05	4.82E+06
small	U+Pu	Pb	0.797	7.0	0.81	3.41E+05	4.91E+06
small	(U+Pu)O ₂	Pb	0.717	9.8	0.68	5.92E+05	5.89E+06
vsmall	(U+Pu)O ₂	Pb	0.667	10.1	0.71	5.26E+05	5.10E+06

Analytical method

Herein is presented estimated method of obtaining α_v based on uniform reactor without reflector

$$k_{norm} = \frac{\eta \Sigma_{f,norm}}{(\Sigma_{a,norm}^{Tot})(1 + B^2 L_{norm}^2)}$$

$$L_{norm}^2 = \frac{1}{3\Sigma_{a,norm}^{Tot} \Sigma_{s,norm}^{Tot} (1 - \bar{\mu}_{norm})}$$

$$k_{void} = \frac{\eta \Sigma_{f,void}}{(\Sigma_{a,void}^{Tot})(1 + B^2 L_{void}^2)}$$

$$L_{void}^2 = \frac{1}{3\Sigma_{a,void}^{Tot} \Sigma_{s,void}^{Tot} (1 - \bar{\mu}_{void})}$$

$$\alpha_V(void) = \frac{k_{void} - k_{norm}}{k_{norm}}$$

$$\alpha_v \approx \frac{\Sigma_{a,norm}^{Tot} - \Sigma_{a,void}^{Tot} + B^2(D_{norm} - D_{void})}{(\Sigma_{a,void}^{Tot})(1 + B^2 L_{void}^2)} = \frac{\Delta \Sigma_a^{Tot} + B^2 \Delta D}{(\Sigma_{a,void}^{Tot})(1 + B^2 L_{void}^2)}$$

$$R_{f,vn} = \frac{\Sigma_{f,void}}{\Sigma_{f,norm}} = 1$$

$$D = L^2 \Sigma_a = 1 / (3 \Sigma_s (1 - \bar{\mu})$$

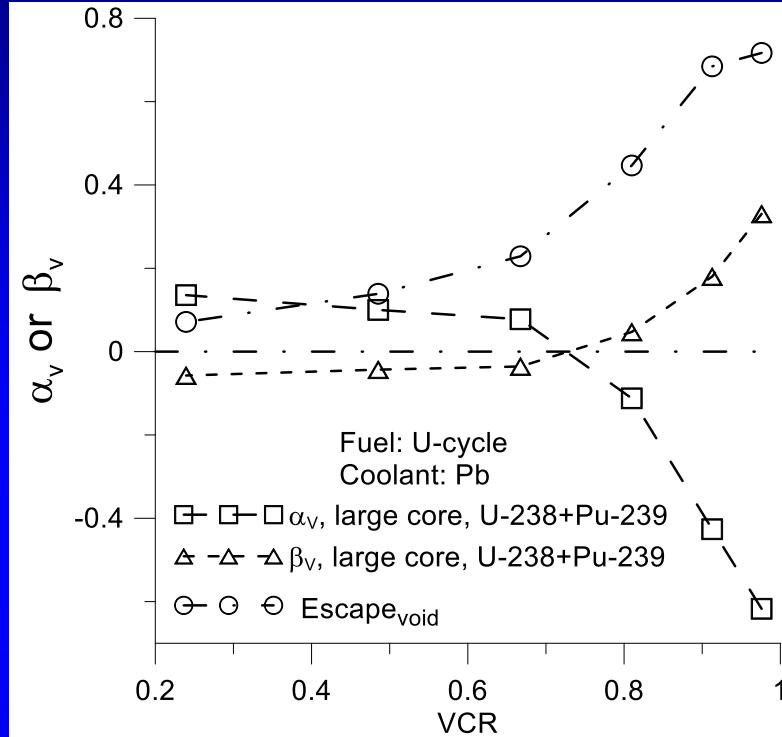
$$B^2 = \left(\frac{2.405}{R} \right)^2 + \left(\frac{\pi}{H} \right)^2$$

neutron diffusion length

$$L_{void}^2$$

geometry buckling (B^2)

Analytical method - calculation results



$$\text{Escape}_{\text{void}} = \frac{B^2 L_{\text{void}}^2}{1 + B^2 L_{\text{void}}^2}$$

$\beta_V^{100\%void}(VCR)$ and $\alpha_V^{100\%void}(VCR)$ and $\text{Escape}_{\text{void}}(VCR)$ obtained from analog method for fuel of U-238+Pu-239 for large core without external steel barrel and vessel

The analytic method is an approximation method of determining α_v only. However, it explains influence of material properties such as corresponding cross sections, atomic density and size and shape of reactor core on decreasing its value

Conclusion and remarks

To reduce value of α_v one can should:

- decrease or delete a reflector
- increase the value of VCR parameter
- decrease size or shape of reactor core
- decrease the average value of fuel density .

In other words, the negative value of α_v one can achieve by employing corresponding value of $B^2 L_{void}^2$ of reactor core .

The enough high value of VCR parameter is the most important to achieve a negative value of α_v . However, the high value of VCR is not advantageous and should be reduced. A very convenient parameter to determine the reduction value of α_v is the $VCR_0^{100\%void}$. This parameter can be used to compare different method of reduction α_v .

The given low value of α_v can be achieved using various methods. The method of reducing of size core decrease the critical mass of fuel whereas the method of reducing of average fuel density increase the critical mass of fuel. The simultaneous use of both methods significantly increases the average neutron energy and may not change the critical mass of the fuel.

The method of reducing the average fuel density can be used to construct a large and effective reactor with negative value of the α_v during LOCA accident.

The End

Thank you for your attention
Questions?

Analitycal method

One can assume the following form of $\Sigma_{a,norm}^{Tot}$ and $\Sigma_{s,norm}^{Tot}$

$$\Sigma_{a,norm}^{Tot} = VCR\rho^c\sigma_a^c + (1 - VCR)\rho^f\sigma_a^f$$

$$\Sigma_{s,norm}^{Tot} = VCR\rho^c\sigma_s^c + (1 - VCR)\rho^f\sigma_s^f$$

ρ^c ρ^f – atomic density of coolant and fuel

σ_a^c σ_a^f σ_s^c σ_s^f - microscopic cross section of (n,a) and $(n,n_{elastic})$ for fuel and coolant.

$$\Sigma_{a,void}^{Tot} = \Sigma_{a,norm}^{Tot} + \frac{\delta\Sigma_{a,void}^{Tot}}{\delta\rho^c}\delta\rho^c = \Sigma_{a,norm}^{Tot} + VCR\sigma_a^c\delta\rho^c = \Sigma_{a,norm}^{Tot} - \Delta\Sigma_a^{Tot}$$

$$\Sigma_{a,norm}^{Tot} - \Sigma_{a,void}^{Tot} = -VCR\sigma_a^c\delta\rho^c = \Delta\Sigma_a^{Tot}$$

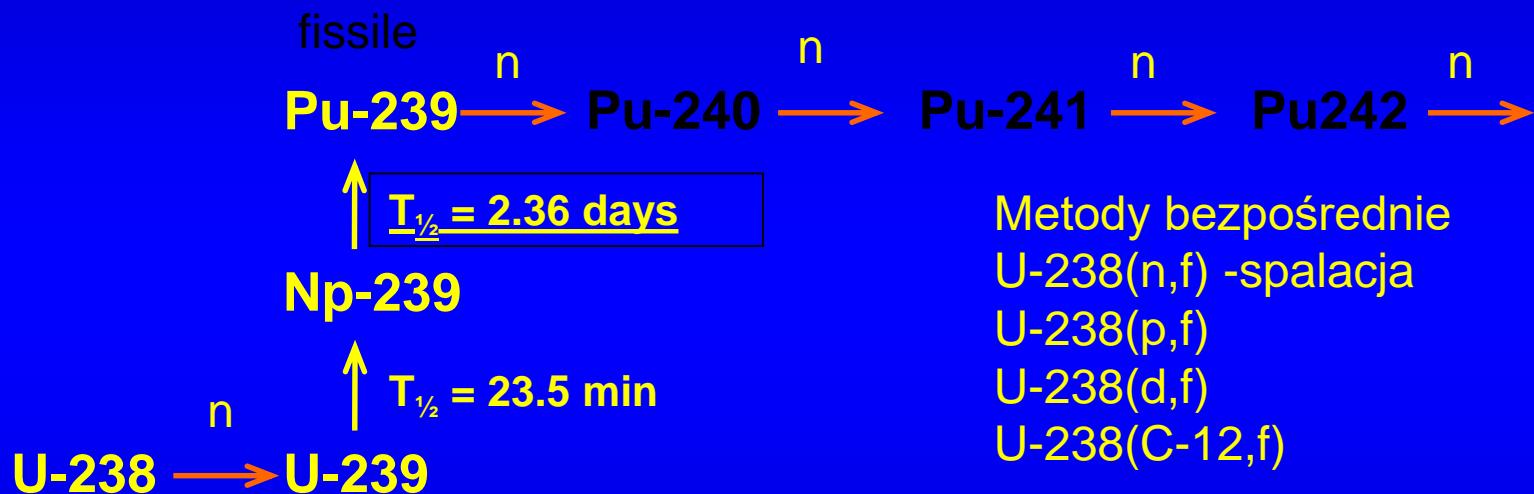
The value $\Delta\Sigma_a^{Tot}$ is positive because $\delta\rho^c$ is negative during evaporation of coolant.

$$D_{norm} - D_{void} = AD_{norm}^2 VCR\sigma^c\delta\rho^c = \Delta D$$

$$A = 3(1 - \bar{\mu})$$

The value ΔD is negative for negative value of $\delta\rho^c$ during evaporation of coolant.

Schemat wypalania U-238 i produkcji Pu-239

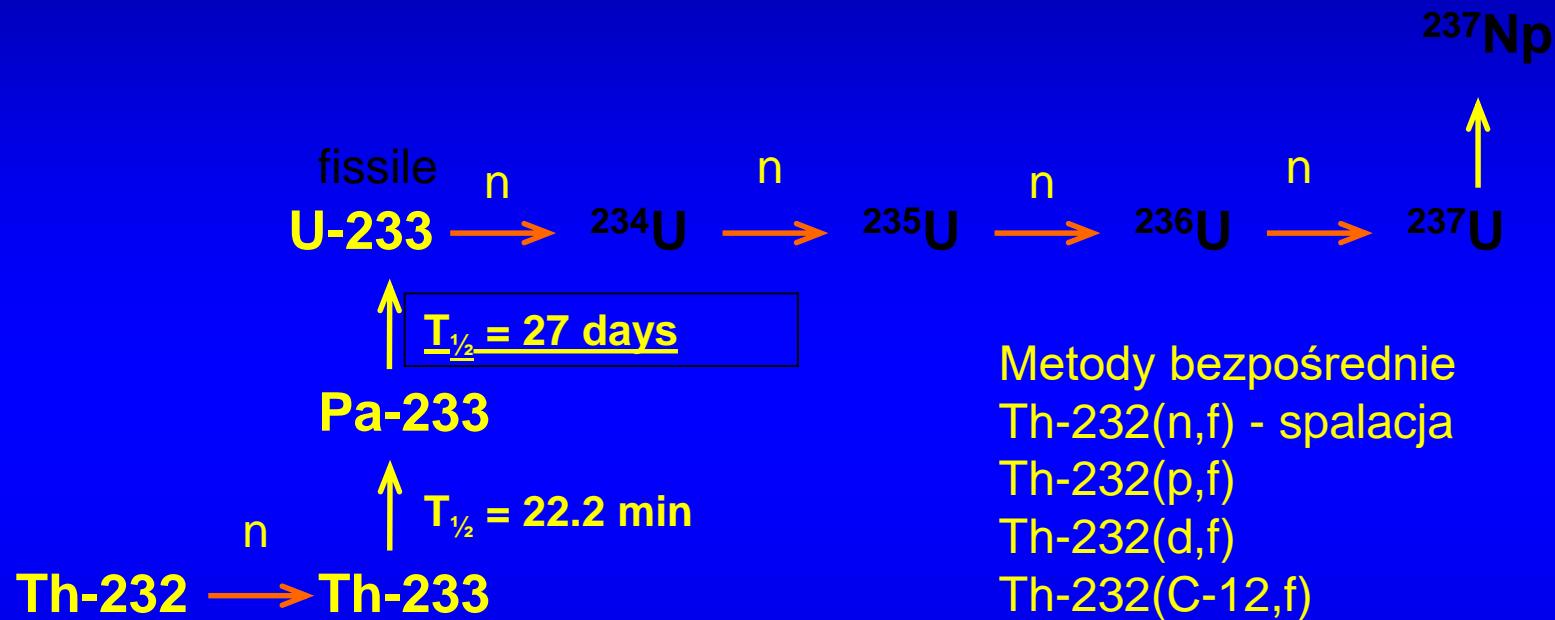


Łańcuch reakcji $U-238(n,g)+\beta+\beta$ to metoda pośrednia rozszczepienia $U-238$.
Jest podstawą breedingu uranowego.

Breeding jest wydajny w strumieniu neutronów reaktora przedkiego.
Jest tańszy od rozszczepiania w strumieniu neutronów spalacyjnych.
Breeding jest najbliższą przyszłością energetyki jądrowej.

Znane zasoby uranu wystarczą na 300 lat wykorzystując aktualną technologię – $U-235$.
Znane zasoby uranu wystarczą na 2000-3000 lat wykorzystując reaktory przedkie powielające.

Schemat wypalania Th-232 i produkcji U-233



Łańcuch reakcji $\text{Th-232}(n,g)+\beta+\beta$ jest pośrednia metodą rozszczepienia Th-232. Jest postawą breedingu torowego. Breeding torowy jest wydajniejszy niż uranowy. Może zachodzić wydajnie w strumieniu neutronów termicznych i prędkich. Jest tańszy od rozszczepiania w strumieniu neutronów spalacyjnych. Th-232 jest 3-4 razy więcej na Ziemi niż U-nat. Rezerwa energetyczna cywilizacji na ponad 10 000 lat.