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NUMERICAL SENSITIVITY STUDY OF IRRADIATED NUCLEAR FUEL EVOLUTION IN THE RBMK REACTOR

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The sensitivity study of the RBMK–1500 spent nuclear fuel (SNF) nuclide composition modelling was performed by investigating the essential physical reactor performance parameters (influence of coolant density, fuel and graphite temperatures, the fuel irradiation history, axial fuel assembly power profile) and specific model parameters depending on the computational code (model geometry description, neutron flux convergence criteria, evaluated nuclear data libraries and energy intervals used, resonance self-shielding parameters, etc.). SNF nuclide composition modelling was performed by using T–DEPL sequence from the SCALE 5 code package. The study showed that some of analysed parameters can influence the calculated SNF nuclide composition significantly, especially for minor actinides and fission products with high neutron capture cross-sections. It has been found that the coolant density and axial power profile have the largest influence on irradiated fuel inventory and in order to obtain the more precise RBMK–1500 SNF nuclide composition these parameters should be modelled as close to real conditions as possible. The lattice cell pitch used for resonance self-shielding can have a significant effect on calculated actinide activities. The correct parameter could be obtained from the experimental data that are presently unavailable.

Keywords: RBMK reactor, spent nuclear fuel, SCALE 5 code, modelling, nuclide composition, sensitivity analysis

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1. Introduction

Knowledge of the precise spent nuclear fuel (SNF) nuclide composition is needed for nuclear management, storage, partition, or disposal strategy. The majority of nuclear reactors in the world are light water reactors with a simple core structure – nuclear fuel surrounded by the moderator is arranged in a regular lattice, therefore almost all nuclear fuel evolution codes are designed for these reactor types. The heterogeneous RBMK reactor active core is composed of the irregular fuel element lattice situated in the coolant and surrounded by the graphite moderator. Current nuclear codes (for instance SCALE 5 [1] or multifunctional MCNP5+Monteburns [2]) allow describing the RBMK–1500 reactor construction with sufficient precision for the SNF nuclide composition evaluation. However, because of lack of experimental SNF nuclide composition measurements, the uncertainty and modelling sensitivity analysis should be performed before the model application in practice.

The modelled spent nuclear fuel nuclide composition mostly depends on the accuracy of neutron cross-section of interaction with matter. In SCALE 5 code NEWT module the neutron transport Boltzmann equation is solved analytically for 2D case. The more precise results can be obtained if 3D approach is taken, but this task cannot be solved analytically, and in this case a more time-consuming Monte Carlo method should be used [3–5].

The neutron spectrum in the reactor core also depends on the fuel and moderator temperature, coolant density, control and protection system rod position, the reactor power density profile, and other essential performance parameters.

This work deals with the sensitivity study of RBMK–1500 SNF nuclide composition modelling by using T–DEPL sequence from SCALE 5 code package. The applicability of this code specifically to the RBMK irradiated fuel nuclide composition calculation is discussed in [6–8]. The essential physical and computational code parameters were analysed and the conclusions for evaluation of the more precise RBMK–1500 SNF nuclide composition were drawn.

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2. Modelling tools and procedure

For modelling of the RBMK–1500 nuclear fuel nuclide composition the code package SCALE 5 was used. SCALE 5 (Standardized Computer Analyses for Licensing Evaluation) code system has the capability to perform depletion/decay calculation, criticality assessment, shielding, and heat transfer analyses using well established functional modules tailored to the SCALE system [1]. Spent fuel characteristics for these analyses can be obtained from a module that performs a depletion/decay calculation. SCALE 5 code sequence T–DEPL was used to model the RBMK–1500 nuclear fuel nuclide composition and radionuclide inventory.

T–DEPL is a 2D depletion sequence for characterization of spent nuclear fuel. As described before T–DEPL consists of TRITON module that couples NEWT code, which analytically solves neutron flux distribution problem for the 2D geometry and ORIGEN–S code. ORIGEN–S calculates the nuclide composition by solving a set of differential equations applying a matrix exponential expansion model. The SNF nuclide composition evolution is calculated using equation for the number of atoms N_i of each isotope i in the material which is affected by the neutron flux:

$$\frac{\mathrm{d}N_i}{\mathrm{d}t} = -\left[\lambda_i + \int \varphi(E,t)\sigma_i(E)\,\mathrm{d}E\right]N_i + \sum_{j\neq i}\left[\lambda_{ji} + \int \varphi(E,t)\sigma_{ji}(E)\,\mathrm{d}E\right]N_j\,, \quad (1)$$

where λ_i is the decay constant of ith isotope, λ_{ji} is the partial decay constant of jth isotope to ith isotope, $\varphi(E,t)$ is the magnitude of the particle flux of energy E, σ_i is the neutron absorption cross-section of ith isotope, and σ_{ji} is the neutron cross-section for transmutation from isotope j to i, N_j is the number of atoms of isotope j.

3. RBMK SNF assembly calculation model and modelling conditions

The RBMK–1500 reactor core consists of different fuel assemblies (FA), having 2% ^{235}U enrichment uranium dioxide fuel or 2.4, 2.6, and 2.8% enrichment uranium–erbium fuel, control rods (graphite displacer, water column, absorber), and some other types of channels (water columns, axial detectors, cluster additional absorbers, etc.). This leads to a very heterogeneous core. The fuel pin and the graphite are cooled by water that can be considered as a moderator and as an absorber. The case of fuel assembly with 2.6% ^{235}U initial fuel enrichment was selected as it is the main fuel in the operating Ignalina Nuclear Power Plant (INPP) reactor Unit 2. The dependences on the following parameters – variable coolant density, fuel and graphite temperatures, fuel irradiation history, axial FA power profile, and model geometry description – were checked during calculation. The burnups from 0 to 30 MWd/kg(U) were calculated in order to assess the radionuclide inventory at different irradiation steps. For sensitivity study of the variable system, the radionuclide inventory was obtained for each reference point of the following parameters listed in Table 1. The radionuclide inventory for RBMK–1500 nuclear fuel consists of a set of eight calculations that include the standard radionuclide production for the stated conditions.

In the model approximation it was accepted that in one pin of FA there is 3.518 kg of UO_2. The 2.6% enrichment UO_2 fuel with 0.5% burnable erbium poison was modelled (weight fraction: ^{235}U 2.6%, ^{238}U 97.3681%, ^{234}U 0.0296%, ^{236}U 0.0023%). The density of the fuel was 9.316 g/cm^3. It was accepted that the central hole of the fuel pellet is filled with fuel and the active length of FA is 700 cm.

T–DEPL calculations were performed by simulating one fuel assembly using mirror boundary conditions. The constant fuel assembly power of 2.53 MW (which corresponds to 4200 MW of full reactor power) was used during all irradiation time (with the exception of the fuel irradiation history calculation case). The cross-sectional view of the RBMK fuel assembly modelled with inner and outer fuel element rings and surrounded by the graphite stack is presented in Fig. 1.

Short- and long-lived nuclides were included in the sensitivity study of RBMK–1500 SNF for the estimation of possible uncertainty of important isotopes for the reactor operation, assessment of beyond design basis accidents, and for decommissioning. The activity data for 50 actinides and 200 fission products in the irradiated RBMK–1500 fuel assembly under different conditions were calculated and compared, but here only the main actinides and fission products with significant uncertainties or some long-lived fission products are presented. All the data can be found in [9].

Table 1. The reference point conditions used in lattice code calculations for a fuel cell model of RBMK–1500 reactor.

Fuel burnup, MWd/kg(U)	0, 5, 10, 15, 20, 25, and 30
Coolant density, g/cm^3	0.2, 0.5, 0.75
Coolant temperature, K	557
Fuel temperature, K	1000, 1500
Graphite temperature, K	600, 750
Reactor irradiation history, number of annual shutdowns	0, 2, 3
Axial FA power profile	No profile, [11] profile, [12] profile
Lattice cell parameter	0.125, 0.375, 0.625

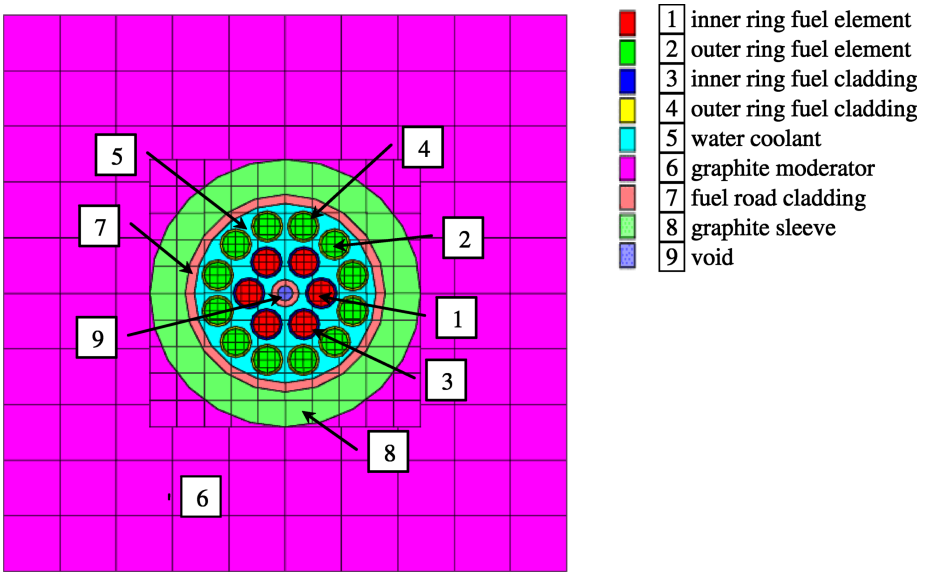

Fig. 1. The cross-sectional view of the RBMK fuel channel with assembly surrounded by the graphite sleeve and stack.

4. Results of sensitivity analysis of RBMK–1500 SNF nuclide composition modelling

4.1. Influence of the coolant density

The water density in the RBMK reactor core varies along the axial length of the fuel channel. It has influence on the neutron spectrum at the fuel location and hence on the burnup. According to the real situation in the reactor core, it is important to evaluate the effect of coolant density in the reasonable range from 0.2 to 0.75 g/cm^3. The coolant density in the RBMK–1500 reactor fuel channel is not constant and depends on the assembly power and coolant debit in the defined fuel channel region. The purpose of analysis of the coolant density variation influence on the nuclide composition is estimation of uncertainties determined by time-dependent fluctuations of the coolant density in a certain part of the fuel channel. The inner and outer rings of fuel pins of the RBMK fuel assembly have been considered, but the trend of variation of the nuclide composition in separate fuel pins is the same as in the full fuel assembly and in further description only the result for the fuel assembly is presented. The calculated data were compared to the reference case of 2.6% enrichment UO_2 fuel with erbium poison irradiated at the constant fuel assembly power. Averaged activities of actinides and fission products were obtained at a variable (0.2, 0.5, and 0.75 g/cm^3) coolant density, but at the constant temperature of the coolant, fuel, and graphite (see Table 1, for details). The relative 1δ uncertainty was calculated and results are presented in Table 2. The relative uncertainty δ was calculated as follows:

$$\delta = \sigma/\overline{A} = \sqrt{\frac{\sum\limits_i (A_i - \overline{A})^2}{(n-1)}}\,\overline{A}^{\,-1}\,, \qquad (2)$$

where σ is the standard deviation, $\overline{A}$ is the average of activities in variable density cases, A_i the activity of nuclide of interest, i is the coolant density case, n is the number of cases.

Out of actinides with a significant activity in the fuel assembly, ^{238}Pu, ^{241}Pu, ^{243}Am, ^{244}Cm are the most sensitive to coolant density variations. Their activities can differ by more than 20%. For other significant ac-

Table 2. Relative uncertainty (%) determined by different coolant density for 2.6% enrichment UO_2 fuel at different burnup.

Nuclide	Burnup, MWd/kg					
	5	10	15	20	25	30
^{238}Pu	12.3	11.6	10.9	10.0	9.2	8.5
^{239}Pu	1.5	0.8	0.9	1.2	1.3	1.4
^{240}Pu	5.9	4.7	4.2	3.8	3.5	3.4
^{241}Pu	13.3	7.3	4.6	3.3	2.7	2.3
^{242}Pu	16.4	10.2	6.4	4.1	2.5	1.4
^{241}Am	14.2	8.4	4.9	2.9	1.9	1.6
^{243}Am	29.8	23.1	18.2	14.5	11.1	8.2
^{244}Cm	24.7	18.0	13.5	10.4	8.0	6.3
All actinides	2.8	2.2	1.7	1.2	0.6	0.1
^{110m}Ag	14.2	12.9	11.4	9.8	8.3	6.8
^{114m}In	14.5	14.7	14.5	14.0	13.4	12.8
^{127}Xe	19.1	18.5	17.8	17.1	16.2	15.3
^{131}I	0.4	0.3	0.3	0.2	0.1	0.1
^{134}Cs	6.1	5.3	4.6	4.0	3.3	2.7
^{135}Cs	2.6	2.9	3.1	3.3	3.4	3.5
^{137}Cs	0.02	0.00	0.00	0.00	0.02	0.02
^{151}Sm	5.3	7.2	7.8	8.1	8.1	8.0
^{154}Eu	5.9	5.7	5.2	4.6	3.8	3.0
^{99}Tc	0.1	0.2	0.2	0.3	0.3	0.3
^{126}Sn	1.0	1.1	1.0	0.9	0.7	0.5
^{129}I	0.7	0.8	0.8	0.7	0.6	0.5
All fission products	0.2	0.1	0.1	0.2	0.3	0.3

Table 3. Difference (%) between actinide and fission product activities per fuel assembly determined by increased (1500 K) and reference (1000 K) fuel temperature.

Nuclide	Burnup, MWd/kg					
	5	10	15	20	25	30
^{238}Pu	2.0	2.0	2.0	2.0	2.0	2.0
^{239}Pu	2.3	2.4	2.5	2.7	2.8	2.9
^{240}Pu	1.7	1.2	0.7	0.6	0.3	0.3
^{241}Pu	2.7	3.1	3.2	2.9	2.7	2.5
^{242}Pu	2.4	2.5	2.3	1.9	1.3	0.8
^{241}Am	2.7	3.4	3.8	4.2	4.6	5.0
^{242m}Am	1.6	2.1	2.1	3.1	3.7	4.3
All actinides	2.1	1.8	1.5	1.0	0.6	0.1
^{99}Tc	−0.02	0.00	0.00	0.00	0.05	0.09
^{126}Sn	0.2	0.3	0.3	0.2	0.1	−0.1
^{129}I	0.2	0.3	0.4	0.4	0.4	0.4
^{134}Cs	−0.7	−0.7	−0.8	−0.8	−1.0	−1.1
^{135}Cs	0.2	0.4	0.5	0.7	0.8	0.9
^{137}Cs	−0.02	0.00	0.00	0.00	0.00	0.03
^{155}Eu	5.0	5.4	5.4	5.2	5.1	4.9
^{168}Tm	2.1	1.5	1.1	0.8	0.5	0.3
All fission products	0.1	0.1	0.1	0.1	0.1	0.1

tinides the difference is usually less than 10%. The uncertainties for the inventory of all actinides comprise a few percent (2.8%). Activities of short-lived fission products such as ^{110m}Ag ($T_{1/2}$ = 249.79 d), ^{114m}In ($T_{1/2}$ = 49.51 d), ^{127}Xe ($T_{1/2}$ = 36.34 d) differ by up to 20%, they are important for assessment of beyond design basis accidents for RMBK–1500 reactors. Activities of long-lived isotopes such as ^{151}Sm ($T_{1/2}$ = 60 y) and ^{154}Eu ($T_{1/2}$ = 8.5 y) differ by 3–8% during the burnup. Activities of other fission product were less sensitive (<1% for ^{129}I, ^{131}I, ^{99}Tc) to the water density in the fuel channel. Out of highly volatile isotopes with significant influence on doses during severe accidents, ^{134}Cs is the most sensitive (up to 6%) to coolant density variations. The uncertainty for the total fission product activity (0.3%) is by an order of magnitude less compared with the actinides contribution. Our results on uncertainty of total nuclide activities are in good agreement with results on the RBMK–1000 reactor, where spent nuclear fuel nuclide mass differences due to water density (0.25, 0.41, and 0.68 g/cm^3) comprise a few percent [10].

4.2. Influence of the fuel and moderator temperatures

The influence of increasing the fuel temperature to 1500 K was calculated and compared to the reference case of 1000 K temperature fuel (2.6% enrichment UO_2 with Er) irradiated under constant power conditions. The difference between nuclide activities was calculated in respect to a reference case:

$$\Delta = \frac{A_i - A_{\text{ref}}}{A_{\text{ref}}}, \tag{3}$$

where A_i is the activity of the nuclide of interest and A_{ref} is the activity of the nuclide of a reference case.

The difference (%) between actinide and fission product activities per fuel assembly is presented in Table 3.

The increased fuel temperature determines the broadening of actinide resonances. The activity of Pu isotopes increases by up to 3% (2% for ^{238}Pu, 2.3–2.9% for ^{239}Pu, 0.3–1.7% for ^{240}Pu, 2.5–3.2% for ^{241}Pu) and the amount of Am isotopes (^{241}Am, ^{242m}Am) increases by up to 5%. The differences for the inventory of all actinides comprise a few percent (2%). Inventory of majority fission products can be assumed as insensitive to the fuel temperature, with the exception of ^{155}Eu activity.

As graphite is used as a moderator in the RBMK–1500 type reactor, its temperature has direct influence

Table 4. Difference (%) between actinide and fission product activities per fuel assembly determined by decreased (600 K) graphite sleeve and reference (750 K) graphite temperature.

Nuclide	Burnup, MWd/kg					
	5	10	15	20	25	30
^{238}Pu	−0.6	−0.6	−0.6	−0.6	−0.7	−0.6
^{239}Pu	0.5	0.8	1.1	1.3	1.3	1.4
^{240}Pu	−1.3	−0.9	−0.7	−0.4	−0.3	−0.2
^{241}Pu	−1.6	−0.9	−0.7	−0.4	−0.2	0.0
^{242}Pu	−2.2	−1.8	−1.5	−1.2	−1.1	−0.9
^{243}Pu	−2.2	−1.8	−1.6	−1.3	−1.1	−0.9
^{242}Am	−2.6	−2.1	−1.7	−1.3	−0.9	−0.6
^{242m}Am	−2.9	−2.4	−2.0	−1.6	−1.2	−0.8
All actinides	0.0	0.1	0.1	0.1	0.1	0.1
^{134}Cs	0.1	0.0	0.1	0.1	0.0	0.0
^{135}Cs	−0.5	−0.5	−0.5	−0.5	−0.5	0.4
^{137}Cs	0.0	0.0	0.0	0.0	0.0	0.0
^{155}Eu	0.8	1.0	1.2	1.3	1.4	1.4
^{171}Tm	−1.1	−0.9	−0.7	−0.5	−0.4	−0.2
^{99}Tc	0.01	0.00	0.00	0.00	0.00	0.00
^{126}Sn	−0.15	−0.15	−0.12	−0.10	−0.07	0.00
^{129}I	−0.09	−0.12	−0.08	−0.06	−0.04	−0.02
All fission products	0.0	0.0	0.0	0.0	0.0	0.0

Table 5. Relative uncertainty (%) determined by irradiation history of fuel assembly for 2.6% enrichment UO_2 fuel at different burnup.

Nuclide	Burnup, MWd/kg					
	5	10	15	20	25	30
^{238}Pu	0.0	0.3	0.4	0.6	0.8	0.9
^{239}Pu	0.0	0.0	0.0	0.0	0.0	0.0
^{240}Pu	0.0	0.0	0.0	0.1	0.0	0.0
^{241}Pu	0.0	0.1	0.1	0.1	0.1	0.1
^{242}Pu	0.0	0.1	0.1	0.1	0.1	0.1
^{241}Am	0.0	4.1	5.5	5.5	5.5	5.5
^{242}Am	0.0	4.1	5.5	5.5	5.6	5.5
^{242m}Am	0.0	6.2	7.1	6.6	6.4	6.2
All actinides	0.0	0.0	0.0	0.0	0.0	0.0
^{95}Nb	0.0	4.8	5.0	5.0	5.0	5.1
^{134}Cs	0.0	0.6	1.0	1.2	1.5	1.7
^{135}Cs	0.0	0.2	0.1	0.2	0.1	0.1
^{137}Cs	0.0	0.1	0.1	0.2	0.2	0.3
^{144}Ce	0.0	3.0	3.5	4.5	5.0	5.6
^{152}Eu	0.0	9.6	6.5	6.8	6.3	6.1
^{99}Tc	0.00	0.00	0.00	0.00	0.00	0.00
^{126}Sn	0.00	0.01	0.01	0.00	0.00	0.00
^{129}I	0.00	0.00	0.01	0.01	0.01	0.01
All fission products	0.0	0.0	0.0	0.0	0.0	0.0

on the neutron spectrum and on macroscopic neutron cross-sections, correspondingly. The reference state was modelled assuming constant temperature of 750 K in the graphite stack and graphite sleeves. The sensitivity study was performed using lower temperature (600 K) of graphite sleeves. The results are presented in Table 4. As the sleeve volume is small compared with the volume of the stack, the influence is negligible for actinides and the fission products.

4.3. Influence of the fuel assembly irradiation history

Operation history might have significant influence on the burnup dependent nuclide concentrations. To evaluate the effect of fuel irradiation history, the fuel irradiation was modelled assuming a 60-day down time period after one year of irradiation. Additional cases have been calculated with 2 and 3 down time periods. The relative uncertainty was calculated according to Eq. (2). Results are presented in Table 5.

Out of actinides, Am isotopes (^{241}Am, ^{242}Am, ^{242m}Am) are the most sensitive to the down time period. Their activities can differ by up to 7%. For other actinides with significant inventory in the fuel assembly the influence is negligible. The activity of some fission products is slightly more sensitive to the fuel assembly power history, but their influence is insignificant concerning all short- and long-lived fission products.

4.4. Influence of the fuel assembly vertical power density profile

The RBMK–1500 reactor has an inhomogeneous axial power distribution profile. It is determined by changing the coolant density in the fuel channel (FC) and the resulting difference in the neutron flux and spectrum. Control rods are inserted from the upper part to compensate vertical inhomogeneity and they disturb the neutron flux. So inhomogeneous axial burnup profile should be taken into account when calculating the radionuclide inventory in the irradiated RBMK–1500 nuclear fuel.

To investigate the influence of the axial power distribution on the radionuclide inventory, a set of input files was created with the power level varying from 0.1 to 1.5 related to the reference set. Calculated activities $A_{i,j}$ in the FA of each isotope i were combined into the total activity A:

$$A = \frac{1}{l}\sum_{j=1}^{n} A_{i,j}\Delta l_j \,, \tag{4}$$

Table 6. Axial power distribution in the fuel channel.

H, m	Fuel assembly power, relative units [11]	H, m	Fuel assembly power, relative units [12]
0.25	0.512	0.4325	0.6145
0.75	0.750	1.3125	0.9847
1.25	0.992	2.1875	1.1999
1.75	1.155	3.0625	1.1966
2.25	1.249	3.9375	1.0869
2.75	1.177	4.8125	1.0966
3.25	1.091	5.6875	1.0478
3.75	1.116	6.5625	0.7728
4.25	1.155		
4.75	1.150		
5.25	1.128		
5.75	1.036		
6.25	0.858		
6.75	0.631		

where l is the total length of the fuel assembly, Δl_j is the partial length of the fuel assembly with a specific power level.

The INPP FC axial power distribution was taken from [11] and [12] according to the INPP Unit 2 measurements carried out in 2006. The data are presented in Table 6.

The difference of RBMK–1500 FA radionuclide inventory calculated with the axial power distribution taken from [11] and [12] compared with the flat averaged axial power profile is presented in Table 7. The difference Δ between nuclide activities was calculated according to Eq. (3)

For actinides the influence of the real power profile is ambivalent, i. e. some actinide activities are lower and some higher. The difference is less than 10% for significant actinides. Main exception is ^{244}Cm activity which differs by up to 46%. For the large burnups there is a substantial difference in ^{235}U amount. This effect is more important for the reactivity and SNF criticality safety than for the radiation safety during severe accidents. The differences for the inventory of all actinides make up a few percent. The fission product inventory calculated assuming a realistic power profile is mostly by a few percent higher compared with the reference case.

The axial power profile according to INPP Unit 2 measurements from [12] has less influence on the radionuclide inventory compared with that from [11].

Axial profile of the plutonium isotopic composition along the fuel assembly is presented in Fig. 2. One can see that ^{239}Pu isotope concentration is dominating at the ends of the fuel assembly but ^{238}Pu and ^{242}Pu isotopes accumulate in the middle of the fuel assembly in the higher power regions.

Table 7. Difference (%) between actinide activities per fuel assembly with the real axial power profile ([11] and [12]) compared with the averaged axial power profile case.

	Burnup, MWd/kg					
	10		20		30	
Nuclide	[11]	[12]	[11]	[12]	[11]	[12]
^{235}U	0.2	0.4	1.9	2.0	9.9	8.5
^{237}Np	1.3	0.5	0.7	0.0	−0.8	−1.1
^{238}Pu	7.5	4.8	6.9	4.2	3.6	1.8
^{239}Pu	−1.8	−1.5	−1.9	−1.3	−0.8	−0.5
^{240}Pu	1.2	0.3	−0.7	−1.0	−2.3	−2.1
^{241}Pu	2.5	1.1	−1.4	−1.6	−3.2	−2.7
^{242}Pu	14.0	9.4	10.0	6.3	6.7	4.2
^{241}Am	1.7	0.4	−4.6	−3.9	−7.2	−5.2
^{243}Am	28.0	20.0	21.0	14.7	15.0	10.0
^{242}Cm	13.0	8.8	6.0	3.4	−0.6	−1.4
^{244}Cm	46.0	33.9	41.0	29.7	36.0	25.7
All actinides	1.5	0.6	3.0	1.7	3.3	1.8
^{109}Pd	4.1	2.3	5.5	3.4	6.0	3.7
^{110m}Ag	11.6	7.7	12.7	8.5	13.3	9.0
^{124}Sb	7.6	4.8	9.9	6.5	11.2	7.4
^{123m}Te	12.8	8.6	17.1	11.8	19.3	13.4
^{127}Xe	18.7	13	23.1	16.3	26.3	18.7
^{134}Cs	5.9	3.6	6.3	3.9	6.0	3.6
^{135}Cs	−1.4	−1.0	−1.5	−1.2	−0.9	−0.6
^{137}Cs	0.5	−0.1	0.5	−0.1	0.5	−0.1
^{139}Ce	3.7	2.1	3.1	1.6	1.5	0.4
^{154}Eu	5.4	3.2	3.9	2.1	1.8	0.6
^{99}Tc	0.3	−0.2	0.0	−0.4	−0.3	−0.6
^{126}Sn	0.3	0.5	1.4	0.6	1.6	0.7
^{129}I	1.0	0.3	1.0	0.2	0.9	0.2
All fission products	0.4	−0.1	0.4	−0.2	0.4	−0.1

The axial profile of activities of cesium isotopes along the fuel assembly is presented in Fig. 3. The ^{137}Cs activity corresponds to the axial power profile and differs a few times compared with that at the ends and

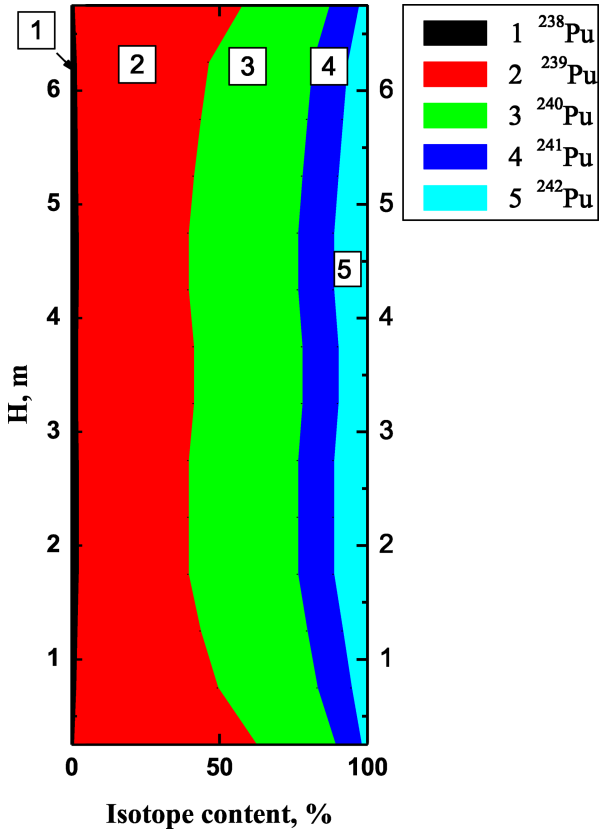

Fig. 2. Dependence of the plutonium isotopic composition on the axial power profile along the fuel assembly.

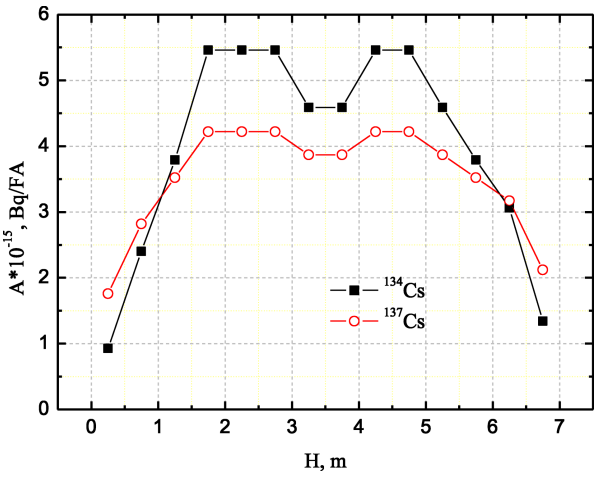

Fig. 3. ^{134}Cs and ^{137}Cs activity dependence on the axial power profile along the fuel assembly.

in the middle part of the fuel assembly. The activity of ^{134}Cs differs by up to 5 times along the fuel assembly vertical axis. The high energy gamma emitter ^{134}Cs is concentrated in the middle part of the fuel assembly.

Table 8. The dependence of major SNF isotope masses (M/M_{ref}) on the relative fuel assembly power (P/P_{ref}).

	Relative power				
Nuclide	0.8	0.9	1	1.1	1.2
^{239}Np	0.80	0.90	1.00	1.10	1.20
^{239}Pu	1.00	1.00	1.00	1.00	1.01
^{240}Pu	1.00	1.00	1.00	1.00	1.00
^{131}I	0.80	0.90	1.00	1.10	1.20
^{133}I	0.80	0.90	1.00	1.10	1.20
^{134}Cs	0.96	0.98	1.00	1.01	1.02
^{137}Cs	1.00	1.00	1.00	1.00	1.00

4.5. Influence of the fuel assembly power

The accumulation of nuclides in the irradiated nuclear fuel depends on the neutron flux and correspondingly on the fuel assembly power. The amount of some actinides and fission products produced under different power conditions was compared to the reference case of the fuel assembly irradiated at 2.53 MW power. One must note that comparisons were made for the same burnup points, so the cases with lower power had a longer irradiation time and vice versa.

The ratios of nuclide masses accumulated under different fuel assembly power conditions compared with reference case for fuel burnup of 15 MWd/kg(U) are presented in Table 8. There are no significant differences (less than percent) for long-lived nuclides. The ratio of amounts of short-lived nuclides (e. g. ^{131}I, ^{133}I) closely corresponds to the relative power of the fuel assembly. ^{134}Cs with the half-time $T_{1/2} = 2.06$ y similar to the irradiation time of the fuel assembly in the core (which depends on the power level and is from 1.51 to 2.27 years) is an intermediate case.

4.6. Sensitivity of the radionuclide inventory calculation on model parameters

In order to evaluate the effect of a mesh size of the RBMK reactor assembly model on the neutron flux, spectrum, and macroscopic cross-section calculations by TRITON (see above), the fuel assembly was modelled with different 2D grid size. The cases were calculated with 0.625 and 0.125 cm mesh. For the reference case the 0.375 cm mesh was used. The relative uncertainty was calculated according to Eq. (2). Results are presented in Table 9.

For actinide and fission product activities in the fuel assembly the influence of calculation model mesh sizes is less than 4%. The fuel assembly model mesh size should be taken into account as a possible source of

Table 9. Relative uncertainty (%) of nuclide activities per fuel assembly calculated assuming various calculation model mesh sizes.

Nuclide	Burnup, MWd/kg					
	5	10	15	20	25	30
^{238}Pu	1.0	1.0	0.9	0.9	0.8	0.7
^{239}Pu	0.2	0.3	0.4	0.5	0.5	0.5
^{240}Pu	0.0	0.1	0.2	0.3	0.3	0.3
^{241}Pu	0.3	0.1	0.2	0.2	0.2	0.2
^{242}Pu	0.5	0.2	0.2	0.2	0.2	0.2
^{243}Am	1.4	1.1	0.9	0.9	0.8	0.7
All actinides	0.7	1.2	1.5	1.8	1.9	1.7
^{110m}Ag	3.4	2.9	2.3	1.8	1.2	0.7
^{126}I	1.6	1.9	2.2	2.4	2.5	2.4
^{127}Xe	1.3	1.8	2.3	2.6	2.7	2.7
^{134}Cs	0.6	0.5	0.6	0.6	0.6	0.5
^{135}Cs	0.2	0.2	0.1	0.1	0.1	0.1
^{137}Cs	0.1	0.1	0.1	0.1	0.2	0.3
^{145}Pm	2.0	2.0	2.1	2.1	2.2	2.2
^{153}Gd	3.3	3.4	3.5	3.6	3.6	3.6
^{171}Er	2.0	2.5	2.9	3.2	3.5	3.4
^{171}Tm	1.3	1.3	1.4	1.6	1.7	1.8
^{87}Rb	0.04	0.08	0.13	0.14	0.15	0.15
^{99}Tc	0.03	0.07	0.09	0.12	0.15	0.14
^{126}Sn	0.03	0.08	0.11	0.11	0.17	0.17
^{129}I	0.05	0.09	0.13	0.15	0.16	0.18
All fission products	0.8	1.3	1.6	1.7	1.7	1.6

uncertainties in the TRITON depletion sequence. The relative uncertainties for inventory of all actinides and fission products comprise a few percent (up to 2%).

Two materials (graphite and water) are used as a neutron moderator of the RBMK reactor. However, only one material can be determined as a moderator in SCALE 5. Therefore, water is considered as a moderator, while the influence of graphite on resonant neutron absorption cross-sections is not evaluated. Therefore, approximation is not correct. For the investigation of the neutron resonance shielding, two different cases have been analysed. The reference case with the lattice pitch equal to 1.605 cm as in a real RBMK–1500 fuel assembly and case with the modified pitch equal to 2.5 cm were calculated. The correct resonance self-shielding estimation is more important for actinide inventory calculations due to more numerous resonances of heavy nuclei. The largest differences in the actinide inventory are of the order of 10–15%. The isotopic composition of actinides in the spent nuclear fuel is important because modelling results will be used to solve SNF disposal and recycling problems.

5. Conclusions

The sensitivity study of RBMK–1500 spent nuclear fuel nuclide composition modelling was performed by investigating the essential physical reactor performance parameters (influence of coolant density, fuel and graphite temperatures, the fuel irradiation history, axial fuel assembly power profile) and specific model parameters depending on the computational code (model geometry description, resonance self-shielding parameters, etc.).

Concerning the fission products accumulated in the fuel assembly, the coolant density and axial power profile have the largest influence on the considered physical reactor core parameters and fuel assembly conditions, therefore they should be modelled as close to real conditions as possible.

The inventory of some isotopes of transuranium elements significantly depends on the neutron spectrum in the fuel. The water density and graphite temperature have the largest influence on the neutron spectrum and thus on the SNF isotopic composition modelling. Of those, water density values are the least determined through the FA irradiation time and should be considered in estimating the total uncertainty. The lattice cell pitch used for resonance self-shielding could have a significant effect on calculated actinide activities. The correct parameter could be obtained from the experimental data that are presently unavailable.

References

[1] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, Version 5.1, Vols. 1–3, (ORNL/TM-2005/39, 2006).

[2] X–5 Monte Carlo Team, *MCNP – A General N-Particle Transport Code*, Version 5, Vol. 1: Overview and Theory (LANL, 2003).

[3] H.R. Trellue and D.I. Poston, *User's Manual, Version 2.0 for Monteburns*, Version 5B, preprint LA-UR-99-4999 (LANL, 1999).

[4] R. Plukienė, A. Plukis, V. Remeikis, and D. Ridikas, Benchmark calculations of RBMK spent nuclear fuel isotopic composition using MCNP and ORIGEN codes, Lithuanian J. Phys. **45**, 281–288 (2005).

[5] D. Ancius, D. Ridikas, V. Remeikis, A. Plukis, R. Plukienė, and M. Cometto, Evaluation of the activity of irradiated graphite in the Ignalina Nuclear Power Plant RBMK–1500 reactor, Nukleonika **50**, 113–120 (2005).

[6] V. Remeikis and A. Jurkevicius, Evolution of the neutron sensor characteristics in the RBMK–1500 reactor neutron flux, Nucl. Eng. Des. **231**, 271–282 (2004).

[7] A. Plukis, V. Remeikis, L. Juodis, R. Plukienė, D. Lukauskas, and A. Gudelis, Analysis of nuclide content in INPP radioactive waste streams, Lithuanian J. Phys. **48**, 375–379 (2008).

[8] V. Remeikis, A. Plukis, L. Juodis, A. Gudelis, D. Lukauskas, R. Druteikienė, G. Lujanienė, B. Lukšienė, R. Plukienė, and G. Duškesas, Study of the nuclide inventory of operational radioactive waste for the RBMK–1500 reactor, Nucl. Eng. Des. **239**, 813–818 (2009).

[9] PHARE project No. 2003/5812.04.02 *Support to VATESI and its TSOs in Assessment of Beyond Design Basis Accidents for RMBK-1500 Reactors* Task 1: Assessment of Radionuclide Inventory in RBMK–1500 Irradiated Fuel. Report 1.5. *Sensitivity study of radionuclide inventory modelling in RBMK–1500 irradiated fuel* (Institute of Physics, Vilnius, 2006).

[10] A.S. Gerasimov, T.S. Zaritskaya, and A.P. Rudik, *Handbook of Nuclide Production in Nuclear Reactors* (Énergoatomizdat, Moscow, 1987) [in Russian].

[11] *INPP unit 2 safety analysis report, Accident analysis. Chapter 1. Methodology of accident analysis. Section 1.1. Database of initial data and engineering handbooks. Subsection 1.1.2. Engineering handbook for RELAP5 MCC model. Description of thermohydraulics components* (Code of Document PTOab2 – 0345 – 5112V1, Ignalina NPP, 2002).

[12] PHARE project No. 2003/5812.04.02 *Support to VATESI and its TSOs in Assessment of Beyond Design Basis Accidents for RMBK-1500 Reactors* Task 3: Analysis of RBMK–1500 Main Circulation Circuit Behaviour in Case of BDBA. Report No 2. *Features of the RBMK–1500 reactor main circulation circuit modeling*, Final Draft (Lithuanian Energy Institute, Kaunas, 2007).

RBMK REAKTORIAUS APŠVITINTO BRANDUOLINIO KURO EVOLIUCIJA: SKAITINIO MODELIAVIMO JAUTRUMAS PARAMETRAMS

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Santrauka

Panaudoto branduolinio kuro (PBK) sudėtį būtina žinoti pakankamu tikslumu, norint tinkamai parinkti jo saugojimo, galutinio laidojimo ar perdirbimo strategiją. Dauguma pasaulyje naudojamų reaktorių yra lengvojo vandens tipo su paprasta aktyviosios zonos sandara, kai branduolinis kuras išdėstytas taisyklinga gardele neutronų lėtiklyje, todėl dabartinės branduolinio kuro nuklidinės evoliucijos vertinimo programos pritaikytos būtent tokiems reaktoriams. RBMK reaktoriaus aktyviosios zonos konstrukcija yra heterogeninė, sudaryta iš kuro elementų, išdėstytų netaisyklinga gardele aušale ir apsuptų grafito lėtikliu. Egzistuojantys modeliavimo įrankiai (tokie kaip SCALE 5 [1] ar universali MCNP5+Monteburns [2, 3] sistema) leidžia pakankamai tiksliai aprašyti RBMK–1500 reaktoriaus konstrukciją nuklidinės sudėties evoliucijos vertinimui. Dėl eksperimentinių šio tipo reaktoriams branduolinio kuro sudėties matavimo duomenų stygiaus, atliekant PBK nuklidinės sudėties skaičiavimus, būtina įvertinti modeliavimo jautrumą bei neapibrėžtis.

Darbe sumodeliuota PBK sudėtis labiausiai priklauso nuo neutronų skerspjūvių vertinimo tikslumo. Naudojamoje SCALE 5 paprogramėje NEWT neutronų pernašos Boltzmann'o lygtis sprendžiama analiziniu būdu dvimačiam atvejui. Tikslesnius rezultatus gautume, jei spręstume trimatę lygtį; deja, toks uždavinys bendru atveju analiziniu būdu neišsprendžiamas, todėl tenka naudoti daug laiko užimantį Monte Karlo metodą. Neutronų spektras reaktoriaus aktyviojoje zonoje taip pat priklauso nuo branduolinio kuro ir lėtiklio temperatūrų, aušalo tankio, valdymo ir apsaugos sistemos strypų padėties, reaktoriaus galios laikinės priklausomybės ir kitų fizikinių parametrų. Tačiau optimaliam vertinimui reikalinga apibrėžti pakankamo tikslumo bei skaičiavimo laiko santykį. Šiame darbe RBMK–1500 PBK nuklidinės sudėties modeliavimo tikslumas, atsižvelgiant į svarbiausius fizikinius bei matematinio modelio parametrus, vertinamas pasitelkiant SCALE 5 programų paketo T–DEPL seką.

Atlikta RBMK PBK nuklidinės sudėties vertinimo analizė parodė, kad aušalo tankis bei išilginis galios pasiskirtymas rinklėje turi didžiausią įtaką nuklidų vertinimo tikslumui, ir šie fizikiniai parametrai modelyje turi būti kuo arčiau realybės.

Šalia fizikinių reaktoriaus parametrų neapibrėžties būtina atsižvelgti ir į paties matematinio modelio parametrus: modeliavimo gardelės žingsnį, neutronų srauto konvergencijos kriterijus bei parametrus modelio rezonansinei savajai sugerčiai įvertinti. Tai gali iki kelių kartų pakeisti apskaičiuotą nuklido kiekį panaudotame branduoliniame kure, ypač aukštesniesiems aktinoidams ir dalijimosi produktams su dideliais neutronų sugerties skerspjūviais.