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BURNUP CREDIT IN THE CRITICALITY SAFETY ANALYSIS OF SPENT FUEL IN THE RBMK-1000 STORAGE SYSTEMS

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Abstract
This paper describes how taking credit for burnup of actinides in the spent nuclear fuel (SNF) of RBMK-1000 was performed. The following characteristics were analyzed: initial fuel enrichment, burnup fraction, axial burnup profile in the fuel assembly (FA) and fuel weight. As the results show, in the first 400 hours after stopping the reactor, there is an increase in the effective neutron multiplication factor (k_{eff}) due to beta decay of ^{239}Np into ^{239}Pu. Further, from 5 to 50 years, there is a decrease in k_{eff} due to beta decay of ^{241}Pu into ^{241}Am. Beyond 50 years there is a slight change in the criticality of the system. Accounting for nuclear fuel burnup in the justification of nuclear safety of SNF systems will provide an opportunity to increase the volume of loaded fuel and thus significantly reduce technology costs of handling of SNF.

Keywords
spent nuclear fuel; RBMK-1000; nuclide composition; k_{eff}; modular code system SCALE

1.Introduction

In the analysis of nuclear safety of SNF storage systems, the existing regulations in Ukraine [1,2] require calculations subcriticality of such storages. If calculations are not done, it is necessary to consider that such systems are loaded with fuel of the maximum permissible reactivity during their design and construction.

Such a conservative approach in calculating the volume of SNF results in a significant increase in the geometric dimensions of the systems and, as a consequence, it increases the cost of handling and storing SNF.

In fact, after the fuel is removed from the reactor and its storage in the spent fuel pool, the k_{eff} of the fuel is significantly reduced due to the decay of fuel into actinides and fission products which absorb neutrons.

Criticality analysis using burnup credit requires two separate steps: evaluation of the concentration of nuclides in the SNF and calculation of k_{eff} using the concentration of nuclides which were obtained during the first step. In this paper nuclide composition were calculated by code SCALE 6.1 [3] code system and compared with the concentration of nuclides obtained by radiochemical analysis. In the criticality analysis using burnup credit, according to [4], only actinides (^{234}U, ^{235}U, ^{236}U ^{237}Np ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am ^{243}Am) were used in the calculations of criticality.

To assess the influence of various factors on the value of k_{eff} an analysis using a wide range of calculations for storage system with the SNF of RBMK reactors is described in this paper.

2. Initial Data

The RBMK-1000 system uses uranium oxide fuel and has a graphite-moderated water-cooled core. The water-cooled fuel assemblies are contained in vertical channels in the graphite moderator. The reactor core itself is cylindrical with a diameter of 11.8 m and a height of 7.0 m. The central portion of the core is fueled and the periphery acts as a reflector. Above and below the active core there is 0.5 m of graphite reflector. There are 2488 vertical columns of 25- by 25-cm graphite blocks. There are 1661 channels in the core that

contain fuel assemblies. The central (i.e., non-reflector) part of the core contains 1884 channels. The 1661 channels that are fueled are referred to as "technological channels," whereas the rest of the channels in the central part of the core are referred to as "control channels".

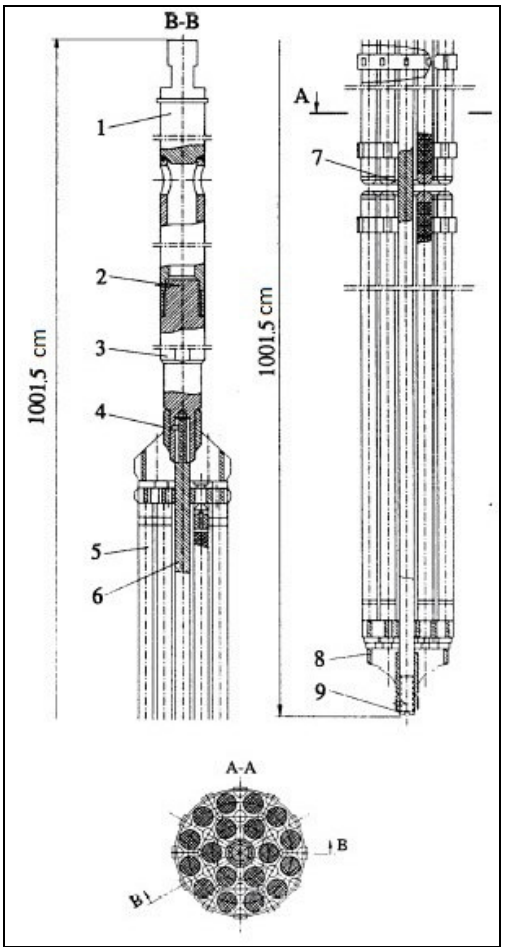

Fig.1. RBMK-1000 fuel assembly: 1 – suspension bracket; 2 – the upper plug; 3 – adapter; 4 – connector; 5 – fuel rod; 6 – carrier rod; 7 – trailer; 8 – the lower head; 9 – retaining nut.

A FA contains of two fuel parts each of them has 18 zirconium-alloy-clad UO_2 fuel rods. As can be seen in Figure 1, the fuel rods are located on two concentric rings and there is a central carrying rod. The light-water coolant enters at the bottom of the assembly and flows up through the assembly. It begins to boil at about 2.5 m from the bottom of the active fuel zone, and at the top of the active fuel zone the average steam quality is 14.5%. As the name implies, the RBMK-1000 is a 1000 MW(e) core and is rated as slightly greater than 3000 MW(t). Although most of the reactor's thermal energy originates in the fuel rods, about 6% is generated in the graphite moderator. When at normal operating power, the average temperature of the graphite is about 873 K [5].

Geometric and material parameters of spent fuel storage (SFS-1) and FA RBMK-1000, were taken as initial data for the calculation. FA are stored, or planned to be stored, in SFS-1.

SFS-1 consists of five spent fuel pools, each pool represent reinforced concrete tank, walls and floor are lined with stainless steel 12Kh18N10T (steel grade A 304 is an analog of this steel). The Spent Fuel Pool has slotted overlap for hanging SNF in canisters (Figure 2). Table 1 gives design parameters for the spent fuel pool of SFS-1.

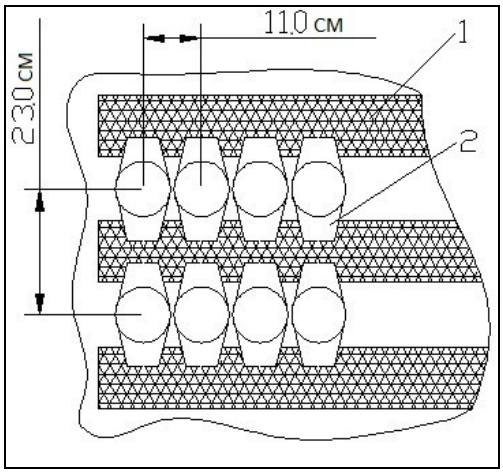

Fig. 2 Part of spent fuel pool of SFS-1 (top view): 1 – slotted overlap; 2 – canister.

Table 1. Design parameters for the spent fuel pool of SFS-1

Parameter	Value
Length, cm	2640
Width, cm	560
Depth, cm	1130
Capacity of the FA, pcs	4380
Water temperature, K	≤ 323
Material	12Kh18N10T (A 304)

3. Design Model of Spent Fuel Storage System

Calculations of criticality were performed by module KENO-VI [6] of the computer code SCALE 6.1, which is based on the use of the Monte Carlo method. In criticality calculations

a 238-group library of neutron-physical constants were used.

In developing the design model (Figures 3-4), the following assumptions and approaches were used: there is no gas gap between fuel and cladding in SNF, the diameter of fuel pellet is increased by a gas gap, and the fuel density decreased by 5% and accounted for 95% of the theoretical density of fuel (or 1% of the calculation).

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff} = k_{eff.calc} + K_c \cdot \sigma_c, \qquad (1)$$

where:

- k_{eff} is the maximum effective neutron multiplication factor;
- $k_{eff.calc}$ is the calculated effective neutron multiplication factor, under the worst combination of tolerances;
- K_c is the multiplier corresponding to the one-sided statistical tolerance limit.

Each final k_{eff} value calculated by SCALE-6 is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. For this analysis a value of 3.0 was assumed for the K multiplier, which is equivalent to 99% probability at the 99% confidence level.

σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code SCALE-6.
To calculate k_{eff} by KENO-VI next parameters were used:

- the numbers of neutron generation (GEN) equal 1000.
- neutrons per generation (NPG) equal 1050.
- neutrons generations skipped (NSK) equal 50.

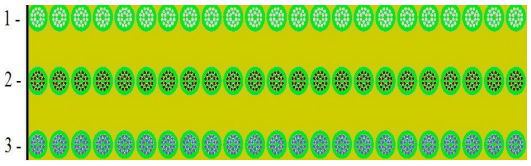

Fig. 3 Part of the design model of SFS-1. Initial enrichment of ^{235}U in SNF, %: 1-1.8; 2-2.0; 3-2.4

Figure 3 shows part of the design model of SFS-1 with three halves of rows. Various colors of spent fuel assemblies on the design model indicate different initial enrichment of SNF (1.8, 2.0 and 2.4 % of ^{235}U). The distance between rows is determined by the construction of the spent fuel pool of SFS-1 (Figure 2).

The nuclide composition was calculated using modules TRITON/T6-DEPL [0] and ORIGEN-ARP [0] of the SCALE-6 code system. Design model for RBMK-1000 FA shown in Figure 4.

The methodology used to build a model of a design basis assembly is to determine which assembly and plant operating parameters are conservatively bounding. In general, for source term calculations, assembly parameters that provide more mass are conservative. For example, the largest pellet diameter and clad thickness would be considered bounding. The longest active lengths and maximum assembly masses (for the UO_2) were also chosen to maximize masses. Uranium enrichments for RBMK-1000 fuel are 1.8%, 2.0%, 2.4%. Three design assembly models were built, one for 1.8%, 2.0% and 2.4%, which covers the entire range of enrichments.

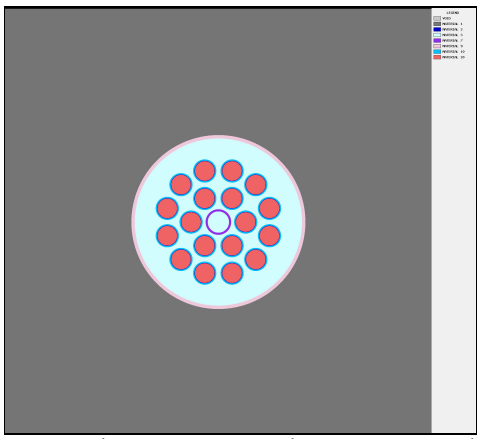
Fig. 4 Design model (top view) for calculating nuclide composition. Materials: 1 – graphite; 2 – helium; 3 – water; 7 – zirconium (Zr+1.25%Nb); 9 – stainless steel (12Kh18N10T); 10 - zirconium (Zr+1.00%Nb); 20 - UO_2

At the first stage, the design model was formed for the TRITON/T6-DEPL module. TRITON/T6-DEPL was used to calculate RBMK FA depletion at average power conditions and to generate burnup dependent cross section libraries. Depletion was performed by constant power and with use AddNux (value 2).

In the second stage the burnup dependent

libraries were used as input for ORIGEN-ARP to perform detailed depletion and decay calculations. These ORIGEN-ARP computations were used to create an extensive database of source terms for RBMK-1000 FA. The ORIGEN-ARP nuclide composition was used for calculations of burnup credit.

To ensure conservative results, the effect of tolerances in the manufacture of fuel and cladding of the fuel, initial fuel enrichment of ^{235}U and the initial mass of fuel in the FA were taken into account. The influence of these characteristics was considered in the creation of the design model for determining the nuclide composition and in the formation of the design model for the calculation of k_{eff}

4. Calculation and Analysis of Changes in Nuclide Composition of SNF

The contribution of each nuclide in the total mass of actinides is different. As can we see from Figure 5, the largest contribution falls on the nuclides, which are fissioned ^{235}U, ^{239}Pu, and ^{241}Pu. They account for about 60% of the total mass of actinides (excluding the mass of ^{238}U).

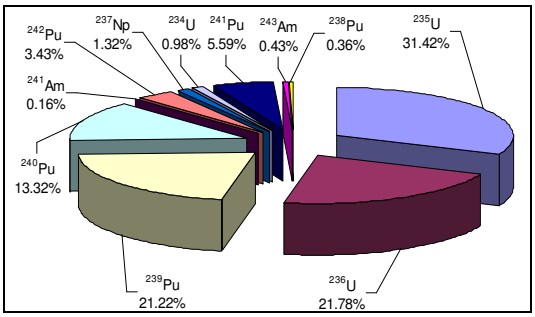

Fig. 5 The contribution of each nuclide, excluding ^{238}U, in the total mass of actinides in SNF, cooling time 1 year

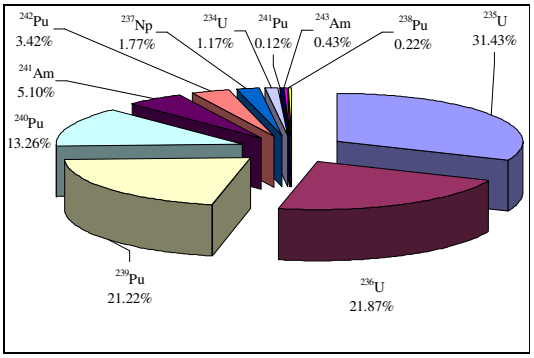

Fig. 5 The contribution of each nuclide, excluding ^{238}U, in the total mass of actinides in SNF, cooling time 60 years

^{241}Pu has a short half-life ($T_{1/2}$) 14.4 years and it is almost completely decayed into ^{241}Am after 60 years of cooling (Figures 6-7). It is the same for all thermal reactor FA.

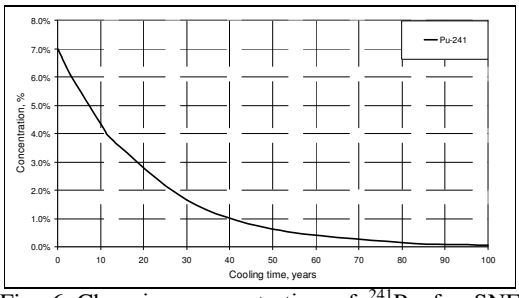

Fig. 6 Changing concentration of ^{241}Pu for SNF with initial enrichment 2.0% of ^{235}U and burnup 22.5 MWd/kg(U)

^{239}Pu has a $T_{1/2}$=24131 years, so a significant decrease in its concentration in the first 100 years does not occur. On the contrary, due to the decay of the ^{239}Np ($T_{1/2}$=2.355 days) in the first year we can see an increase in its concentration, as shown in Figure 7.

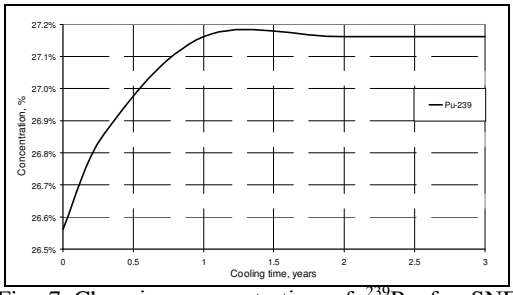

Fig. 7 Changing concentration of ^{239}Pu for SNF with initial enrichment 2.0% of ^{235}U and burnup 22.5 MWd/kg(U)

The calculated actinide nuclide composition was compared with that measured for spent-fuel samples from a group of RBMK-1000 assemblies [9]. Table 2 lists some details about the samples

Table 2. Design parameters for the spent fuel pool of SFS-1

Sample No	^{235}U Enrichment,%	Coolant density, g/cm^3	Burn, days	Power, MW/t	Burnup,GWd/t
1	1.8	0.45	1684	10.36	17.45
6	1.8	0.76	1685	13.42	22.62
9	1.8	0.50	1010	11.45	11.56
26	2	0.76	1281	16.69	21.38

Table 3. Measured and calculated results for 4 study samples (values are grams per gram of ^{238}U)

Isotope	Sample No 1			Sample No 6		
	Measured	Calc.	Calc/ Meas	Measured	Calc.	Calc/Meas
^{234}U	9.14E-05	1.17E-04	1.28	8.68E-05	1.04E-04	1.20
^{235}U	5.02E-03	4.86E-03	0.97	2.31E-03	2.75E-03	1.19
^{236}U	2.13E-03	2.25E-03	1.05	2.41E-03	2.51E-03	1.04
^{237}Np	1.06E-04	1.16E-04	1.09	1.26E-04	1.57E-04	1.25
^{238}Pu	3.01E-05	3.46E-05	1.15	4.95E-05	5.61E-05	1.13
^{239}Pu	2.70E-03	2.46E-03	0.91	2.42E-03	2.51E-03	1.04
^{240}Pu	1.70E-03	1.50E-03	0.88	2.10E-03	1.82E-03	0.87
^{241}Pu	5.27E-04	5.97E-04	1.13	6.04E-04	7.12E-04	1.18
^{242}Pu	2.29E-04	2.86E-04	1.25	4.49E-04	5.13E-04	1.14
^{241}Am	-	2.86E-05	-	-	2.96E-05	-
^{243}Am	1.55E-05	3.02E-05	1.95	4.83E-05	6.92E-05	1.43
Isotope	Sample No 9			Sample No 26		
	Measured	Calc.	Calc/Meas	Measured	Calc.	Calc/Meas
^{234}U	1.06E-04	1.33E-04	1.26	1.12E-04	1.23E-04	1.10
^{235}U	7.47E-03	8.03E-03	1.07	3.53E-03	4.10E-03	1.16
^{236}U	1.79E-03	1.76E-03	0.99	2.79E-03	2.66E-03	0.95
^{237}Np	9.64E-05	6.16E-05	0.64	1.91E-04	1.50E-04	0.78
^{238}Pu	1.31E-05	1.02E-05	0.78	4.61E-05	4.34E-05	0.94
^{239}Pu	2.35E-03	2.30E-03	0.98	2.43E-03	2.54E-03	1.05
^{240}Pu	1.20E-03	9.96E-04	0.83	1.98E-03	1.67E-03	0.84
^{241}Pu	3.23E-04	3.61E-04	1.12	5.98E-04	6.76E-04	1.13
^{242}Pu	8.83E-05	9.30E-05	1.05	3.82E-04	3.95E-04	1.04
^{241}Am	-	1.16E-05	-	-	2.28E-05	-
^{243}Am	4.82E-06	5.55E-06	1.15	3.76E-05	4.65E-05	1.23

A comparison of measured and calculated actinide nuclide composition is shown in Table 3. Figure 8 shows the relative error in determining the concentration of nuclides ^{235}U, ^{239}Pu, ^{241}Pu, which are major contributors to the k_{eff}. Relative error equals:

$$\frac{(\text{Measuredconcentraion} - \text{Calculatedconcentraion}) \cdot 100}{\text{Measuredconcentraion}}, \% \quad (2)$$

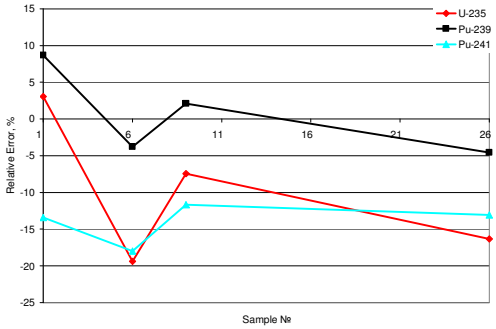

Fig. 8 Relative error in determining the concentration of nuclides

A comparison of the results for 4 nuclide assay samples indicates that the TRITON/T6-DEPL model for the RBMK is valid.

5.Calculation Results of k_{eff}

During operation in core, the burnup of the FA is uneven in height. This is due to the axial field profile of energy release. The effect of the distribution of burnup in FA at different heights at criticality of the system has been determined.

Calculating the burnup profile for RBMK-1000 FA was carried out taking into account the variation factor 1.2. The RBMK-1000 FA was divided into seven layers (Table 4). Values of burnup on each layer of FA are presented in Figure 9.

Figures 10-11 show the dependence of k_{eff} on the cooling time for the SNF with initial enrichment 1.8% and average burnup 15.0 MWd/kg(U).

Table 4. Burnup profile for RBMK-1000 FA, initial enrichment 2.4 % of ^{235}U

Layer number	Layer height, cm	Burnup, MWd/kg(U)
1	0 ÷ 70	20.0
2	70 ÷ 140	20.0
3	140 ÷ 210	22.5
4	210 ÷ 280	25.0
5	280 ÷ 350	27.5
6	350 ÷ 525	27.5
7	525 ÷ 700	22.5

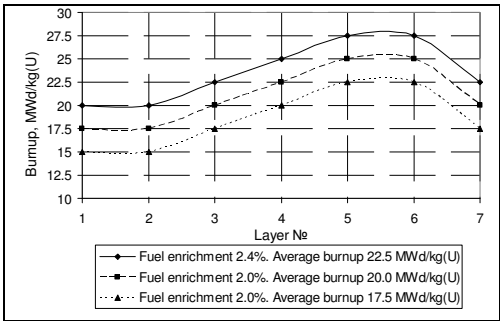

Fig. 9 Burnup profile for RBMK-1000 FA with different initial enrichment

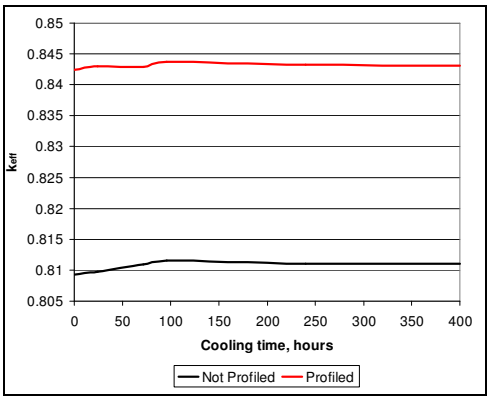

Fig. 10 Dependence of k_{eff} on the cooling time (hours)

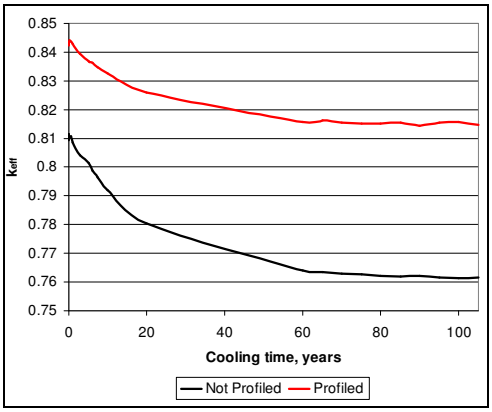

Fig. 11 Dependence of k_{eff} on the cooling time (years)

From these graphs we can see that it is very important to take into account burnup profile of SNF since values of k_{eff} are higher for FA with burnup profile (Table 4) than without profiling (in non profiled FA the average burnup was taken the entire height).

The above graphs show the same trend in k_{eff} depending on the cooling time. In the first 200 hours after stopping the reactor, there is an increase in k_{eff} due to beta decay of ^{239}Np into ^{239}Pu (Figure 10):

$$^{239}_{93}Np \xrightarrow[T_{1/2}=2.355d]{} ^{239}_{94}Pu \qquad (3)$$

Further, from 5 to 50 years, a decrease in k_{eff} due to beta decay of ^{241}Pu into ^{241}Am (Figure 11):

$$^{241}_{94}Pu \xrightarrow[T_{1/2}=14.4y]{} ^{241}_{95}Am \xrightarrow[T_{1/2}=432.2y]{} ^{237}_{93}Np \qquad (4)$$

In the next 50 years there is only a slight change in criticality of the system.

The regulation [0] says that k_{eff} must be less than 0.95. As written, this is a run on sentence Figure 12 shows the Δk_{eff}, difference between the regulation k_{eff} (k_{eff}=0.95) and the calculated k_{eff} for the SNF with initial enrichment 2.4% and different burnup.

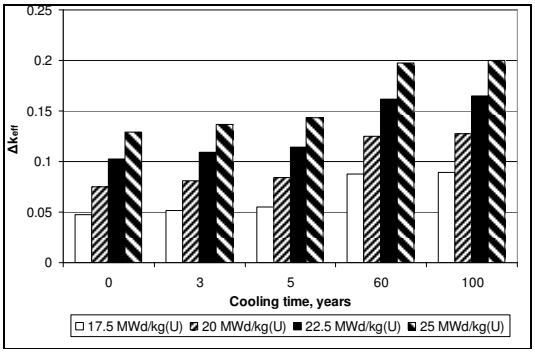

Fig. 12 Dependence of Δk_{eff} on the cooling time

As we can see from the Figure 12 the minimum difference between regulation k_{eff} and calculated k_{eff} is for first cooling year for FA with 17.5 MWd/kg(U) burnup (Δk_{eff} = 0.047).

Figure 13 shows the effect of tolerances in the manufacture of initial mass of fuel in the FA. The graph shows that in the preparation of a design scheme for the calculation of the isotopic composition more conservative would consider bigger value of mass since it increases k_{eff}.

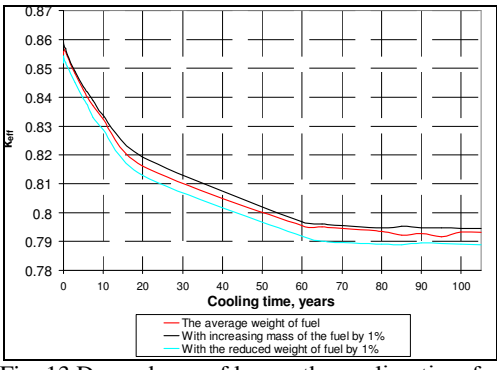

Fig. 13 Dependence of k_{eff} on the cooling time for the SNF with initial enrichment 2.4%

6. Conclusion

This paper analyzed some of the problems of nuclear safety justification of spent nuclear fuel storage systems using the burnup credit approach.

An account of the fuel burnup in the justification of nuclear safety of the spent nuclear fuel systems of an RBMK will provide an opportunity to increase the volume of loaded fuel and thus significantly reduce technology costs of handling of SNF. It is shown that it is very important to take into account the burnup profile of SNF to justify the nuclear safety of spent nuclear fuel storage systems using a burnup credit approach because, in this case, the values of k_{eff} are higher than without profiling. Not accounting for this distribution requires additional conservative assumptions in performing the calculations to justify the nuclear safety of spent nuclear fuel storage systems.

Also, it is very important in the preparation of input data to take into account tolerances and fits in the manufacture of FA.

For further work on the treatment of the burnup fuel for storage of RBMK fuel is important to obtain experimental data of the nuclide composition of SNF. The availability of such data would refine the methodology of calculation of the nuclide composition of SNF that would provide sufficient conservatism in the calculation of criticality using burnup credit.

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