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ASSESSMENT OF THE REACTIVITY BIAS AND BIAS UNCERTAINTY DUE TO WWER-440 FUEL DEPLETION UNCERTAINTIES

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In this paper, bias and bias uncertainty in the neutron multiplication factor due to biases and bias uncertainties in the calculated nuclide concentrations in the spent nuclear fuel of WWER-440 type was assessed. To determine isotopic biases and bias uncertainties in the calculated nuclide concentrations, they were compared to the results of the measurements of isotopic compositions from destructive radiochemical assay of WWER-440 spent fuel carried out in the RIAR. In calculations Monte Carlo uncertainty sampling method was used.

***Keywords*:** criticality safety, spent fuel, burnup credit, bias, bias uncertainty.

Introduction. One of the key milestones in implementation of the burnup credit approach is proper quantification and accounting of uncertainties in depletion analysis in terms of a reactivity difference. Uncertainty quantification involves two main stages [1]:

• Determination of the isotopic biases and bias uncertainties in the calculated nuclide concentrations by calculation of the ratios of calculated to measured nuclide concentrations from radiochemical assay of spent fuel.

• Determination of the bias and bias uncertainty in k_{eff} by applying the isotopic biases and bias uncertainties to the fuel compositions of representative safety analysis models, in this case, model of Armenian NPP (ANPP) spent fuel transport cask model. There are several methods for applying isotopic bias and bias uncertainties [1, 2]. In this work the Monte Carlo uncertainty sampling method was used.

Description of the WWER-440 Spent Fuel Assay Data. Destructive assay data of spent fuel assembly rod segments with an 38.5 $MW \cdot d/Kg$U burnup from a single 3.6% initially enriched VVER-440 fuel assembly from the Novovorenezh NPP (Russia) were used. Four rods from the fuel assembly were selected and removed from the assembly for further analysis. Furthermore, 8 sections were cut from the four fuel rods and sent for destructive analysis of radionuclides by radiochemical analyzes.

Fig. 1 demonstrates the location of fuel rods with the assigned numbers $1-126$ in analyzed fuel assembly. Fuel rods №№ 65, 67, 68, 69 selected to determine nuclide composition, isotopic mass and fuel burnup applying radiochemical and mass-spectrum analysis techniques, are marked in grey. Vertical arrangement of the fuel cuts is also shown in the Fig. 1. So, 3 cuts were taken from fuel rods №№ 65, 69, one cut from fuel rods №№ 67, 68. Selected cuts well represent all neutron characteristics of fuel assembly: axially and radially centered, intermediate and peripheral parts of the fuel rods.

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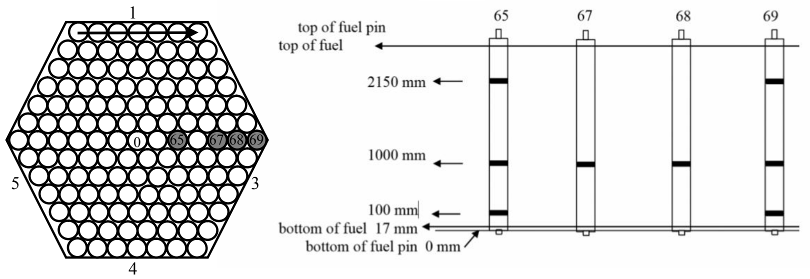

Fig. 1. Geometrical and vertical arrangement of fuel rods in fuel assembly.

Determination of the Isotopic Biases and Bias Uncertainties. The measured-to-calculated nuclide concentration ratio for n-th burnup credit isotope was calculated by following equation $R_n^j = (M_n^j)/(C_n^j)$, $j = 1, N_n$, where M_n^j is the measured concentration of nuclide n in the fuel sample j, the C_n^j is calculated concentration of nuclide n in the evaluated fuel sample, N_n is number of fuel samples containing n-th burnup credit isotope [1].

The sample mean and sample standard deviation are derived by following way:

$$\overline{R_n} = \sum_{j=1}^{N_n} \frac{R_n^j}{N_n}, \quad \sigma_n = \left(\sum_{j=1}^{N_n} \frac{\left(R_n^j - \overline{R_n}\right)^2}{(N_n - 1)} \right)^{1/2}.$$

Due to a limited number of measured fuel sample, tolerance intervals are used instead of confidence intervals in the sampling procedure to bound the uncertainty in a sample standard deviation. Following to the approach proposed in [1], tolerance interval was determined with tolerance limit factors for the normal distribution, which depend on the sample size, the specified proportion of the population within the bounds, and the specified certainty.

Since the number of WWER-440 fuel samples [3] used in this work is less than 10, therefore, uniform probability distribution was used in the Monte Carlo uncertainty sampling. Uniform probability distribution allows conservative sampling of a larger uncertainty range than a normal distribution.

The one-sided tolerance-limit factor for the normal distribution corresponding to the sample size, 95% certainty, and 95% of the population used as an adjustment factor to determine a uniform sampling interval. Since in this work number of measured samples is 8, adjustment factor of 3.031 [4] was applied to σ_n.

As the burnup of the most of fuel samples are within 30–47 $MW \cdot d/kg$U, single set of isotopic bias and bias uncertainty values was determined for each burnup credit isotope. In this work, in course of Monte Carlo uncertainty sampling, it was assumed that isotopic concentrations relevant to the burnup credit are independent variables. This assumption for WWER spent fuel needs further analysis to find out any correlations between different isotopes of uranium and plutonium since they are the most important contributors in the system reactivity. Isotopic compositions of the fuel samples were calculated by MCNP6.1 code [5]. The continuous energy cross-sections model was used to properly model neutron spectrum. ENDF/B-VII.1 [6] continuous energy cross-section library was used in the transport model. For adequate account of the thermal scattering effects of hydrogen and oxygen bound in the molecule of the water, appropriate S (α, β) scattering functions [7] were used. 3D depletion model of the spent WWER-440 fuel and operational history is based on [3] report. The isotopic vector that was used to quantify bias and bias uncertainty based on ISG-8 [8].

Calculated isotopic bias and bias uncertainty values for WWER-440 spent fuel based on 8 fuel samples [3], within burnup range of 22–47 $MW \cdot d/kg$U, are shown in the Table.

Isotopic bias and bias uncertainty values for WWER-440 spent fuel

Nuclide	Isotopic bias	Isotopic bias uncertainty
^{234}U	0.7246	0.1187
^{235}U	0.9901	0.0339
^{238}U	1.0017	0.0057
^{238}Pu	0.7576	0.0898
^{239}Pu	0.9804	0.0142
^{240}Pu	1.0109	0.0123
^{241}Pu	1.0307	0.0132
^{242}Pu	1.0204	0.0284
^{237}Np	1.9608	0.0178
^{241}Am	0.9947	0.0189

Quantification of the Reactivity Bias and Bias Uncertainty. The calculated nuclide concentrations in each axial node of the fuel in the ANPP spent fuel transport cask model were adjusted with taking into account for isotopic bias and bias uncertainty by following way, $C_n^k = C_n\left(\overline{R_n} + 3.034 \cdot \sigma_n \cdot r\right)$, where n is nuclide taken into account in burnup credit analysis; C_n^k is concentration of nuclide n in a fuel mixture for criticality calculation k; C_n is calculated concentration of nuclide n; $\overline{R_n}$ is isotopic bias; σ_n is isotopic bias uncertainty; r is random number sampled from the uniform distribution ranging from -1 to 1 (see [1]). Mean and standard deviation of the k_{eff} were calculated based on 500 MCNP 6.1 Monte Carlo calculations by following equations:

$$\overline{k_{eff}} = \sum_{i=1}^{N_c}(k_{eff}^i)/N_c, \quad \sigma_{k_{eff}} = \left(\sum_{i=1}^{N_c}\left(k_{eff}^i - \overline{k_{eff}}\right)^2/(N_c-1)\right)^{1/2},$$

where k_{eff}^i is k_{eff} value for criticality calculation i in the series of N_c criticality calculations, N_c is number of calculated k_{eff} values.

The radial cross-section of the the ANPP spent fuel transport cask model is shown in Fig. 2, a.

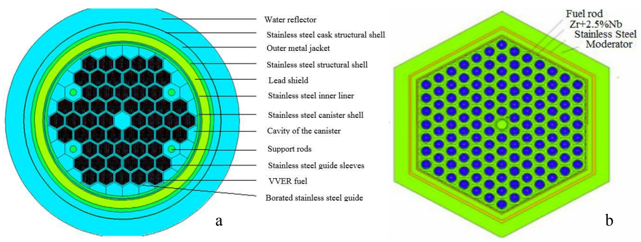

Fig. 2. a. Radial cross-section of spent fuel transport cask, b. Geometrical fuel units in spent fuel transport cask.

Spent fuel transport cask model consists of 56 identical geometrical units (see Fig. 2, a), which contains fuel assembly/follower model, canister basket, internal and external moderators. It contains also the 4 stainless steel tubes which are used to provide a mechanical strength to the canister. 56 spent fuel assemblies are surrounded by gamma (Pb) and neutron shielding (H_2O) layers and water reflector (Fig. 1).

Fuel assembly model consists of 126 fuel rods and its surrounding moderator and other sub-models (see Fig. 2, b).

Geometrical and material parameters applied in the model were taken from Safety Analysis Report of the ANPP Spent Fuel Storage. Fuel part of the model axially was subdivided into 41 nodes to properly cover spent fuel burnup axial profile. Selection of the 41 axial nodes based on 2 factors: 1) nodalisation is used in ANPP for core neutronics analysis, so ANPP actual axial profiles can be directly incorporated in the model; 2) it was shown that for PWR burnup credit analysis 18 axial nodes are enough to account peculiarities of axial burnup shapes for criticality applications [9]. To ensure proper sampling and source convergence as well as statistical reliability of calculated k_{eff} values the following Monte Carlo simulation parameters were used [10]: number of neutrons per generations are 100000; number of modeled neutron generations are 500; number of skipped neutron generations are 300. Source convergence also was controlled by convergence of Shannon entropy [11]. In all modeled cases, skipping of initial 300 neutron generations allowed to reach well-converged solutions. To properly model neutrons interactions with media at thermal energies thermal scattering $S(\alpha,\beta)$ functions [7] were assigned to the cross-sections of hydrogen and oxygen in the water as well as oxygen in UO_2.

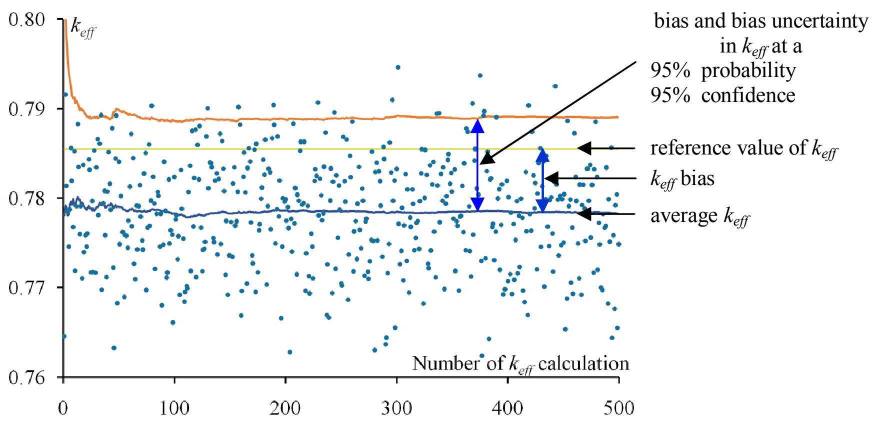

Fig. 3. Distribution of the k_{eff} values and their statistical estimates.

Bias in k_{eff} of ANPP spent fuel transport cask was calculated as $k_{eff}^{bias} = k_{eff}^{Ref} - \overline{k_{eff}}$, where k_{eff}^{Ref} is k_{eff} value of ANPP spent fuel transport cask with calculated nuclide concentrations without adjustments. Bias uncertainty in k_{eff} at a 95% probability, 95% confidence level is calculated by $k_{eff}^{bias\ uncertainty} = \sigma_{k_{eff}} \cdot tf_1^{N_c}$, where $tf_1^{N_c} = 1.763$ is the one-sided tolerance-limit factor for the normal distribution corresponding to the $N_c = 500$ calculated k_{eff} values, at a 95% probability, 95% confidence level.

The bias and bias uncertainty in k_{eff} due to biases and bias uncertainties in the calculated nuclide concentrations were calculated following way [1, 10]

$$k_{eff}^{bias} + k_{eff}^{bias\ uncertainty} = \begin{cases} \left(\overline{k_{eff}} - k_{eff}^{Ref}\right) + \sigma_{k_{eff}} \cdot tf_1^{N_c}, & \text{if } \overline{k_{eff}} > k_{eff}^{Ref}, \\ \sigma_{k_{eff}} \cdot tf_1^{N_c}, & \text{if } \overline{k_{eff}} \le k_{eff}^{Ref}. \end{cases}$$

The k_{eff} and $\sigma_{k_{eff}}$ values for 500 Monte Carlo calculations, the sample mean, $\overline{k_{eff}}$, the upper limit of the $95\%/95\%$ tolerance interval, and the bias and bias uncertainty in k_{eff} are shown in Figs. 3 and 4.

According to [1] k_{eff}^{bias} and $\sigma_{k_{eff}}$ are considered converged if their respective values change insignificantly (within $\pm 5 \cdot 10^{-4}$) with additional simulation. As we can see on Fig. 3

after 375 Mote Carlo k_{eff} calculations change of standard deviation is less than $\pm 4\cdot 10^{-5}$, so it could be considered well converged. After 500 calculations $\sigma_{k_{eff}}$ converges to the $\pm 6\cdot 10^{-3}$.

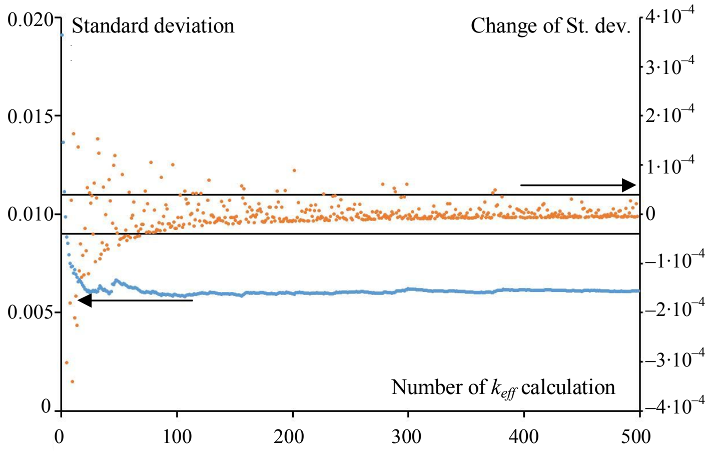

Fig. 4. Distribution of the $\sigma_{k_{eff}}$ and its change with additional simulation.

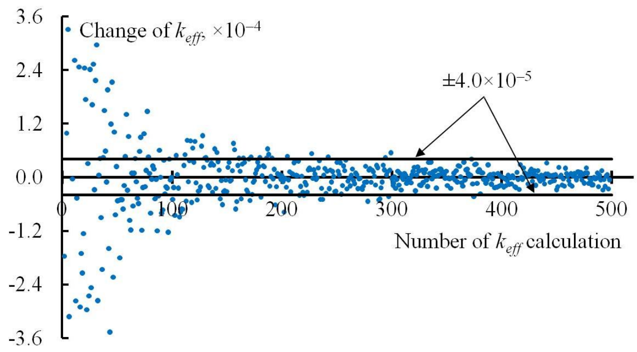

Fig. 5. Distribution of the change of k_{eff} with additional simulation.

On Fig. 5 change of k_{eff} with additional simulation is presented. After 375 Monte Carlo k_{eff} calculations change of k_{eff} with additional simulation is less than $\pm 4\cdot 10^{-5}$, so it could be considered well converged.

After 500 calculations average k_{eff} converges to the 0.77828. Taking into account that reference neutron multiplication factor is 0.78550 and $tf_1^{250} = 1.763$, the bias and bias uncertainty in k_{eff} for WWER-440 fuel with 3.6% enrichment and discharge burnup is: $k_{eff}^{bias} = 0.00722$ and $k_{eff}^{bias\ uncertainty} = 0.01075$ within 22–47 $MW\cdot d/kg$U. Usually in criticality safety analysis positive bias is not taken into account [10], therefore, 1.075% bias uncertainty should be applied to k_{eff} in burnup credit applications with actinides only option.

Conclusion. Bias and bias uncertainty in the neutron multiplication factor due to biases and bias uncertainties in the calculated nuclide concentrations in the spent nuclear fuel of WWER-440 type was assessed by using Monte Carlo uniform sampling procedure.

Further analysis is needed with involvement other WWER-440 spent fuel assay data to increase confidence on isotopic bias and bias uncertainties.

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