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THE FEW-GROUP DATA LIBRARY GENERATION FOR A VVER-1200 CORE USING THE MONTE CARLO CODE SERPENT

Abstract. This article presents the results of a comprehensive investigation focused on developing and validating a methodology for creating two-group cross-section libraries for the DYN3D reactor code in the formula format (IWQS = 2), utilizing the high-precision Monte Carlo code Serpent. The study specifically targets the VVER-1200 nuclear reactor at the Belarusian Nuclear Power Plant. The methodology involves calculating two-group cross-sections and other constants for various reactor states. It is demonstrated that the largest deviation for k_∞ for all types of fuel assemblies when calculated using the DYN3D code with the obtained library differs from precise Monte Carlo Serpent calculations by approximately 1000 ppm. This level of precision is deemed sufficient for safety analysis calculations using this library for the VVER-1200 reactor. Calculations of boric acid concentrations during the reactor's first and stationary fuel cycles were performed, along with the distribution of energy release at the beginning of the campaign. The obtained results underscore the reliability and effectiveness of the developed methodology.

Keywords: reactor safety analysis, accident analysis, acceptance criteria, computer simulation, diffusion approximation, two-group cross sections, Monte Carlo simulation

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ГЕНЕРАЦИЯ МАЛОГРУППОВОЙ БИБЛИОТЕКИ ДЛЯ АКТИВНОЙ ЗОНЫ ВВЭР-1200 С ИСПОЛЬЗОВАНИЕМ МОНТЕ-КАРЛО КОДА SERPENT

Аннотация. Представлены результаты комплексного исследования, направленного на разработку и валидацию методологии создания двухгрупповых библиотек сечений для реакторного кода DYN3D в формульном формате (IWQS = 2) с использованием прецизионного Монте-Карло кода Serpent. Исследование ориентировано на ядерный реактор ВВЭР-1200 Белорусской АЭС. Методика предполагает расчет двухгрупповых сечений и других констант для различных состояний реактора. Показано, что наибольшее отклонение для k_∞ у всех типов тепловыделяющих сборок (ТВС) при расчете по коду DYN3D с полученной библиотекой отличается от прецизионного расчета по Монте-Карло коду Serpent на величину порядка 1000 ppm. Этот уровень точности является достаточным для проведения расчетов с данной библиотекой по анализу безопасности работы реактора ВВЭР-1200 в различных условиях эксплуатации. Проведен расчет концентраций борной кислоты в процессе кампании для первой и стационарной топливных загрузок реактора и распределения энерговыделения в начале кампании. Полученные результаты подчеркивают надежность и эффективность разработанной методики.

Ключевые слова: анализ безопасности реактора, анализ аварий, критерии приемки, компьютерное моделирование, диффузионное приближение, двухгрупповые сечения, Монте-Карло моделирование

Для цитирования. Бабичев, Л. Ф. Генерация малогрупповой библиотеки для активной зоны ВВЭР-1200 с использованием Монте-Карло кода Serpent / Л. Ф. Бабичев, И. В. Руденков // Весці Нацыянальнай акадэміі навук Беларусі. Серыя фізіка-матэматычных навук. – 2026. – Т. 62, № 2. – С. 125–135. https://doi.org/10.29235/1561-2430-2026-62-2-125-135

Introduction. Safety analysis of nuclear reactors is of paramount importance, especially for reactors like the VVER-1200 at the Belarusian Nuclear Power Plant (NPP). The safety assessment of nuclear power plants[1] and accident modeling is often carried out using codes that employ a diffusion approximation for qualitative and quantitative safety parameter assessments. These codes typically utilize a nodal method to determine spatial zones, often requiring the preparation of constants in multiple groups that account for the specific behavior of fuel assemblies under various conditions (coolant temperature, boron concentration, etc.). The DYN3D reactor dynamics code [1; 2], developed at the Helmholtz –Zentrum Dresden – Rossendorf (HZDR), employs a two-group diffusion approximation to model reactor core behavior. This code can be used to conduct safety assessments and research in the field of neutron physics in VVER and PWR type reactors. In combination with thermohydraulic codes, it could be used to simulate reactor campaigns.

This study is dedicated to developing a methodology for creating a data library for the DYN3D code using the Monte Carlo code Serpent [3; 4]. Serpent is a multi-purpose three-dimensional continuous-energy neutron and photon transport code, developed at Technical Research Centre of Finland (VTT). The Serpent code enables high-precision modeling of neutron-physics characteristics of fuel assemblies, considering various changes in external parameters and fuel burnup. Its accuracy is limited only by the precision of the cross-sections in the libraries used by the Serpent code.

The aim of this work is to create a data library for the DYN3D code to conduct safety-related research at the Belarusian NPP. This approach is geared towards improving the accuracy by leveraging the precision of the Serpent code and expanding the capabilities of the DYN3D code for safety analysis under specific conditions of the VVER-1200 reactor at the Belarusian NPP.

Calculation of data libraries for DYN3D code using the Monte Carlo code Serpent. In the calculation of two-group constant libraries for large-scale reactor core simulation codes within the Serpent code [5; 6], two approaches for cross-section calculations and three options for diffusion coefficient calculations are employed. The first cross-section calculation method involves a direct approach (INF-P_1 approximation), while the second utilizes the critical spectrum in B_1 approximation. Three diffusion coefficient calculation options include transport correction (TRC), simple (calculation for an infinite lattice (INF)), and the use of the critical spectrum in B_1 approximation. The first option determines a transport correction by considering the anisotropy of scattering in the coolant (water), while the second one provides homogenized values in an infinite spectrum. The last option homogenizes cross-sections using the critical spectrum defined in B_1 approximation [7–13].

All combinations of these calculation methods were explored to identify the optimal approach for calculating cross-sections and diffusion coefficients in the Serpent code for the DYN3D code data library.

The selection of the best combination was based on a comparison of criticality coefficients and energy deposition distributions calculated in both codes. All calculations were performed under the following reactor core conditions: fuel temperature T_f = 900 K, coolant density ρ_W = 0.764 g/cm^3, coolant temperature T_m = 600 K, and coolant inlet pressure P = 15.7 MPa. Criticality coefficients and energy deposition distributions were calculated using both infinite (INF) and critical spectra in B_1 approximation.

For further calculations we use the INF-P_1 approximation for cross-sections and the TRC correction for the diffusion coefficient. This combination results in the best agreement between the DYN3D/Serpent code combination and the precision Serpent code.

To determine the reactor state in DYN3D, specific parameters must be set, including moderator density ρ_W, boron concentration C_b, fuel temperature T_f, moderator temperature T_m, and burnup B. Preceding the calculation of isotopic composition and burnup step files for each fuel assembly type, considering absorbing materials. These files include fission spectra, reaction rates, decay constants, delayed neutron fractions, xenon ^{135}Xe, samarium ^{149}Sm, iodine ^{135}I, and promethium ^{149}Pm cross-sections, and yields based on the specified parameters. These calculations are necessary for subsequent approximations based on existing dependencies for different types of libraries (IWQS = 2, 22, 24, 27).

[1] IAEA Safety Standards. Deterministic safety analysis for nuclear power plants. Specific safety guide No. SSG-2. Vienna, IAEA Publ., 2019. 108 p.

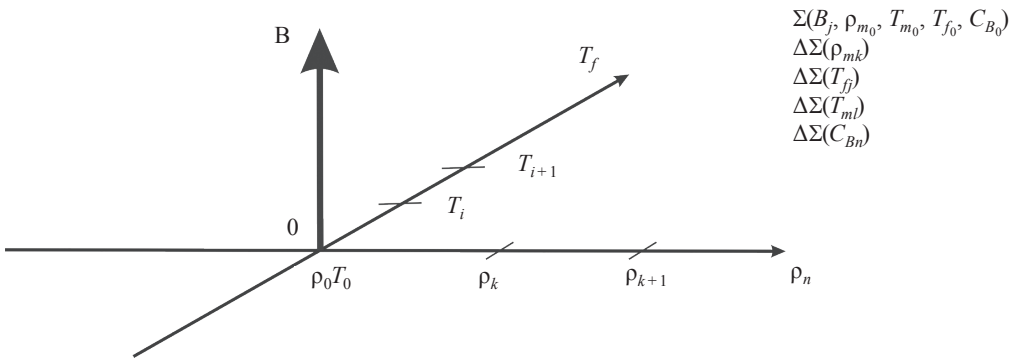


Fig. 1. Scheme for calculating two-group constants of fuel assemblies in the IWQS = 2 format

Subsequently, a data library file (for the formula format IWQS = 2) is computed based on user-defined parameters using two-group energy approximation:

1) two-group transport cross-sections $\Sigma_{tr,1}$, $\Sigma_{tr,2}$;
2) absorption cross-sections $\Sigma_{a,1}$, $\Sigma_{a,2}$;
3) generation cross-sections $\nu\Sigma_{f,1}$, $\nu\Sigma_{f,2}$;
4) energy release cross-sections $\varepsilon\Sigma_{f,1}$, $\varepsilon\Sigma_{f,2}$;
5) scattering cross-sections $\Sigma^{1\to1}$, $\Sigma_{1\to2}$, $\Sigma_{2\to1}$, $\Sigma_{2\to2}$;
6) discontinuity factors for fuel assemblies and hexagonal reflector cells;
7) two-group microscopic absorption cross-sections for xenon and samarium, xenon yields, iodine, and promethium (only for burnup).

After calculating all arrays for each burnup point and forming data arrays, values independent of other parameters are added to the library:

1) fission spectrum in two groups χ_1, χ_2;
2) inverse mean velocities $1/V_1$, $1/V_2$;
3) decay constants of the delayed neutron precursors $\lambda_1,\ldots,\lambda_6$;
4) fractions of the delayed neutron groups $\beta_1,\ldots,\beta_6$.

Creating a constant library for the DYN3D code using the Serpent code necessitates developing the corresponding models for all types of fuel assemblies (FAs) and reflectors used in accident scenario simulations. The calculated constant results should be appropriately processed to obtain a constant library for further DYN3D calculations. A Python script has been developed for interpolating cross-sections for the library in the IWQS = 2 format and generating the corresponding constant library.

Figure 1 illustrates the calculation scheme for creating a data library in the formula format. This scheme is based on representing the dependency of cross-sections as a system of equations (1)–(11).

The general function embedded in the formula format of the IWQS = 2 library describes the dependency of nine cross-sections on four reactor parameters:

$$\Sigma(C_b,\rho,T_f,T_m)=\Sigma_0 F_1(T_m)F_2(\rho_m)F_3(C_b)F_4(T_f), \tag{1}$$

$$F_1(T_m)=1+\alpha\left(\frac{1}{\sqrt{T_m}}-\frac{1}{\sqrt{T_{m0}}}\right), \tag{2}$$

$$F_2(\rho_m)=1+\beta_1(\rho_m-\rho_{m0})+\beta_2(\rho_m-\rho_{m0})^2, \tag{3}$$

$$F_3(\rho_m)=1+\delta_1(C_B\rho_m-C_{B0}\rho_{m0})+\delta_2(C_B\rho_m-C_{B0}\rho_{m0})^2, \tag{4}$$

$$F_4(T_f)=\exp\left(\gamma\sqrt{T_f}-\sqrt{T_m}\right). \tag{5}$$

When introducing burnup into the model, all coefficients α, β_1, β_2,… are determined by a second-degree polynomial of burnup:

$$\alpha(B) = \alpha_0 + \alpha_1 B + \alpha_2 B^2 . \tag{6}$$

In this case, the library function factors in the IWQS = 2 format take the form

$$F_1(T_m, B) = 1 + \left(\alpha_0 + \alpha_1 B + \alpha_2 B^2\right)\left(\frac{1}{\sqrt{T_m}} - \frac{1}{\sqrt{T_{m0}}}\right), \tag{7}$$

$$F_2(\rho_m, B) = 1 + \left(\beta_{10} + \beta_{11} B + \beta_{12} B^2\right)(\rho_m - \rho_{m0}) + \left(\beta_{20} + \beta_{21} B + \beta_{22} B^2\right)(\rho_m - \rho_{m0})^2 , \tag{8}$$

$$F_3(\rho_m, B) = 1 + \left(\delta_{10} + \delta_{11} B + \delta_{12} B^2\right)\left(C_B \rho_m - C_{B0} \rho_{m0}\right) + \\ + \left(\delta_{10} + \delta_{11} B + \delta_{12} B^2\right)\left(C_B \rho_m - C_{B0} \rho_{m0}\right)^2 , \tag{9}$$

$$F_4(T_f, B) = \exp\left[\left(\gamma_{10} + \gamma_{11} B + \gamma_{12} B^2\right)\left(\sqrt{T_f} - \sqrt{T_m}\right)\right]. \tag{10}$$

In this case, the general function

$$\Sigma(C_b, \rho_m, T_f, T_m, B) = \Sigma_0 F_1(T_m, B) F_2(\rho_m, B) F_3(C_b, B) F_4(T_f, B) \tag{11}$$

is a function of five variables with 18 free parameters, where Σ is the calculated cross-section, Σ_0 is the same cross-section calculated at the reference point $(C_{B0}, T_{f0}, T_{m0}, \rho_{m0})$. The variables C_B, T_f, T_m, ρ_m, represent current thermophysical parameters (boric acid concentration C_b in the coolant (H_3BO_3 /H_2O_3, g/kg), fuel temperature (T_f, K), moderator temperature (T_m, K), and coolant density (ρ_m, g/cm^3), respectively), α, β_1, β_2, δ_1, δ_2 and g are parameterization coefficients.

In addition to fuel assembly models, it was necessary to create corresponding reflector models to account for their impact on the reactor core. The axial reflector model is illustrated in Fig. 2, *a*. The upper and lower axial reflectors consist of three homogenized layers. They were modeled as two hexagonal nodes on both sides of the 3D model of any fuel assembly. Periodic lateral and axial boundary conditions were applied. The material composition of each layer was calculated for each type of fuel assemblies.

Radial reflector models for the DYN3D code are shown in Fig. 2, *b*, *c*. Radial reflectors were modeled in Serpent and validated by the MCU-PD codes as hexagons equal in size to the FA, containing the actual geometry of the grid and the reactor's inner core VVER-1200 shaft [14; 15]. Considering 30-degree symmetry of the grid for the first reflective layer allows for the simulation of 5 types of radial reflectors (Fig. 2, *a*). In addition to these, additional 5 types of models were used for the second layer of the radial reflector. The second layer enables the use of black boundary conditions in the full reactor core model in the DYN3D code, replacing a set of albedo coefficients, which may also depend on parameters. The calculations demonstrated that modeling the third layer of the radial reflector is not essential.

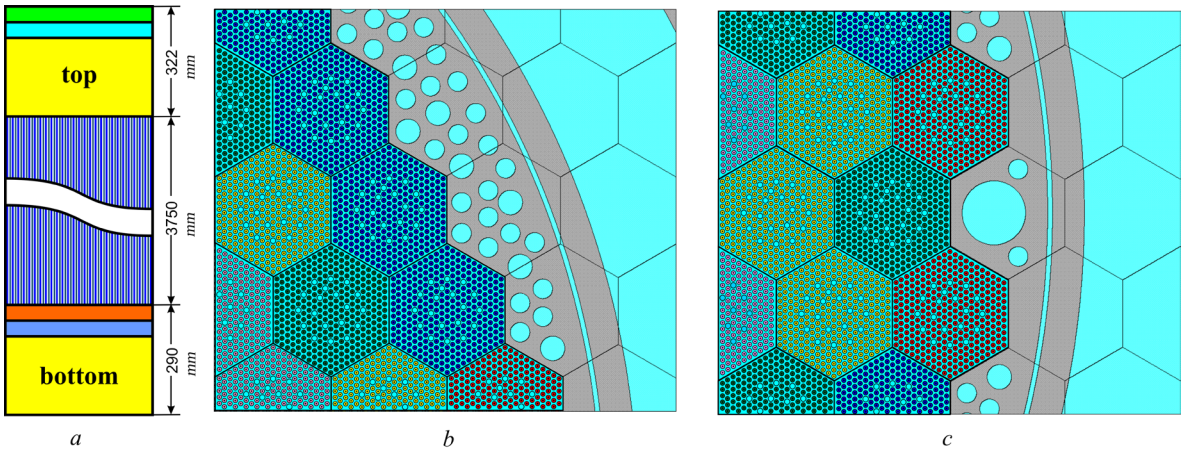


Fig. 2. Models of the upper and lower axial (*a*) and various types of radial (*b*, *c*) reflector cells of the VVER-1200 reactor in the Serpent code used to calculate the parameters of the reflector cells for the DYN3D code. Height in (*a*) is in millimeters

A detailed description of the fuel assemblies of the VVER-1200 reactor is provided in reports[1] and in works [16–20]. Key modifications compared to VVER-1000 include the elongation of the fuel column from 353 to 373 cm and optimization of the dimensions of fuel pellets and cladding to increase the loading of uranium dioxide. The central hole in the fuel pellets was justified to be reduced to 1.2 mm. These enhancements allowed for an over 8 % increase in fuel loading, with the maximum burnup reaching 70 MW · day/kgU.

The initial and stationary loadings for the base four-year fuel cycle were formed using various types of fuel assemblies. The types of fuel assemblies are described in Table 1.

Table 1. **VVER-1200 fuel assembly types**

Average Enrichment of ^{235}U, wt.%	Enrichment of Fuel Rods in ^{235}U, wt.%	Enrichment of Burnable Absorbers in ^{235}U, wt.%	Number of Fuel Rods (Burnable Absorbers)	Gd_2O_3, wt.%
1.30	1.30	–	312	–
2.40	2.40	–	312	
3.27	3.30	2.40	300 (12)	8
3.27	3.30	2.40	303 (9)	8
4.00	4.00	–	312	–
3.97	4.00	3.30	300 (12)	5
4.39	4.40	3.60	306 (6)	5
4.37	4.40	3.60	300 (12)	5
4.95	4.95	–	312	–
4.90	4.95	3.60	300 (12)	5
4.92	4.95	3.60	306 (6)	5

As an example, Fig. 3 shows two-dimensional models of individual fuel assemblies with reflective boundary conditions developed in the Serpent code for the first fuel loading. Different colors indicate fuel elements with different UO_2 enrichment, and gray areas (Fig. 3, *b*) indicate fuel elements with burnable absorbers ($UO_2 + Gd_2O_3$). The model also includes guide channels for control rods and a center hole for instrument controls. In this fuel assembly model, spacer grids are taken into account as an additional thickness of the zirconium cladding of the fuel elements. In the fuel assemblies of the VVER-1200 project, there is a lack of symmetry with respect to 60-degree rotations. This asymmetry arises due to the displacement of instrumentation channel from the center, in contrast to the fuel assemblies designed for the VVER-1000 reactor.

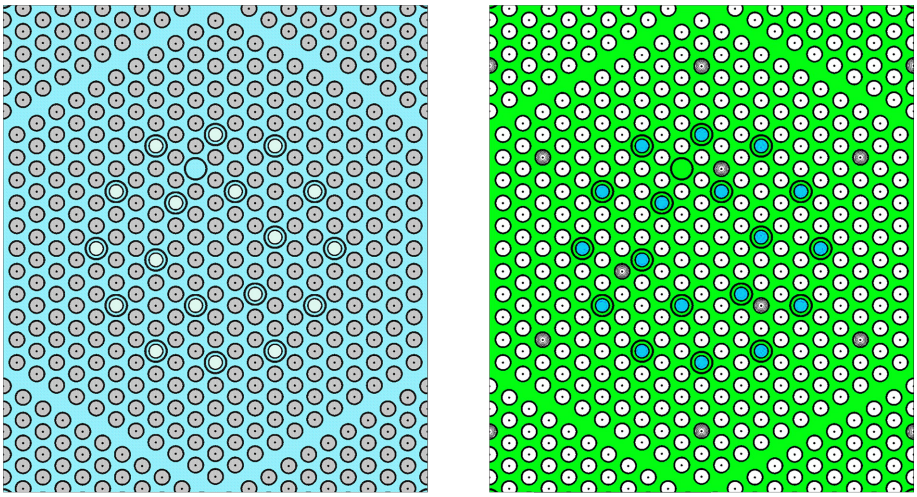
In the VVER-1200 reactor, absorber rods consist of two materials: the upper part is the boron carbide (B_4C), and the lower part is the dysprosium oxide-titanate dispersion $Dy_2O_3TiO_2$.

When calculating two-group cross-sections, FA without absorber rods (the corresponding space is filled with the coolant) and with each type of absorber rods placed in the guide channels are treated as separate FA models.

To construct a data library in the IWQS = 2 format, it is essential to choose an appropriate reference point and a set of points along the axes of variables corresponding to the physical parameters of the reactor state. The reference point is selected to represent the "average" reactor state, while other points of physical parameters are chosen to cover the entire range of reactor states. The results are presented in Tabl. 2 and 3.

The range of design points covers the range of conditions that the reactor may experience. The tables provide a complete set of states for data library construction in the IWQS = 2 format, including changes in burnup and physical parameters.

[1] IAEA: Status report for advanced nuclear reactor designs – Report 107. VVER-1200 (V-392M). Vienna, International Atomic Energy Agency, 2011. 32 p. URL: https: // aris.iaea.org / PDF / VVER-1200 (V-392M).pdf (date of access 10 January 2023); IAEA: Status report for advanced nuclear reactor designs – Report 108. VVER-1200 (V-491). Vienna, International Atomic Energy Agency, 2011, 32 p. URL: https: // aris.iaea.org / PDF / VVER-1200 (V-491).pdf (date of access 10 January 2023); Tran Vinh Thanh, Tran Viet Phu, Ta Duy Long, Nguyen Thi Dung. Developing the core loading pattern for the VVER-1200/V491. The Annual Report for 2016, VINATOM-AR 16–02, pp. 23–30. URL: https://inis.iaea.org/collection/NCLCollectionStore/_Public/49/044/49044527.pdf?r=1 (date of access 10 January 2023).



a *b*

Fig. 3. Two-dimensional models of individual types of fuel assemblies for calculating few-group cross-sections in the Serpent code: *a* is 2.4 % ^{235}U no Gd; *b* is 3.3 % ^{235}U with 12 Gd rods

Table 2. **Reference Point**

ρ_{m0}, g/cm^3	C_{b0}, g/kg	T_{m0}, K	T_{f0}, K
0.700	0.0	587	900

Table 3. **Calculation Range of Constants**

B, MW · day/kgU	ρ_m, g/cm^3	C_b, g/kg	T_f, K	T_m, K
0.0	0.810	0	533	533.15
3.0	0.766	1	587	587
6.0	0.722	2	1081.18	618.15
Step: B = 3 MW · day/kgU	0.700	4	1629.15	–
	0.678	6	2177.15	–
	0.634	8	2725.15	–
	0.590	10	3273.15	–
…	Total: 21 states for one burnup state			
70.0				

Verification of two-group cross-section library through $k_{\text{eff}}^{\text{inf}}$ comparison for assemblies calculated in Serpent and DYN3D. For convenient data processing of each fuel assembly simulated in the Serpent code, an automated data processing script was developed using the Python programming language. This script streamlines the data processing by employing various approximation methods to generate a file containing all the required parameters and cross-sections for each fuel assembly. These values are essential for the construction of the DYN3D data library.

After calculation by Serpent for each fuel assembly, the program reads the output file and prepares constants for the data library in tabular form. Then the program calculates the values of the coefficients for functions (7)–(11) so that the resulting function passes through the reference points as closely as possible. The use of automatic calculation ensures stable results independent of the human factor and facilitates the rapid processing of results during the optimization stage of the methodology.

By utilizing the data fitting function in Python, we obtain a five-dimensional approximation for all parameters (7)–(11) for each fuel assembly. The selection of coefficient values is carried out through the method of least squares, with each data point assigned a weight vector. The reference point is assigned the maximum weight, while equal weights are assigned to the remaining points, ensuring that the curve passes through the reference point.

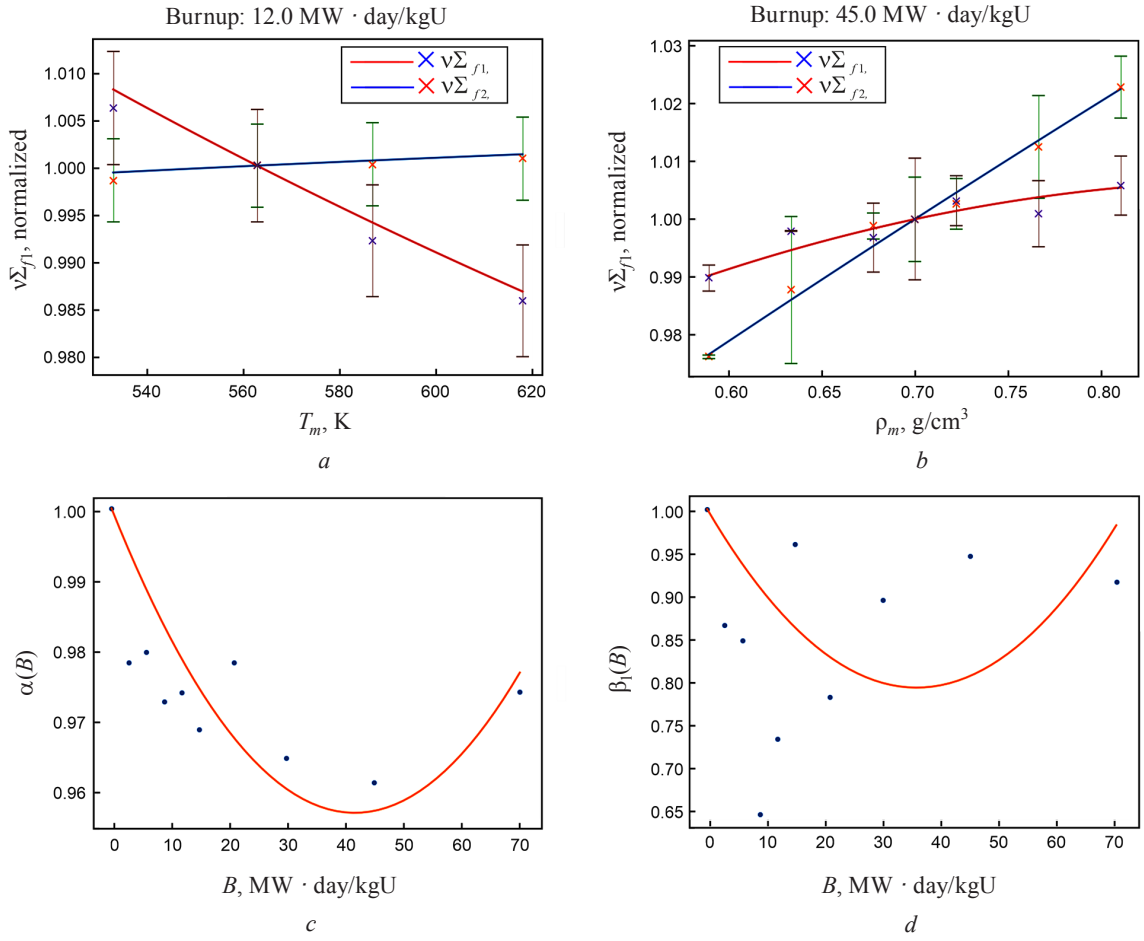


Fig. 4. Examples of cross-sections and interpolation coefficients calculated for the highly enriched fuel assembly 4.9 % ^{235}U. Normalized dependence of the generation cross-sections $\nu\Sigma_{f1/f2}$ on moderator temperature for the burnup of 12 MW · day/kgU (*a*) and on moderator density for the burnup of 45 MW · day/kgU (*b*). Normalized dependence on the burnup of the interpolation coefficients a(*B*) for the transport diffusion coefficient (*c*) and $\beta_1(B)$ for the generation cross-sections $\nu\Sigma_{f1}$ (*d*)

Results of the constant processing methodology application are shown in Fig. 4.

After the coefficient selection in the first stage (dependence on reactor parameters), in the second stage, the coefficients themselves are described by a second-order curve depending on the burnup. Let α be one of the coefficients describing the dependence of a certain cross-section on a specific reactor parameter; then, $\alpha = \alpha_0 + \alpha_1 B = \alpha_2 B^2$. At this stage, maximum weight is assigned to the point with zero burnup, and as a result, the library records the actual value of α for zero burnup as α_0.

The program generates a data library file in formula format (IWQS = 2), which includes a header and data describing an individual fuel assembly. Various control rod materials are set as distinct types for a given fuel assembly. To create a library encompassing data for the entire reactor core, it is necessary to consolidate all data files for individual fuel assembly types into a unified data library file. This file should have fields for data pertaining to all fuel assembly types and all reflectors.

To compare the results, a fuel assembly model in DYN3D corresponding to the Serpent fuel assembly model for approximating an infinite multiplying medium with fuel assembly data was created. Both models calculate $k_{\text{eff}}^{\text{inf}} = k_\infty$, the effective multiplication coefficient of the infinite medium.

The final results of creating data libraries in the IWQS = 2 formula format are presented in Figure 5. These figures show the discrepancy in $\Delta k_\infty = k_\infty^{\text{DYN3D}} - k_\infty^{\text{Serpent}}$ for individual fuel assemblies between DYN3D and Serpent. Each point corresponds to a new combination of reactor parameters, with only one parameter changing at each point.

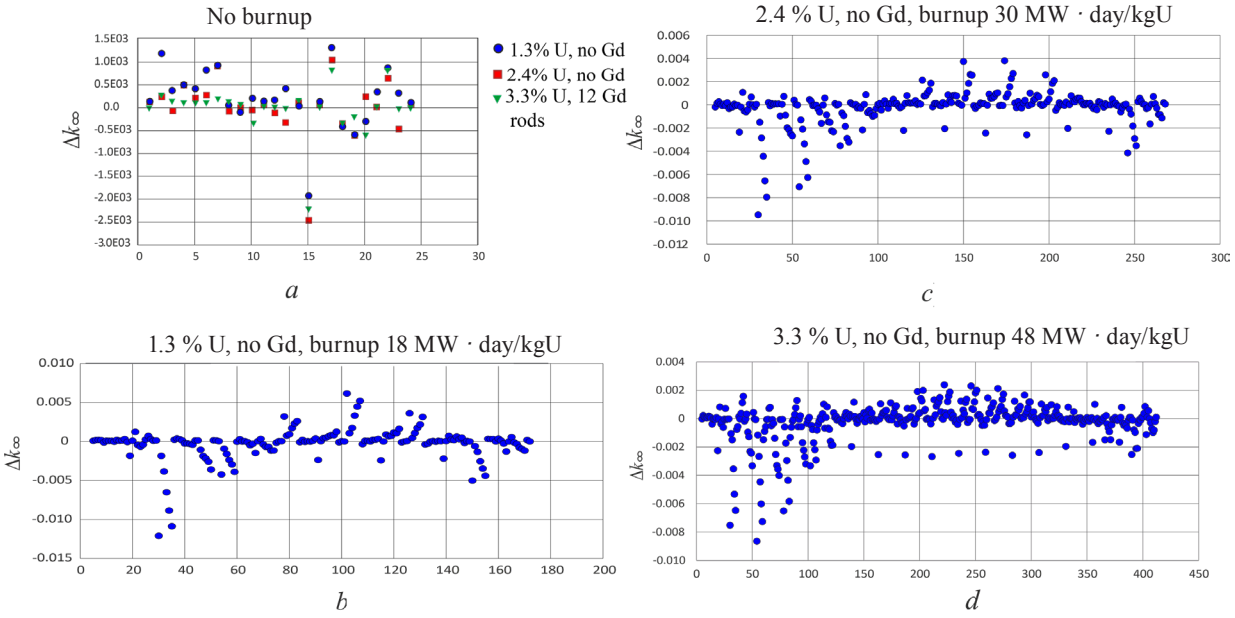


Fig. 5. Difference in Dk_∞ for various types of fuel assemblies with different burnup: *a* it is without burnup for fuel assemblies with low (1.3 %, no Gd fuel rods), medium (3.3 %, 12 Gd fuel rods), and high enrichment (4.9 %, no Gd fuel rods); *b* is 1.3 % ^{235}U, burnup = 18 MW · day/kgU; *c* is 2.4 % ^{235}U, no Gd fuel rods, burnup = 30 MW · day/kgU; *d* is 3.3 % ^{235}U, 12 Gd fuel rods, burnup = 48 MW · day/kgU

In the absence of burnup, the comparison was performed at 21 points, with burnup present, the comparison was performed at up to 576 points for 4.9 % ^{235}U enrichment Fas (Fig. 5).

It is observed that for assemblies with high enrichment, the results fall within 500–600 ppm (parts per million), which is a good result for a formulaic library. However, for assemblies with lower enrichment, the result significantly deteriorates, and for the assembly with the lowest enrichment, unsatisfactory approximation results are evident.

It was concluded that the burnup limit of fuel was considered in the library format itself when selecting the function describing the dependence of cross-sections on the reactor parameters and burnup. Indeed, with deep burnup, ^{239}Pu becomes the main fissile isotope much earlier, leading to significant changes in neutron-physical characteristics. For assemblies with low initial enrichment (1.3–2.4 % in our case), the stage when plutonium becomes the primary fissile isotope occurs at much lower burnup.

After excluding burnup values exceeding a certain maximum value realistically achieved during the reactor campaign, the accuracy of the results significantly improves and practically coincides with the accuracy for assemblies with high enrichment. An exception remains for the assembly with the lowest enrichment, for which the maximum deviation observed at high boric acid concentration is roughly twice that for other assemblies.

We calculated power distribution at a date of 160 effective full-power days and boric acid concentration changes during campaign for first and stationary loadings. The results are presented in Fig. 6.

Estimated first loading campaign duration is about 343 days, stationary campaign duration on boric acid is about 335–337 days, which is in good correlation with real cases.

Conclusion. A method has been developed for calculating two-group constants in the formula format (IWQS = 2) of the DYN3D code for the VVER-1200 reactor using the Serpent Monte Carlo code.

Models of all fuel assemblies and reflectors are created in the Serpent code to perform the calculations of two-group constants and cross-sections necessary to create a DYN3D data library in a formula format. Using the obtained cross-sections and constants, the data library was created through the Python-script.

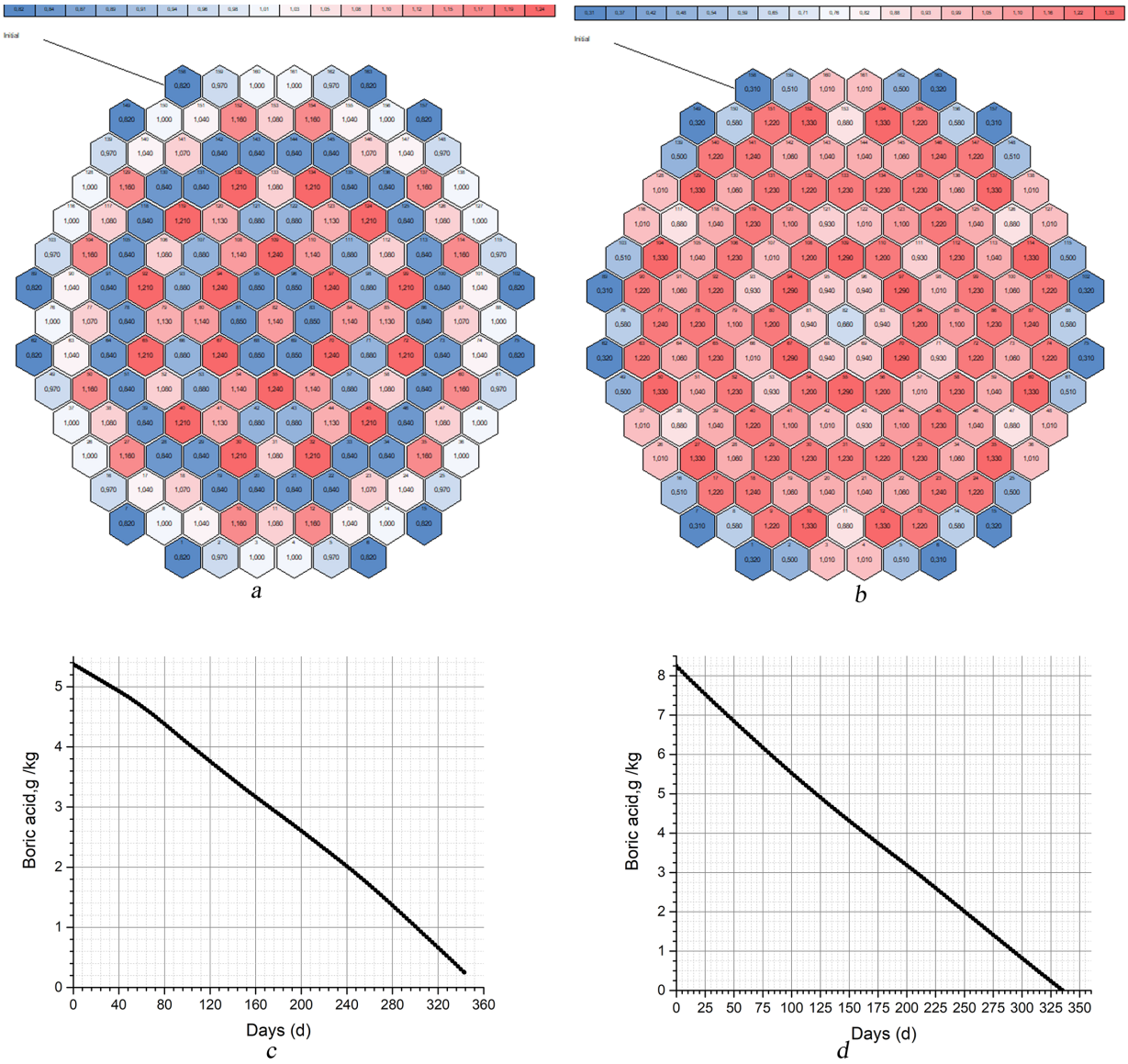


Fig. 6. Power distribution at 160 effective full-power days and boric acid concentration changed during the campaign for the VVER-1200 loadings: first (*a*) and (*c*), stationary (*b*) and (*d*)

A self-consistency check was conducted for the results obtained from $\Delta k_\infty = k_\infty^{\text{DYN3D}} - k_\infty^{\text{Serpent}}$ calculations in the Serpent code and the DYN3D code with the library of two-group constants created in the IWQS = 2 formula format. It has been demonstrated that the maximum discrepancy in $\Delta k_\infty = k_\infty^{\text{DYN3D}} - k_\infty^{\text{Serpent}}$ values for fuel assemblies between these calculations does not exceed 1000 ppm in extreme cases. Furthermore, it has been shown that the maximum discrepancy in $\Delta k_\infty = k_\infty^{\text{DYN3D}} - k_\infty^{\text{Serpent}}$ for low-enriched fuel assemblies (1.3 % ^{235}U) at a maximum burnup of up to 18 MW · day/kg is 1200 ppm.

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