SECTION 2

PROBLEMS OF MODERN NUCLEAR ENERGY

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THE VERIFICATION OF CONSTANT PREPARATION METHODOLOGY FOR IMCORE CODE

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For fuel loading maintenance at Ukrainian nuclear power units, as well as to ensure the reactor unit economical and safe operation, it is necessary to have accurate and timely information on the energy release fields, temperature distributions and other thermal and nuclear parameters inside core. This task is performed by the in-core control systems, which usually include sensors, communication lines, electronic measuring equipment, as well as computers, algorithms and programs for processing the received information [1]. To date, in Ukraine In-Core Monitoring System's (ICMS) with three different neutron-physics calculation subsystem (NPCS) are exploited. They are ICMS's based on "Khortytsya", "Voyage" and "Beacon". Each system has both advantages and disadvantages that occur during the exploitation. Therefore, the study of methods for solving the reactor equation, methods for preparing constant software for neutron-physical calculation module is an urgent task to improve the algorithms for determining the energy release fields in the reactor core. In PJSC «SRPA "Impulse" a working group was created to implement such studies. The paper describes the ImCore code for power distribution calculation. The constant software for ImCore preparation methods analysis and some results of comparison between the fields and the operational data are given.

INTRODUCTION

The obtaining power distribution in the reactor core in different operation modes and the fuel loading calculation task is carried out in two stages. At the first stage, the calculation with the help of the fine-grid code for the core solid element (for example, Fuel Assembly (FA)) is performed in different states to obtain few-group homogeneous macroscopic cross-sections (characteristics) [2, 3]. At the next stage, the entire core calculation in the few-group approximation is carried out using the cross sections prepared in the previous step. Such stages calculations of the reactor core within an acceptable time maintaining the required accuracy level.

Usually the diffusion equation is solved in the two-group energy approximation. The most widespread here is the nodal method, on which basis such computer codes as NESTLE, DYN3D and others [4, 5]. The BIPR computer code solves the diffusion equation in the asymptotic approximation using the finite-difference method [3].

To obtain a high accuracy degree in the diffusion equation solution while using the finite difference method, it is necessary to use a computational grid with a small spacing in the nodes arrangement, which requires more variables and, as a result, more memory space and calculating time. That is why nodal methods are more developed.

But the computational tools development makes it possible to consider the use of the finite-difference method for solving the reactor kinetics equation. At the same time, the method itself makes it possible to obtain a sufficiently high accuracy in calculations, and the existing accelerating calculations methods – to obtain the necessary computation speed [6]. In this paper, a grid method for solving the reactor kinetics equation in two-group diffusion approximation is presented.

1. IMCORE COMPUTER CODE DESCRIPTION

In the general case, the diffusion equation can be presented as:

$$-\mathrm{div}(\mathrm{I}) + \Sigma\Phi = \mathrm{q}, \qquad \mathrm{I} = -\mathrm{D}\,\mathrm{grad}(\Phi). \quad (1)$$

Let this equation be defined in a two-dimensional area S with boundary Γ. Here $\Phi = \Phi(\mathrm{r})$ is a function characterizing the neutron flux density at the point r; $\mathrm{D} = \mathrm{D}(\mathrm{r})$, $\Sigma = \Sigma(\mathrm{r})$ – the diffusion coefficient and the removal cross-section (absorption and transition to another energy group); $\mathrm{q} = \mathrm{q}(\mathrm{r})$ – the neutrons source. As a rule, q includes neutrons that have appeared in the current group as a result of fission and deceleration from the overlying energy groups. When the diffusion equation is solved, the involved in the q formation, function Φ is assumed to be known. The source structure is analogous to the term $\Sigma\Phi$ of the equation (1), and so, the source term finite-difference approximation does not introduce any singularities in comparison with the term $\Sigma\Phi$. It is assume that S area is to consist of homogeneous zones S_f with boundaries Γ_f a finite number.

If the grid nodes are located in the assemblies' centers and the unit cells ΔS_{ij} are correlated with the assemblies, the function Σ and D discontinuity line coincides with the boundary ΔS_{ij}. Fig. 1 shows the calculation scheme with the numbering adopted in ImCore.

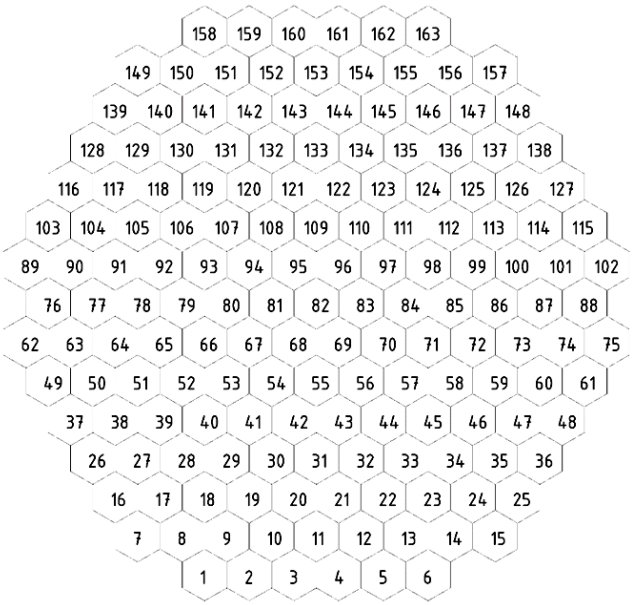

Fig. 1. A hexagonal cell with nodes in the assemblies' centers, "coarse" spacing

Let's consider the assembly number 9. Since the neighboring assemblies sizes are the same, the vector I normal component at the assemblies 8 and 9 boundary can be presented in such a form:

$$I = \frac{\Phi_9 - \Phi_1}{\frac{h}{2}\left(\frac{1}{D^9} + \frac{1}{D^1}\right)} \quad . \tag{2}$$

Integrating the equation (2) over the area of the assembly, we obtain:

$$-a_9^1\Phi_1 - a_9^2\Phi_2 - a_9^{10}\Phi_{10} - a_9^{19}\Phi_{19} - a_9^{18}\Phi_{18} - \\ -a_9^8\Phi_8 + d_9\Phi_9 = q_9 \; . \tag{3}$$

Here

$$a_9^1 = \frac{2}{\sqrt{3}}\frac{D^9D^1}{(D^9 + D^1)}; \qquad a_{22}^{31} = \frac{2}{\sqrt{3}}\frac{D^9D^2}{(D^9 + D^2)};$$

$$d_{22} = \frac{h^2\sqrt{3}}{2}\Sigma^9 + \frac{2}{\sqrt{3}}\frac{D^9D^1}{(D^9 + D^1)} + \frac{2}{\sqrt{3}}\frac{D^9D^2}{(D^9 + D^2)} + \ldots$$

In the event of the calculation scheme crushing and transition to the six-point scheme, each calculation area is divided into 6 equal sections and numbered as shown in Fig. 2.

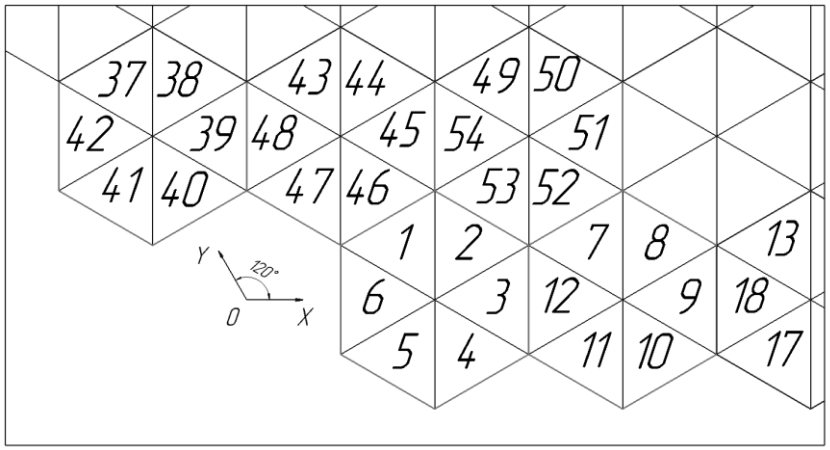

Fig. 2. The calculation areas in a six-point scheme numbering

Also, due to the change in the calculation area form, in particular, due to the transition from the hexagon to the triangle, the coefficients are changed as follows:

$$a_i^j = 2\sqrt{3}\frac{D^iD^j}{(D^i + D^j)};$$

$$d_i = S\cdot\Sigma^i + 2\sqrt{3}\frac{D^iD^j}{(D^i + D^j)} + 2\sqrt{3}\frac{D^iD^{j+1}}{(D^i + D^{j+1})} + \ldots$$

Such a numbering order is used due to the application algorithm simplicity that saves the machine resource.

During the transition from a two-dimensional to a three-dimensional task, the forming calculation matrix logic is preserved, for three-dimensionality an interaction with the "neighbors" along the Z axis is added to each computational area.

A scheme is used where the source is normalized at each iteration. The normalization introduction is advantageous for reducing the number of internal

iterations, since at each successive iteration, the source amplitude is preserved and it is convenient to take the solution obtained at the previous external iteration as a first approximation. Termination condition:

$$K^{(t)}_{eff} - K^{(t-1)}_{eff} \le \varepsilon,$$

where ε is the given calculation accuracy $\varepsilon = 10^{-6}$.

To date, a partition is implemented with different grid spacing, as shown in Fig. 3.

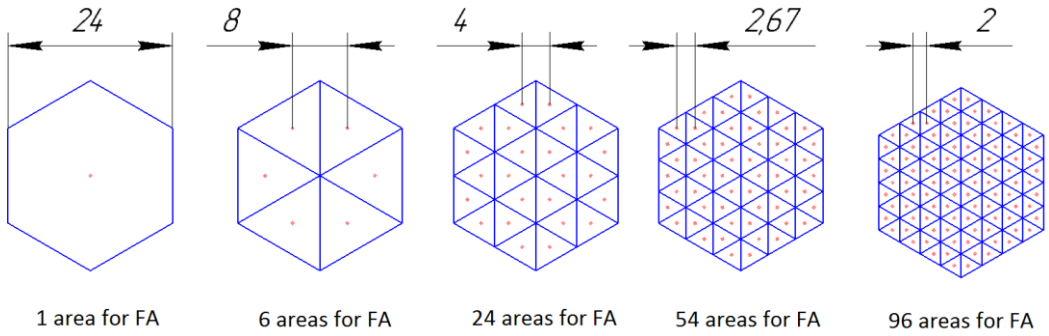

Fig. 3. Variants of the calculation cell partitioning

Preliminary analysis showed (Fig. 4) that even in the case of six division points it is possible to obtain acceptable calculation accuracy with little span time. There is possibility to obtain the calculation results with and without parallel process calculations [7]. Moreover, high accuracy level is achieved already when FA id divided into 54 cells. In addition, the division into 54 cells makes it possible to recover well the fuel elements power distribution.

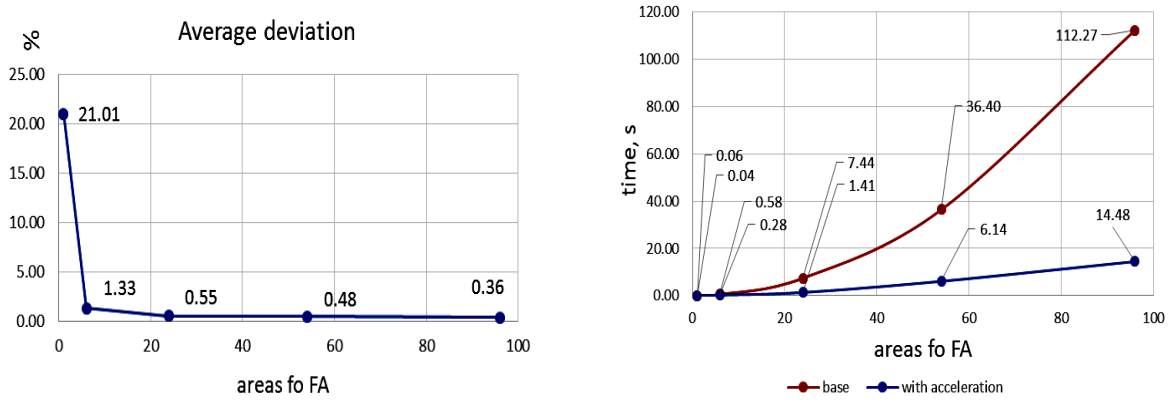

Fig. 4. Variants of cell division. Dependence of calculation speed and accuracy

The program implements schemes both with the reflector (Fig. 5), and without it, with the appropriate boundary conditions specification.

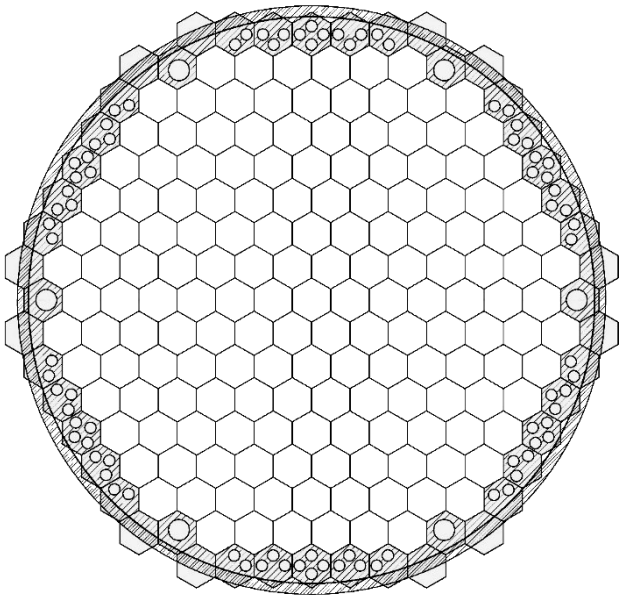

Fig. 5. Calculation scheme with reflector

2. IMCORE CODE CONSTANTS PREPARING METHODOLOGY DESCRIPTION

The obtained homogenized few-group constants greatly affect the further calculations accuracy [8]. The procedure for accounting the feedback and obtaining new updated cross-sections can take up 25% (or more) estimated time at one iteration step. Therefore, great attention is paid to the constant preparation procedure.

To date, the most progressive approach is the one in which a huge multidimensional cross-sections matrix is created depending on various state parameters [9]. The resulting cross-section for specific parameters in the cell is determined from such a matrix by interpolation (cubic splines) between the values.

But such an approach will require a sufficient number of preliminary calculations and, when the cross-section is restored, it will take up a large amount of the estimated time, so its application requires further research.

At this stage, an approach, in which the reactor unit's operation state at power is selected as a reference

point and then the values of each parameter are changed, was chosen for obtaining the homogenized macroscopic constants [10]. It is important to note here that the density and temperature effect on the coolant temperature are considered separately. The basic and changing states parameters are given in the Table.

Basic and changing states parameters

Parameter	Basic state	Range of the parameter change
Coolant/ moderator temperature, K	575	300, 393, 473, 524, 553, 600
Fuel temperature, K	575	300, 600, 900, 1200, 1500
Coolant/ retarder density, g/cm^3	0.7241	0.35, 0.55, 0.75, 0.95
Boric acid concentration, g/kg	0.0	2.0, 4.0, 6.0, 8.0, 10.0, 12.0, 14.0, 16.0

The resulting cross-section is defined as:

$$S = S_0 \cdot \left\{1+\alpha\left(\frac{1}{T_{\text{mod}}}-\frac{1}{T_{\text{mod},0}}\right)\right\}\times$$
$$\times\left\{1+\beta_1\left(\gamma_{\text{mod}}-\gamma_{\text{mod},0}\right)+\beta^2\left(\gamma_{\text{mod}}-\gamma_{\text{mod},0}\right)^2\right\}\times$$
$$\times\left\{1+\gamma_1\left(C_b\gamma_{\text{mod}}-C_{b,0}\gamma_{\text{mod},0}\right)+\right.$$
$$\left.+\gamma_2\left(C_b\gamma_{\text{mod}}-C_{b,0}\gamma_{\text{mod},0}\right)^2\right\}\times\left\{1+\delta\left(\sqrt{T_f}-\sqrt{T}_{\text{mod}}\right)\right\}\ .$$

The preliminary analysis showed that for the state at the Hot zero power one more reference state with the parameters $T_{\text{cool}} = T_{\text{fuel}} = 473$ K, coolant density – 0.8754 g/cm^3, boric acid concentration – 0.0 g/kg should be singled out.

This need is mainly due to the correct consideration of feedbacks on the boric acid concentration, as it is shown in the Fig. 6.

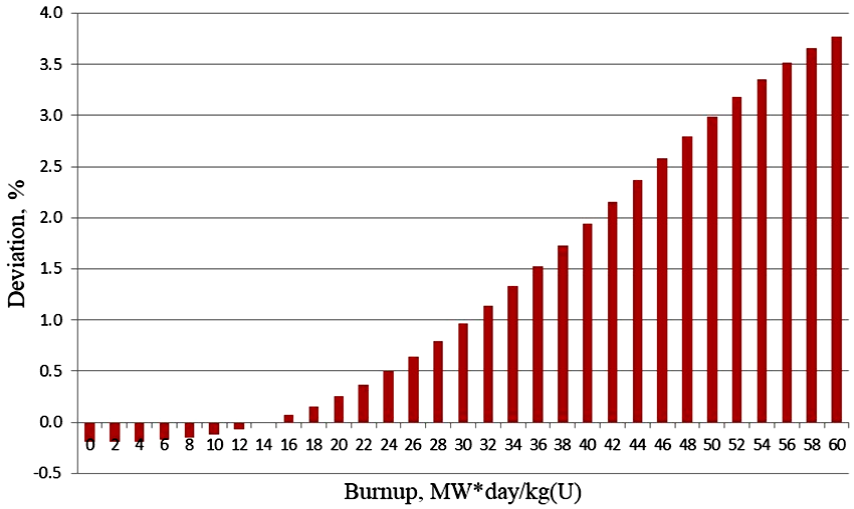

Fig. 6. The multiplication factor deviation for two reference states

Therefore, for the second reference state, the boric acid concentration series will be sufficient. The coefficients for derivatives with respect to other parameters remain unchanged.

3. CALCULATIONS AND RESULTS

Test calculations are performed for various fuel loads. Comparisons are made using the regular physical calculations subsystem "Cruise" calculation results with the values correction according to the in-reactor direct-charge detectors. Constant software for the results given in this paper was prepared using WIMSD-5B computer code [11].

Understanding WIMSD-5B limitations the transition was began to solve this problem with Serpent Monte Carlo computer code [12, 13].

To carry out the calculation using the ImCore code, a six-point calculation scheme was made with a partition into 16 layers in height (similar to the regular NPCS). The comparison results between the ImCore code calculations and operational data for k_q and k_v fields are shown in the Figs. 7 and 8.

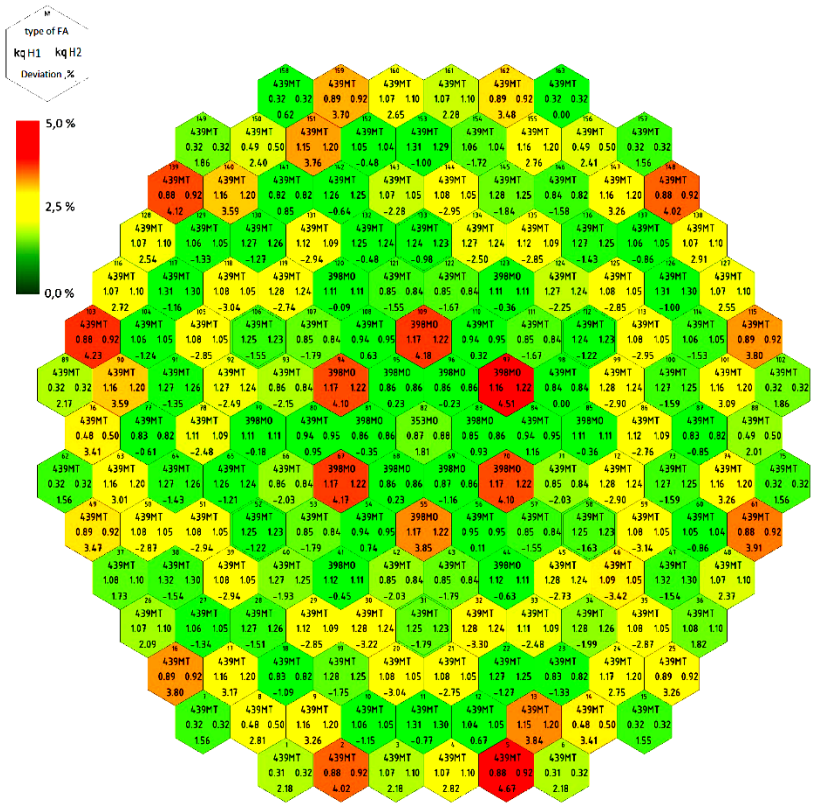

Fig. 7. The k_q field comparison results (set1 – operational data, set2 – ImCore)

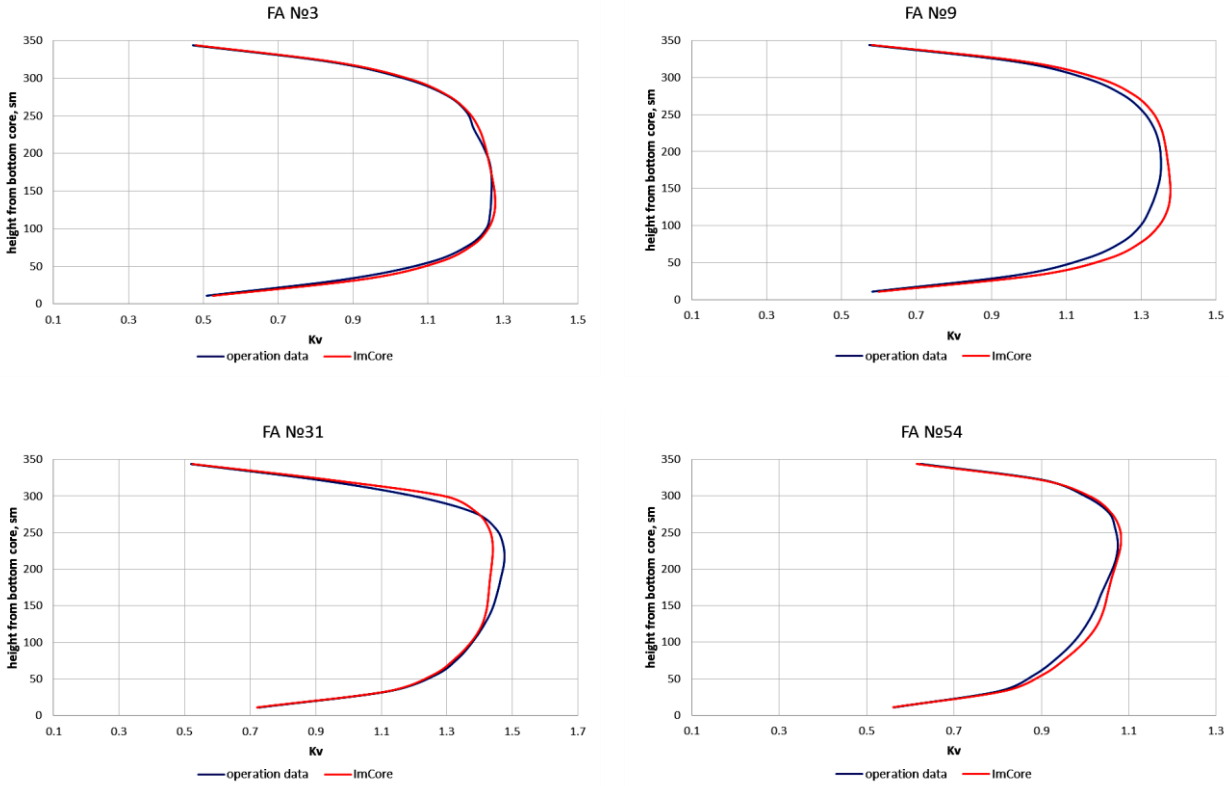

Fig. 8. The comparison results between the ImCore code calculations and operational data for k_v field

4. CONCLUSIONS AND OUTLOOK

The information on the ImCore software for the reactor core calculation in the few-group diffusion approximation is presented. One of the few-group constants preparation methods for the code is given.

The finite-difference method for the few-group diffusion equation solution with presented macro-cross section preparation procedure give required precession with needed calculation speed and can be used for ICMS for calculate and retrieval core power distribution.

REFERENCES

1. В.А. Брагин, И.В. Батенин, М.Н. Голованов и др. *Системы внутриреакторного контроля АЭС с реакторами ВВЭР*. М.: «Энергоатомиздат», 1987, 128 с.

2. Е.А. Жолкевич. *Аппроксимация зависимостей нейтронно-физических характеристик решеток ВВЭР от параметров состояния активной зоны*: Препринт ИАЭ-3558/5. М.: ИАЭ, 1982, 69 с.

3. Д.М. Петрунин, Е.Д. Беляева, И.Л. Киреева. *Программа БИПР-5. Описание структуры и входных даннях*: Препринт ИАЭ-2519. М.: ИАЭ, 1975, 37 с.

4. U. Grundmann, U. Rohde, S. Mittag, et al. *DYN3D Version 3.2. Code for Calculation of Transients in Light Water Reactors (LWR) with Hexagonal or Quadratic Fuel Elements*. Forschungszentrum Rossendorf: Institute of Safety Research, 2005, 140 p.

5. P.J. Turinsky, R.M.K. Al-Chalabi, P. Engrand, et al. *NESTLE: A few-group neutron diffusion equation solver utilizing the nodal expansion method for eigenvalue, adjoint, fixed-source steady-state and transient problems*. Raleigh: NC 27695-7909, 2003, 169 p.

6. Л.К. Шишков. *Системы внутриреакторного контроля АЭС с реакторами ВВЭР*. М.: «Энергоатомиздат», 1987, 128 с.

7. The OpenMP API specification for parallel programming, http://www.openmp.org/specifications/ (accessed 14.01.18).

8. J. Watson, K. Ivanov, R. Macian, et al. Cross-Section Generation Methodology for Three-Dimensional Transient Reactor Simulation //*Transactions of the American Nuclear Society*. 1997, v. 77, p. 175-176.

9. J. Watson, K. Ivanov. Improved cross-section modeling methodology for coupled three-dimensional transient simulations // *Annals of Nuclear Energy*. 2002, v. 29, p. 937-966.

10. В.В. Гальченко, А.А. Мішин. Порівняльний аналіз нейтронно-фізичних характеристик кампанії реактора з використанням різних наборів бібліотек ядерних даних для програмного продукту WIMSD5B // *Ядерна та радіаційна безпека*. 2015, №3(67), с. 8-12.

11. J.R. Askew, F.J. Fayes, P.B. Kemshell. A general Description of the lattice Code WIMS // *Journal of British Nuclear Energy Society*. 1966, v. 5, p. 564-584.

12. J. Leppänen, M. Pusa, T. Viitanen, et al. The Serpent Monte-Carlo code: Status, development and applications in 2013 // *Annals of Nuclear Energy*. 2015, v. 82, p. 142-150.

13. В.В. Гальченко, В.І. Гулік, І.І. Шлапак. Використання програмного продукту Serpent на основі методу Монте-Карло для розрахунку характеристик тепловиділяючої збірки реактора ВВЕР-1000 // *Ядерна фізика та енергетика*. 2016, №17(3), с. 232-239.

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ВЕРИФИКАЦИЯ МЕТОДОЛОГИИ ПРИГОТОВЛЕНИЯ КОНСТАНТ ДЛЯ КОДА IMCORE

В.В. Гальченко, И.И. Шлапак, В.И. Гулик, Д.В. Будик

Для сопровождения топливных загрузок на украинских атомных энергоблоках, а также для обеспечения экономичной и безопасной эксплуатации реакторной установки необходимо наличие точной и оперативной информации о распределении полей энерговыделения, температуры и других теплотехнических и ядерно-физических параметров внутри активной зоны. Эту задачу выполняют системы внутриреакторного контроля (СВРК), в состав которых в общем случае входят датчики, линии связи, электронная измерительная аппаратура, а также ЭВМ, алгоритмы и программы для обработки полученной информации. На базе ЧАО «СНПО «Импульс» создана рабочая группа для реализации проекта СВРК-М2, в котором в качестве подсистемы физического расчета используются украинские разработки программных продуктов для расчета и восстановления поля энерговыделения в активной зоне реактора. В работе представлено описание расчетного модуля для расчета и восстановления поля энерговыделения программного комплекса ImCore. Приведен анализ методик подготовки константного обеспечения для ImCore, а также некоторые результаты сравнения полей с эксплуатационными данными.

ВЕРИФІКАЦІЯ МЕТОДОЛОГІЇ ПРИГОТУВАННЯ КОНСТАНТ ДЛЯ КОДУ IMCORE

В.В. Гальченко, І.І. Шлапак, В.І. Гулік, Д.В. Будік

Для супроводження паливних завантажень на українських атомних енергоблоках, а також для забезпечення економічної та безпечної експлуатації реакторної установки необхідна наявність точної та оперативної інформації про розподіл полів енерговиділення, температури та інших теплотехнічних і ядерно-фізичних параметрів всередині активної зони. Це завдання виконують системи внутрішнього реакторного контролю (СВРК), до складу яких у загальному випадку входять датчики, лінії зв'язку, електронна вимірювальна апаратура, а також ЕОМ, алгоритми і програми для обробки отриманої інформації. На базі ПрАТ «СНВО «Імпульс» створена робоча група для реалізації проекту СВРК-М2, в якому в якості підсистеми фізичного розрахунку використовуються українські розробки програмних продуктів для розрахунку і відновлення поля енерговиділення в активній зоні реактора. У роботі представлено опис розрахункового модуля для розрахунку і відновлення поля енерговиділення програмного комплексу ImCore. Наведено аналіз методик підготовки константного забезпечення для ImCore, а також деякі результати порівняння полів з експлуатаційними даними.