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The concerted calculation of the BN-600 reactor for the deterministic and stochastic codes

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The concerted calculation of the BN-600 reactor for the deterministic and stochastic codes

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Abstract. The solution of the problem of increasing the safety of nuclear power plants implies the existence of complete and reliable information about the processes occurring in the core of a working reactor. Nowadays the Monte-Carlo method is the most general-purpose method used to calculate the neutron-physical characteristic of the reactor. But it is characterized by large time of calculation. Therefore, it may be useful to carry out coupled calculations with stochastic and deterministic codes. This article presents the results of research for possibility of combining stochastic and deterministic algorithms in calculation the reactor BN-600. This is only one part of the work, which was carried out in the framework of the graduation project at the NRC "Kurchatov Institute" in cooperation with S. S. Gorodkov and M. A. Kalugin. It is considering the 2-D layer of the BN-600 reactor core from the international benchmark test, published in the report IAEA-TECDOC-1623. Calculations of the reactor were performed with MCU code and then with a standard operative diffusion algorithm with constants taken from the Monte - Carlo computation. Macro cross-section, diffusion coefficients, the effective multiplication factor and the distribution of neutron flux and power were obtained in 15 energy groups. The reasonable agreement between stochastic and deterministic calculations of the BN-600 is observed.

1. Introduction

Verification of modern program codes and methods of fast reactors core calculation is an important step in research activity devoted to possibilities of plutonium usage and minor actinides transmutation. Precision of neutron transfer modeling predictions and related reactivity coefficients should be evaluated.

Such evaluations should be received from comparison of calculations with real reactor conditions. However, the processes in reactor are very complicated. Precise and reliable information about these processes is difficult to obtain. Necessary detectors cannot be positioned everywhere and their indications have significant uncertainties. Moreover, the information about reactor design is not absolutely correct. Small geometry or composition deviations related with technological limits can make significant influence on calculation results.

This situation is simplified by the fact that the multiform process is calculated with complicated program complex, which gives an opportunity to verify parts of this complex separately. Thus, the quality of neutron calculation helps to establish the application of mathematical Monte-Carlo based

model. According to the verified program such model has no differences from real reactor and allows obtaining any data necessary for results analyses. Information about cross-sections, reactions rates, fluxes and diffusion coefficients is located in MCU output [1] and can be transferred to some operative algorithm with the results that are possible to compare with corresponding MCU results. Such comparisons for VVER reactors were published earlier [2], [3].

2. Formulation of the problem

In this paper the 2-D layer of BN-600 core from international mathematic test published in IAEA report IAEA-TECDOC-1623 [4] is considered. Cartogram of the core in MCU coding is shown in Figure 1. Core is surrounded by five rows of reflector cells. There are seven fuel assembly types in this configuration.

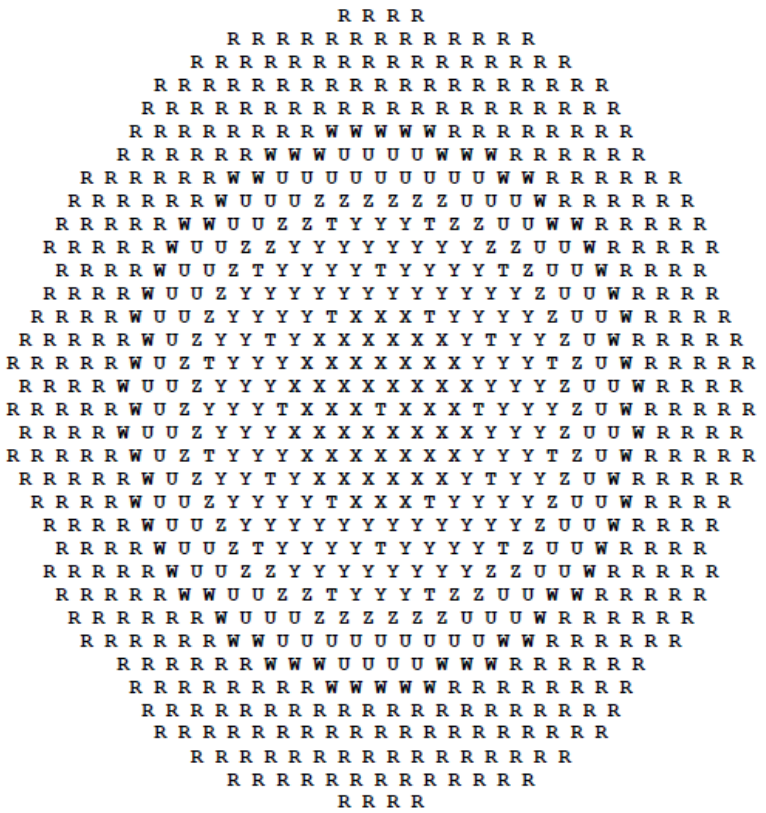

Figure 1. Core cartogram in MCU coding. T - control rod, R - reflector, U - mixed oxide zone, W - high enrichment outer zone, X - low enrichment zone, Y - another low enrichment zone, Z - middle enrichment zone

Nuclear concentrations are described in [4]. Approximate «fission product» is included in these concentrations. Also there is a significant axial leakage in 3-D variant. In 2-D variant the leakage is absent and the fission product was not included in MCU input. Therefore, the model we used turned notably over-critical.

The supposing condition on flux equality to zero on core side surface is difficult too model with Monte-Carlo code in described 2-D tests. In our calculations we changed it by the demand of full neutron absorption on side surface. This surface has complicated geometry. In order to compute it the core was surrounded by the additional row of cells, which were united in special zone. On the surface of this zone full absorption condition can be imposed. This zone is marked with gray color in the Figure 2.

After the calculations on MCU code we computed this core with standard diffusion program which used the constants obtained by Monte-Carlo calculations. The mirror reflection condition was stated on upper and lower surfaces and zero flux condition in special zone cells surrounding the reactor was stated on the side surface.

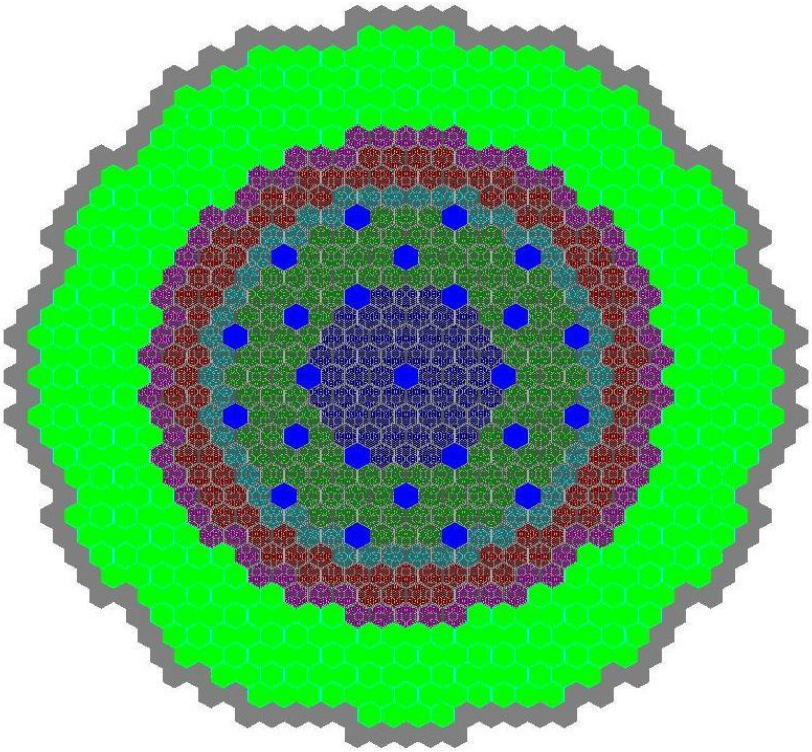

Figure 2. Core in calculation environment.

3. Stochastic and deterministic calculations

In calculation of the described source case 4.8 billion of neutrons were modeled so that the statistic uncertainty of fluxes distribution and reactions rates is about 0.1%. 96 processors were used, therefore the problem was solved in 11 hours.

Macro cross-section and diffusion coefficients in 15 energy groups were obtained for each of 823 assembly cells with MCU code. Boundaries of energy groups are shown in Table 1. Those cell characteristics were transferred to HEM-3 code [5], which made 15-group calculation of K_{eff} and distribution of group neutron fluxes and power that is proportional to MCU registered fission reaction intensity in standard finite-difference homogenous diffusion algorithm.

It is interesting that 40 years ago HEM-3 code was originally made in NRC "Kurchatov Institute" for 2-group calculations of cores with square cells. 25 years ago the ability of multigroup calculations and hexagonal cells were added. After that, the code was rarely used in 1-4 groups mode. About three years ago several cases with up to 12 groups were calculated. In these cases cell characteristics were obtained from whole core calculations with application of Monte-Carlo method and the results of finite-difference calculations were compared to stochastic as benchmarks.

4. Comparison of stochastic and deterministic calculations results

Both power production calculations were normalized by single fission energy. The differences in received distributions are often measured in percent of functional average value in whole calculation field, however in our case the major part of this field is filled with reflector cells. That is the average values are decreased and deviations are noticeably increased. Therefore, deviations are calculated as percent of functional maximum value. The main results are shown in Tables 2, 3.

Table 1. Boundaries of energy groups.

Group	Emin, eV	Emax, eV
1	6.50E+06	1.05E+07
2	4.00E+06	6.50E+06
3	2.50E+06	4.00E+06
4	1.40E+06	2.50E+06
5	8.00E+05	1.40E+06
6	4.00E+05	8.00E+05
7	2.00E+05	4.00E+05
8	1.00E+05	2.00E+05
9	4.65E+04	1.00E+05
10	2.15E+04	4.65E+04
11	1.00E+04	2.15E+04
12	4.65E+03	1.00E+04
13	2.15E+03	4.65E+03
14	1.00E+03	2.15E+03
15	0.00E+00	1.00E+03

Table 2. Effective neutron multiplication factor and maximum of energy discharge in stochastic and deterministic calculations.

	MCU	HEM-3
Effective neutron multiplication factor	1.1912	1.1893
Maximal value of assembly power per 1 neutron	0.001323	0.001330

Table 3. Average and maximum deviation calculations as percent of functional maximum value.

Group	Average deviation, %	Maximum deviation, %
1	0.4	3.2
2	0.5	2.2
3	0.5	2.5
4	0.4	1.6
5	0.3	1.7
6	0.6	1.7
7	1.5	3.5
8	1.7	3.7
9	1.8	3.9
10	2.7	6.1
11	2.7	6.3
12	2.0	4.5
13	4.4	8.8
14	4.6	22.0
15	12.9	28.3

As seen, the differences K_{eff} reach ~ 0.2%. Such and slightly bigger underestimations are observed regularly when MCU is used as benchmark for operative calculations. Hopefully, further experience will bring some clarity to this issue.

The differences in lower groups fluxes as shown in Table 3 are at first glance too big. However, reactor radial flux distribution shows that these fluxes make low contribution on the whole flux. The distribution of fluxes is presented in the Figure 3. According to the reactor radial flux distribution obtained with MCU and HEM-3 (Figure 4, 5) main biases in these fluxes are not in the core so that the importance of these biases is low. Favorable impression of performed calculations is provided by Figure 4, 5.

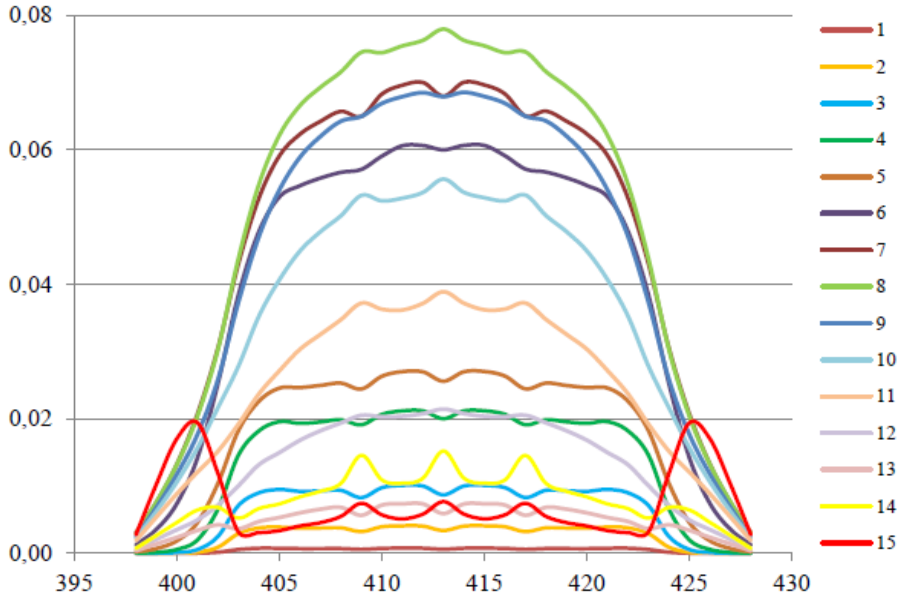

Figure 3. Distribution of group neutron fluxes calculated by MCU in horizontal diameter of reactor.

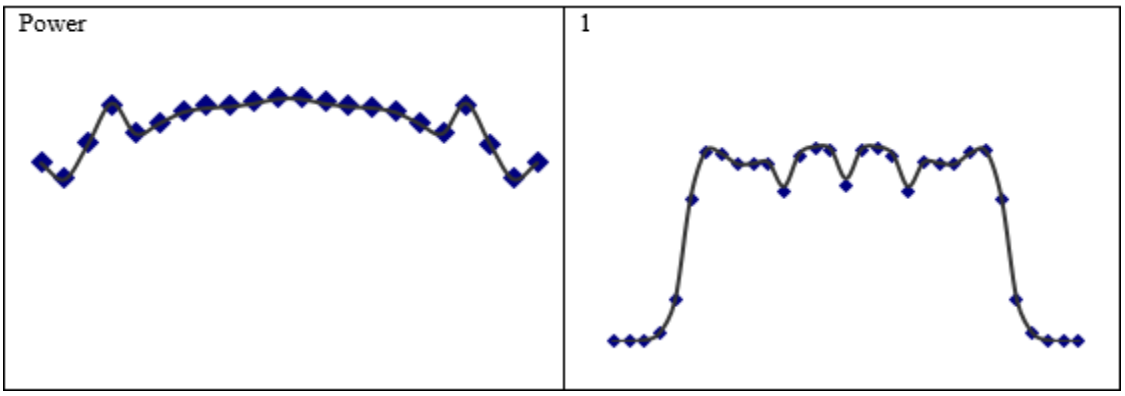

Figure 4. Flux distribution obtained with MCU (♦) and HEM-3(—) in horizontal diameter of reactor on the first energy group and energy production along the horizontal line next to the diameter on Figure 1.

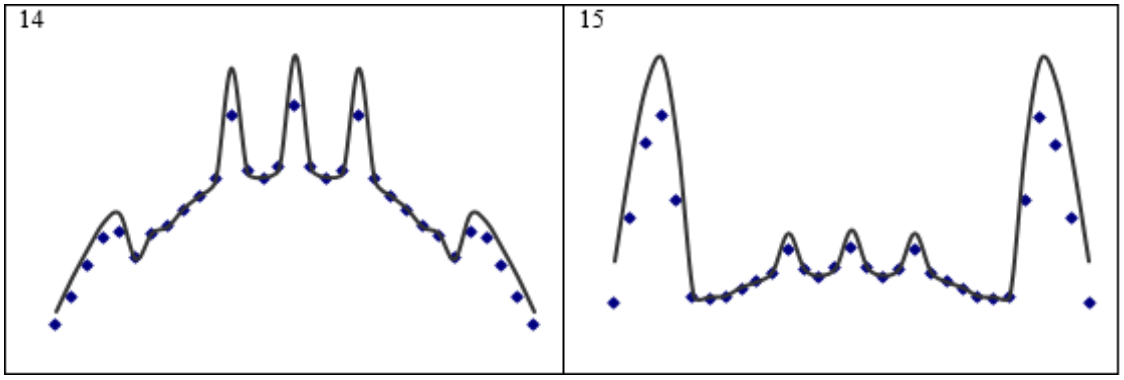

Figure 5. Flux distribution obtained with MCU (♦) and HEM-3 (—) in horizontal diameter of the reactor in the 14 - 15 energy groups.

5. Conclusion

The results of this work show reasonable agreement in stochastic and deterministic calculations of BN-600 reactor. The necessary cell characteristics for deterministic calculation were obtained as a byproduct of stochastic calculation without any subsequent corrections. Further researches are advisable for the issue of agreement improvement possibility with the use of the same group number or with single node of finite-difference mesh for a cell and alteration of these parameters.

References

[1] Alekseev N I, Bolshagin S N, Gomin E A, Gorodkov S S, Gurevich M I, Kalugin M A, Kulakov A S, Marin S V, Novoselsev A P, Oleinik D S, Prianichnikov A V, Sukhino-Fomenko E A, Shkarovskiy D A, Yudkevich M S 2011 Status of MCU-5 *Probl. of At. Sci. and Techn.: Phys. of Nucl. React.* ed 4 pp 4 – 23

[2] Gorodkov S S and Kalugin M A 2014 Applying full multigroup cell characteristics from MCU code to finite difference calculations of neutron field in VVER core *Proc. of the 24th Symp. of AER* Sochi Russian Federation pp 181 - 188

[3] Gorodkov S S, Kalugin M A and Kuznetcov A N 2015 Finite - difference fine mesh 2d calculations of neutron fields in VVER reactors using full sets of multigroup cell characteristic obtained with the MCU package *Proc. of 25th Symp. of AER* Balatongyörök, Hungary pp 571 – 579

[4] BN-600 hybrid core benchmark analyses 2009 IAEA_TECDOC-1623. Vienna

[5] Gorodkov S S 1984 Annotation of the program HEM-3 *Probl. of At. Sci. and Techn.: Phys. and Techn. of the Nucl. React.* ed 85(4) pp 42 – 42

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