ANALYSIS OF ONE-GROUP AND MULTIGROUP TRANSPORT EQUATION SOLUTION METHODS FOR NUCLEAR REACTOR ASSEMBLIES

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ABSTRACTS

First collision probability method is analyzed for solving of transport equation for neutron flux distribution in the assemblies of the VVER-1000 types nuclear reactor. The module of combinatorial geometry is used to describe the geometrical parameters of the system. The collision probabilities are calculated with using of Korobov "equal distributed" sequences. One-group problem with uniform isotropic source is considered. On this stage the possibility of this method to get the reference solution for different types of the assemblies is discussed. Different ways of the spatial subdivision are considered. The time expenditures versus the definite accuracy are given for PC Pentium-100.

Multigroup eigenvalue problem for the neutron flux distribution in the VVER-type reactor assembly is analyzed. The transport approximation is used to take into account anisotropic scattering The iteration way to get the transport equation solution in the thermalization region is considered. The different ways of the group subdivision in the thermalization region are analyzed. The calculation results –the multiplication factor K_{inf} and energy distribution over assemblies fuel rods are compared with the same results published for international benchmark-tests.

1.INTRODUCTION

One of the most important tasks of the nuclear reactors theory and calculation methods is the increasing of the neutron-physical calculations precision. The complicate structure of the modern reactors causes the very careful analysis of the used approximations that are the basis of the calculation methods because it is necessary to estimate the discrepancy on different stages of the calculation process.

So it is very important to have a set of sharp formulated problems-so called benchmark-tests which should include the physical and construction specific features of real reactors. Because of the analytical methods hardly ever are valid for the modern reactors the calculation methods, algorithms and codes are the base instrument for this purpose. To estimate the validity of one or another approximation it is necessary to analyze the available solution of the benchmark-tests which are given with using of different codes; and the appropriated approximations must to be formulated very carefully.

Also it is actually to estimate the time expenditure versus the appropriate calculation precision.

There are two main steps of approximations in the spatial-energy transport neutron flux calculation: the definition of the group cross section on the basis of the estimate nuclear data library and just the solution of the many–group transport equation with definite group cross section in the considered volume. Just the second part of this problem is analyzed in this work. First collision probabilities method (FCPM) (Bell, Glasstone,1974) is used and it has the important property: it converges to exact solution of the transport equation versus enough fine subdivision into calculation zones (neutron flux is supposed to be constant in every such zones). A set of problems "attached " to the reactors of VVER-1000 type are formulated. Algorithm and code based on the FCPM are described. Possibility of this code to get "precision " solution for one-group problem with the definite group cross sections and the definite isotropic source is analyzed. The results of comparison of neutron flux distribution with these results obtained with another codes are given.

Multigroup transport problem is considered too. The transport approximation is used to take into account anisotropic scattering The group cross section are calculated with using the library of nuclear dates of WIMS-code (Askew, 1966). Algorithm and code have created to get the transport solution . Algorithm is based on the FCPM too, and iteration procedure is used in the thermalization region. This code is used for analyses of the international benchmark-tests. The results obtained with using the different ways of the group subdivision in the thermalization region are compared with the results obtained with another codes.

2. ONE GROUP PROBLEM CALCULATION ANALYSIS OF THE ASSEMBLIES OF THE VVER-1000 TYPE NUCLEAR REACTOR.

In this part a set of one–group problem of the neutron flux distributions in the typical assemblies of the VVER-1000 nuclear reactor is analyzed.

Geometry of the problem is specified in Fig.1 that depicts the 60-deg radial section of fuel assembly; rotational symmetry is assumed. There are 312 fuel rods and 18 absorber rods in the assembly. This problem is analyzed in the paragraph 2.1 The problem when the absorber rods are removed is analyzed in the paragraph 2.2. In the paragraph 2.3 the problem when there are two fuel elements with Cd_2O_3 in the 60-deg radial section of the assembly. More detail figures and descriptions of the VVER-1000 assemblies see (Lazarenko, 2000)

Regular triangular step for fuel rods (FR) and absorbers rods (AR) is h=1.275cm. Fuel assembly pitch size (including one half of water gap between two neighbouring assemblies is H=23.6 cm.)

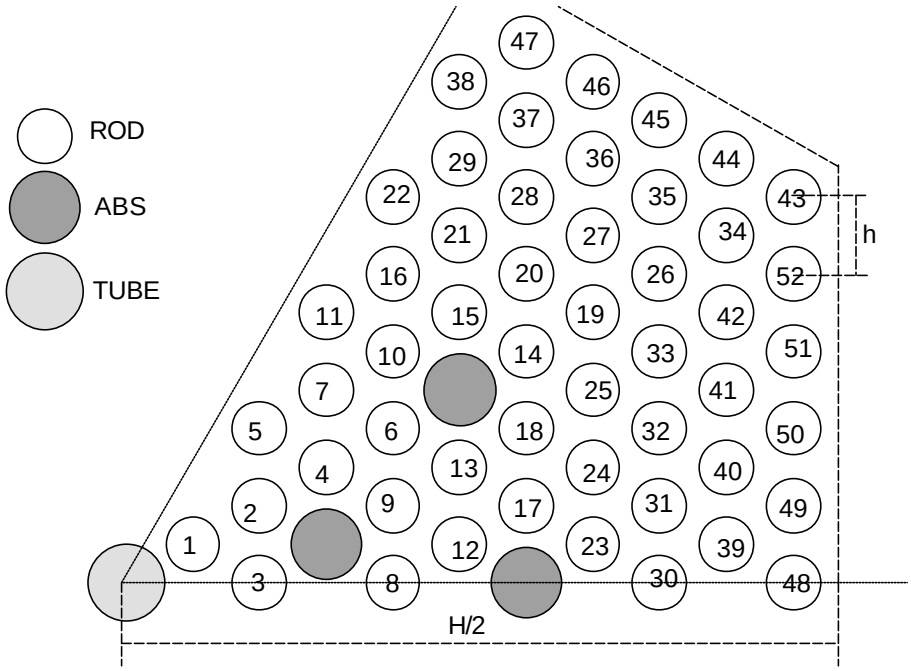

Fig.1. VVER-1000 fuel assembly (60-deg. sector, rotational symmetry)

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The cross sections for considered materials are supposed to be typical for one group problem. Five materials are used: water, cladding, absorber, fuel 1 (for ordinary FR) and fuel 2 (for U-Gd FR). These cross sections are given in the table 2.1

Table 2.1 One-group cross section for VVER-1000 type reactor assembly

Material	Absorbtion, cm^{-1}	Total, cm^{-1}
Water	$6.86*10^{-3}$	1.314
Cladding	$3.30*10^{-3}$	0.27
Absorber	0.568	0.577
Fuel 1	0.242	0.743
Fuel 2	20.16	20.17

The following assumptions are supposed to be valid when these one- group problems are analyzed :

- one-group transport equation with isotropic scattering is valid in the considered volume;
- there are isotropic uniform sources in the water zones;
- the conditions of mirror reflection are valid at the hexagonal boundary ;
- the integral surface current equal zero

$$f^{0}(\bar{r}) = \int_{V} d\bar{r}^{\prime\prime} \left[f^{0}(\bar{r}^{\prime\prime}) s_{s}^{0}(\bar{r}^{\prime\prime}) + q(\bar{r}^{\prime\prime}) \right] P(\bar{r}^{\prime\prime} \to \bar{r}) + \int_{S} d\bar{r}^{\prime} I^{+}(\bar{r}^{\prime}) P(\bar{r}^{\prime} \to \bar{r}) \qquad (1)$$

here the following designations are used :

$t\left(\bar{r}' \to \bar{r}\right) = \int_0^1 s\left(\bar{r}' + s\left(\bar{r} - \bar{r}'\right)\right)\left|\bar{r} - \bar{r}'\right| ds$ - optical distance between $\bar{r}$ and $\bar{r}'$;

$u\left(\bar{r}' \to \bar{r}\right) = \exp\left(-t\left(\bar{r}' \to \bar{r}\right)\right)$ - the probability for neutron to rich from $\bar{r}'$ to $\bar{r}$ without any collision;

$P\left(\bar{r}' \to \bar{r}\right) = \dfrac{u\left(\bar{r} - \bar{r}'\right)}{4p\left|\bar{r} - \bar{r}'\right|^2}$ - usual kernel of the integral transport equation

$\int_V$ means integration over the considered volume; $\int_S$ means integration over the surface surrounding the considered volume

The main assumption of FCPM is that the neutron flux is approximated by it's middle value in every calculation zone (flat flux approximation)

$$f^0(\bar{r}) = \sum_t g_t(\bar{r}) j_t^0, \qquad (2)$$

and the system of the linear algebraic equations is as result

$$j_t^0 V_t = \sum_{t'} \left(s_{st'} + q_{t'}\right) \int_{V_t} d\bar{r} \int_{V_{t'}} d\bar{r}' P\left(\bar{r}' \to \bar{r}\right) + \int_{V_t} d\bar{r} \int_S d\bar{r}' I\left(\bar{r}'\right) P\left(\bar{r}' \to \bar{r}\right) \qquad (3)$$

The coefficients of this system are FCP. Algorithm of the calculation of these coefficients in arbitrary geometry is described in (Shevelev, 1986). Numerical integration way uses Korobov uniform distributed sequences. (Sobol, 1981)

2.1 The assembly with 18 absorber rods.

To analyze the convergence of the FCP-solution to the solution of the transport equation the different ways of the subdivision of the considered system into the calculation zones are used. First way is that every fuel rod is one calculation zone; water filling the hexagonal surround every fuel rod is one calculation zone too. The claddings of all FR are associated into the one calculation zone. The same subdivision way with AR: one zone over the rod, one zone over the surrounded water and the claddings are associated into two calculation zones accordingly (there are two cladding types in the AR cell, see for ex. (Poveschenko, 1999)). There are 118 calculation zones in this subdivision way; the properties of rotation symmetry are taken into account. The accuracy of the convergence of the solution is analyzed versus the trajectories number. The number integration way use Korobov's sequences, so the trajectories' number is defined by Korobov's number.The results are given in tables 2.3

Table 2.2 Choose of the trajectories number when the subdivision way 1 is used

Line's number	1	2	3
Korobov's number	10007	20039	52091

Table 2.3 Neutron flux density distribution over fuel rods when the subdivision 1 is used

6.159 6.148 6.149	5.876 5.873 5.876		5.757 5.760 5.762	5.804 5.814 5.810	5.952 5.945 5.947	6.242 6.240 6.238	6.456 6.462 6.456	6.670 6.681 6.682	6.984 6.989 6.984
5.853 5.838 5.841	5.780 5.788 5.789	5.768 5.773 5.772	5.779 5.790 5.790	5.907 5.898 5.900	6.177 6.170 6.171	6.398 6.408 6.403	6.652 6.639 6.632	6.940 6.940 6.942	
5.763 5.770 5.771	5.856 5.853 5.853	5.802 5.800 5.800		6.051 6.047 6.048	6.359 6.357 6.355	6.595 6.605 6.602	6.913 6.916 6.920		
5.787 5.786 5.786	5.820 5.827 5.827	5.903 5.904 5.902	6.063 6.051 6.051	6.328 6.332 6.335	6.592 6.586 6.586	6.938 6.904 6.908			
	6.045 5.951 5.950	6.238 6.171 6.173	6.410 6.353 6.354	6.613 6.589 6.585	6.910 6.902 6.905				
6.045 6.031 6.030	6.238 6.240 6.239	6.410 6.405 6.403	6.613 6.600 6.601	6.910 6.912 6.909					
6.331 6.331 6.328	6.450 6.460 6.458	6.641 6.628 6.633	6.912 6.922 6.920						
6.531 6.528 6.539	6.676 6.686 6.680	6.939 6.938 6.942							
6.770 6.768 6.772	6.977 6.986 6.984								
7.080 7.080 7.080									

This table is given as a symmetry angle of the assembly. First table raw corresponds the assembly diagonal where FR are located. The “missed” seculars corresponds the AR, every table line corresponds the fuel rod’s raw between the assembly center and the assembly boundary. “Symmetrical” fuel elements (over symmetrical assembly angle) are located on the table diagonal. Calculation accuracy concerning the “symmetrical FR” is <0.05% when number of trajectories is 20039. Calculation time is 2.08 min

Second way of the subdivision of this assembly into the calculation zones is more detail. To be more precise just the hexagonal calculation water zone surround every FR and every AR has been divided into 6 supplement calculation zones There are 378 zones when this subdivision way is used. Precision of the solution convergence versus the trajectories number is analyzed too. The precision over “symmetrical FR” <0.05% has

been reached when the trajectories number is 20039 .The results are given in the table 2.4 The appropriation of the FR and the table seculars is the same that for the table 4.3

Table2.4 Neutron flux density over the assembly FE when the supplement calculation zones are used in the assembly water.(second way of subdivision)

6.152	5.881		5.769	5.818	5.952	6.243	6.461	6.687	6.982
5.846	5.793	5.778	5.796	5.905	6.176	6.408	6.635	6.939	
5.776	5.857	5.805		6.052	6.360	6.607	6.917		
5.792	5.831	5.907	6.054	6.340	6.590	6.905			
	5.954	6.179	6.360	6.590	6.904				
6.035	6.245	6.407	6.604	6.908					
6.333	6.462	6.638	6.919						
6.544	6.683	6.942							
6.777	6.984								
7.079									

Comparison of these results with the results of last line of the table 4.3 demonstrates that maximum discrepancy for neutron flux calculated by subdivision way 1 and subdivision way 2 is <0.1%.

Also these results have been analyzed in dependence from introduction of supplement zones in the FR. Every FR has been divided into two zones along radius. (Water surround FR is considered as one calculation zone.) There is 170 calculation zones in this way. Accuracy of the solution convergence concerning "symmetrical" FR is reached when 20039 trajectories is used; maximum discrepancy between the results obtained with subdivision way number 1 is <0. 01%..

This analysis leads to the conclusion that the solution obtained with using of first way subdivision and 20039 trajectories is as "basic solution" with accuracy 0.1%. Calculation time for this variant is 2.08 min.

2.2 Assembly with 18 removed absorber rods.

Also one-group transport problem has been analyzed for VVER-1000 type reactor assembly when the absorber rods are removed. The space inside the guide tubes are filled with water. One-group cross sections are chosen accordingly the table 2.1. It is possible to assume in advance that the neutron flux gradients are less in this problem than in the first one (paragraph 2.1) so it is needn't to analyze more precision subdivision than the subdivision number 1 (118 calculation zones). The accuracy of solution convergence has been analyzed versus trajectories number too. Choose of trajectories number is given in the table 2.5 .

Table 2.5 Choose of trajectories number for the assembly when AR are removed

Line's number	1	2	3

Korobov's number	10007	8191	2129

Calculation results-neutron flux density distribution over the FR of the symmetry angle of the assembly are given in the table 2.6

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Table 2.6 Neutron flux density distribution over FR of the assembly with removed AR

7.383 7.390 7.382	7.374 7.357 7.314		7.309 7.288 7.296	7.248 7.242 7.248	7.173 7.176 7.226	7.002 7.000 7.004	6.991 6.979 6.976	7.108 7.117 7.117	7.383 7.384 7.372
7.312 7.336 7.357	7.344 7.299 7.280	7.283 7.278 7.294	7.278 7.267 7.296	7.195 7.214 7.220	7.035 7.033 7.049	6.994 6.989 6.990-	7.087 7.089 7.090	7.345 7.350 7.357	
7.301 7.304 7.278	7.201 7.202 7.196	7.255 7.249 7.212		7.138 7.148 7.128	7.019 7.025 7.023	7.095 7.091 7.096	7.332 7.333 7.336		
7.252 7.272 7.224	7.255 7.241 7.254	7.179 7.215 7.204	7.160 7.140 7.156	7.041 7.047 7.038	7.094 7.120 7.095	7.344 7.331 7.340			
	7.178 7.162 7.170	7.043 7.025 7.014	7.027 7.010 7.007	7.102 7.085 7.112	7.335 7.340 7.347				
7.121 7.111 7.137	6.999 6.997 7.003	6.997 7.000 7.001	7.100 7.098 7.092	7.335 7.345 7.359					
6.982 6.996 7.985	6.989 6.996 6.986	7.084 7.105 7.087	7.330 7.352 7.344						
7.010 7.009 7.028	7.111 7.105 7.114	7.348 7.349 7.357							
7.178 7.178 7.186	7.379 7.376 7.374								
7.465 7.472 7.468									

These results demonstrate that precision convergence of the solution for symmetric FR is reached to maximum discrepancy 0.3% when 8000 trajectories are used. The calculation time is 0.88min

2.3 The assembly with 18 AR and 12 FR with U-Gd fuel

Also one-group problem for VVER-type reactor assembly with 12 FR that contains 8% Gd_2O_3. One-group cross sections for five materials are supposed the same ones that are

in the table 2.1.The main approximations of the transport equation are the same too. Strong flux gradients must be in the FR with U-Gd fuel as well in the FR surrounded them, so it is necessary to analyze carefully the results in dependence from calculation zones subdivision way .First subdivision way is described in p.2.1, i.e. one calculation zone over every FR and one calculation zone over all water hexagonal cell surrounded every FR as well AR. When this way is used the accuracy of the solution convergence for "symmetrical" rods is reached 0.1% with 20000 trajectories of numerical integration.

Also the second way of calculation zones subdivision is analyzed for this assembly. Just every hexagonal water zone surrounded every rod has been divided into 6 supplementary zones and every rod with U-Gd fuel has been divided into 4 zones. There is 400 calculation zones when this way is used. Accuracy of the solution convergence for "symmetrical" rods has been reached to 0.1% with 50 thousand of trajectories.. Comparison of the results corresponded these subdivision ways are given in table 2.7 : line number 1 corresponds the initial subdivision; line number 2 corresponds more detail subdivision.

Table 2.7 Flux density distribution over the FR for the assembly with U-Gd rods for different subdivision ways

5.695 5.706	5.376 5.392		5.185 5.208	5.290 5.310	5.483 5.496	5.796 5.807	6.018 6.026	6.207 6.214	6.396 6.398
5.314 5.333	5.086 5.122	5.058 5.095	5.230 5.250	5.365 5.383	5.629 5.645	5.883 5.896	6.128 6.135	6.355 6.355	
5.063 5.100	0.270 0.272	5.078 5.115		5.278 5.313	5.574 5.607	5.997 6.012	6.302 6.305		
5.088 5.123	5.124 5.157	5.310 5.330	5.270 5.305	0.291 0.294	5.758 5.788	6.227 6.232			
	5.444 5.458	5.611 5.630	5.569 5.601	5.757 5.789	6.192 6.200				
5.594 5.606	5.786 5.799	5.878 5.890	5.994 6.007	6.227 6.234					
5.913 5.922	6.016 6.025	6.127 6.136	6.302 6.306						
6.108 6.116	6.205 6.211	6.355 6.358							
6.278 6.283	6.397 6.398								
6.451 6.451									

These results demonstrate that the influence of supplementary subdivision is more essential: the flux density discrepancy is reached to 0.8%. Calculation time of the variant with detail subdivision is 8.01 min

As a result of one-group calculations of different assembly types is that FCPM is able to get the basic (converged with accuracy 0.1% for symmetrical rods) solution. Calculation time depends of problem type, subdivision way and trajectory number .Table 2.8 demonstrates this dependence

Table 2.8 Comparison analysis of time expenditure when the basic solution are obtained for different assemblies VVER-1000 with FCPM

Problem type	Subdivision way	Trajectories number	Calculation time, min.
Assembly with removed AR	1	8 thousand	0.88
Assembly with inserted AR	1	20 thousand.	2.08
Assembly with U-Gd rods	2.	50 thousand	8.01

3.MULTIGROUP CALCULATION VVER-1000 ASSEMBLY ANALYSIS

3.1 Iteration procedure for the solution in the thermalization region

Multigroup eigenvalue problem is analyzed for VVER-1000 type reactor assembly too. Boundary condition are formulated in the form either mirror reflection or escape condition (black surface).

The iteration procedure of fission source is used.(Bell, Glasstone, 1974) The group transport equation with isotropic scattering is solved by the first collision probabilities method (FCPM) . The transport approximation is used to take into account anisotropic scattering. The group cross sections are calculated with using the library of nuclear dates of WIMS-code .

The integral transport equation in the thermalization region is reduced to the linear algebraic system. The FCPM equations in the thermalization region when flat flux approximation and isotropic scattering in the laboratory coordinate system approximations are valid are as following:

$$\tilde{\Phi}(l,j)=\sum_{j'=1}^{N_R} S(j',l)P(j'/\,j,l) \tag{4}$$

$$S(j',l)=\sum_{l'=1}^{N_G} \tilde{\Phi}(l',j')H(l',/l,j')+S^{inn}(l,j') \tag{5}$$

$$\tilde{\Phi}(l,j)=\Sigma_t(l,j)V(j)\Phi(l,j) \tag{6}$$

$$H(l'/l,j')=\Sigma_s(l'/l,j')/\Sigma_t(l',j') \tag{7}$$

Where N_G is energy group number; N_R is calculation zones number; $\Phi(l, j)$ is neutron flux in the group l in the calculation zone j; V(j) is volume of calculation zone j; $\Sigma_t(l, j)$ is total cross section of the neutron of the group l with the material of calculation zone j; $\Sigma_s(l'/l, j)$ is the neutron transmission matrix from group l' to the group l in calculation zone j; S^{inn}(l,j) is internal sources in the group l and in the zone j which are epithermal neutrons slowing down in this zone

This system of the linear algebraic equations is solved by iteration way with using of the spatial normalization at every iteration step. The normalization coefficients are got through the averaging of the linear equation system over group index. The obtained system is a system of N_R-dimension, and it is possible to solve it through direct of matrix inversion. The formulas for spatial normalization are as following:

$S_0(j,l) = S(j,l)N(j)$ - this is normalization source value

The correct multiplication factors $N(j)$ are obtained from the following equation:

$$N(j) = \sum_{j'=1}^{N_R} N(j')P(j'/j) + F^{inn}(j) \quad (8)$$

Here

$$P(j'/j) = \frac{1}{S(j)} \sum_{l=1}^{N_G} S^{inn}(l, j)P(j'/j, l)\frac{\Sigma_s(l, j)}{\Sigma_t(l, j)}; \quad (9)$$

$$F^{inn}(j) = \sum_{l=1}^{N_G} S^{inn}(l, j)/S(j); \quad (10)$$

$$S(j) = \sum_{l=1}^{N_G} S(j, l) \quad (11)$$

3.2 Comparison calculation results analysis for benchmark-tests

There are publications concerning description of the international calculation tests for VVER-1000 type nuclear reactors.(Lazarenko, 2000). In this work the detail description of the set of the fuel assemblies of VVER-1000 type nuclear reactor. These assemblies are different each from other with the FR composition, the fuel composition (U, Pu, Gd), moderator composition, the temperature, and Xe and Sm concentration. The output calculation date are defined and comparison them with the results obtained with different codes are given.

To estimate the ability of these based on FCPM algorithm and code to analyze this described benchmark-tests it is necessary to have the way of group cross sections preparation. For this purpose the method, algorithm and library of the nuclear data of WIMS-code has been used. The group cross sections are obtained from the calculation

analysis of the set of different kinds of elementary cells with the definite boundary conditions. In this paper the calculation analysis results of one of the benchmark-tests-variant 1, state S2 -are given. This variant has the same geometrical and material parameters that the assembly described in paragraph 2 part 2, i.e. there are 312 FR with U-235 reached 3.7% and 19 cells with guide tubes filled with water in this assembly. It is necessary to prepare the group cross section for 3 types of the elementary cell: the cell with FR, the cell with water (when AR is removed), and the central tube. For cells without any fission materials the group cross sections have been prepared with consideration of the closely next cells that contain the fission materials. For the water next to the assembly boundary the group cross sections are suppose to be the same that for the cell without any fission materials.

Procedure described in the 3.1 is used for solving of multigroup eigenvalue problem in the assembly volume. Calculation analysis has been carried out in 17 groups ;number of trajectories are chosen from the results of appropriate one-group source problem that has been formulated as "attached" to this benchmark-test. (p.2.2) If these results are available it is possible to estimate the calculation discrepancy on the stage of the transport calculation solution of spatial neutron distribution with definite cross section versus one or another subdivision way. The output calculation results $-K_{inf}$ and energy distribution over the assembly FR are given in the table 3.2. The precision of calculation convergence analysis versus trajectories number and subdivision way is carried out. This table also is given as symmetry angle of the assembly (see p.2.1). The results obtained with TVS-M-code are given on line number 1. The results obtained with the method described here when calculation zone subdivision has been carried out by the way number 1 (see p.2.3) and the trajectories number is ~ 10 thousands are given on line 2; the results obtained with the same method and the same calculation zone subdivision and the trajectories number is ~20 thousands are given on line 3; the results obtained with the same method when the calculation zone subdivision number has been carried out by the way number 2 and the trajectories number is 20 thousand are given on line 4 The output calculation results for K_{inf} are given in table 3.1

Table3.1 K_{inf} obtained for variant 1 state 2 with different codes. *

Code name	TVS-M	APOLLO-2*	WIMS-FCPM**
K_{inf}	1.2858	1.2939	1.2923

*) APOLLO-2 see (Sanchez, 1987)
**) WIMS-FCPM is work name for code based on the methods and algorithms described here.

Table 3.2 Energy distribution over FR obtained with code TVS-M (line 1) and with code (WIMS-FCPM (different spatial approximations correspond the lines 2-4)

1.051	1.047		1.034	1.023	1.003	0.961	0.955	0.977	1.036
1.038	1.035		1.022	1.021	1.004	0.967	0.964	0.979	1.019
1.038	1.034		1.026	1.020	1.004	0.971	0.965	0.982	1.023

1.038	1.034		1.026	1.020	1.004	0.971	0.965	0.982	1.023
1.045 1.034 1.033 1.033	1.038 1.029 1.027 1.028	1.032 1.019 1.024 1.024	1.026 1.022 1.022 1.022	1.011 1.012 1.010 1.010	0.969 0.973 0.977 0.977	0.957 0.963 0.967 0.967	0.973 0.980 0.979 0.979	1.028 1.020 1.019 1.019	
1.036 1.027 1.027 1.028	1.011 1.010 1.008 1.008	1.023 1.012 1.020 1.020		0.995 1.000 0.998 0.998	0.962 0.974 0.973 0.972	0.973 0.979 0.980 0.980	1.025 1.015 1.018 1.018		
1.027 1.023 1.021 1.021	1.019 1.023 1.016 1.016.	1.010 1.005 1.009 1.009	0.995 0.998 0.998 0.998	0.966 0.971 0.977 0.974	0.974 0.978 0.981 0.981	1.024 1.022 1.018 1.018			
	1.002 1.009 1.003 1.003	0.969 0.981 0.978 0.978	0.962 0.980 0.972 0.972	0.974 0.982 0.982 0.981	1.024 1.018 1.018 1.018				
0.992 0.991 0.996 0.996	0.961 0.968 0.971 0.970	0.957 0.973 0.967 0.967	0.973 0.981 0.980 0.980	1.024 1.014 1.018 1.018					
0.955 0.968 0.968 0.966	0.955 0.963 0.966 0.965	0.973 0.977 0.980 0.979	1.025 1.018 1.018 1.018						
0.959 0.971 0.968 0.968	0.977 0.982 0.981 0.981	1.028 1.018 1.019 1.019							
0.991 0.992 0.992 0.991	1.036 1.028 1.023 1.023								
1.059 1.042 1.038 1.038									

Also the calculation analysis of the influence of different group subdivision ways on output results has been carried out. The table 3.3 shows this analysis. There are the energy distribution over FR obtained with the TVS-M-code (line 1) and with WIMS-FCPM-code. The latter is given in different group subdivision: 17 groups (9 slowing down and 8 thermal) -line 2;. and 10 groups (9 slowing down and 1 thermal) -line 3. The thermalization boundary-2.6 ev. ..

Table 3.3 Energy distribution over FR obtained with code TVS-M (line 1) and with code WIMS-FCPM (different energy approximations correspond the lines 2,3)

1.051 1.036 1.025	1.047 1.033 1.022		1.034 1.025 1.016	1.023 1.020 1.012	1.003 1.004 1.002	0.961 0.967 0.983	0.955 0.964 0.980	0.977 0.982 0.990	1.036 1.023 1.012
1.045 1.032 1.021	1.038 1.026 1.018	1.032 1.023 1.016	1.026 1.021 1.013	1.011 1.010 1.006	0.969 0.978 0.987	0.957 0.963 0.981	0.973 0.980 0.988	1.028 1.020 1.009	
1.036 1.027 1.017	1.011 1.010 1.007	1.023 1.019 1.012		0.995 0.998 0.998	0.962 0.973 0.984	0.973 0.979 0.989	1.025 1.018 1.008		
1.027 1.021 1.013	1.019 1.016 1.010	1.010 1.009 1.005	0.995 0.998 0.998	0.966 0.975 0.986	0.974 0.982 0.989	1.024 1.018 1.008			
	1.002 1.003 1.001	0.969 0.978 0.987	0.962 0.973 0.984	0.974 0.982 0.989	1.024 1.018 1.008				
0.992 0.996 0.997	0.961 0.971 0.983	0.957 0.968 0.981	0.973 0.980 0.988	1.024 1.018 1.008					
0.955 0.967 0.980	0.955 0.963 0.980	0.973 0.977 0.988	1.025 1.018 1.008						
0.959 0.969 0.982	0.977 0.982 0.989	1.028 1.019 1.009							
0.991 0.992 0.995	1.036 1.023 1.012								
1.059 1.038 1.019									

4. CONCLUSIONS

In this paper the results of calculation analysis of one-group and multigroup transport equations solutions are given for the VVER-1000-type nuclear reactors assemblies. These results lead to following conclusions

FCPM and algorithm and code based on this method are able to get the precision solution for one-group problem with source and the definite cross-section. The optimal calculation zones subdivision is given for specific assembly type.

- For the assembly with 18 AR the calculation time of getting the precision solution (precision <0.1%) is 2.08 min
- For the assembly with 18 removed AR the calculation time of getting the precision solution is 0.88 min
- For assembly with 18 AR and 12 U-Gd FR the calculation time of getting the precision solution is 8.01 min

The analysis of the results of multigroup calculation leads to the conclusion that spatial subdivision for precision one-group solution (one reg. zone for every hexagonal water, one reg. zone for every FR) provides 1% calculation precision for energy distribution; .20 thousands trajectories provides the precision of "energy distribution symmetry convergence" <0.1%

The detail group subdivision in the thermalization region provides the calculation precision energy distribution ~1%

NOMENCLATURE

AR	absorption rod
FR	fuel rod

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REFERENCES

Askew J.R, Fayers F..J. and Kemshell P.B 1966 A general description of the lattice code WIMS *J. British Nuclear Energy Society*, 5, p.564-585

Bell G.J , Glasstone S , 1974 *Nuclear Reactor Theory* " Moscow, Atomizdat, 1974

Lazarenko A., .Kalugin M. and Bychkov S., 2000 Benchmark Calculations for VVER-1000 Fuel Assemblies Using Uranium or Mox Fuel In: *PHISOR 2000* May 7-12, Pittsburg PA, USA

Sanchez R., Mondot J., Stankovki Z, Cossc A., Zmijarevic I (1987) Apollo-2: A user-oriented portable, modular code for multi-group assembly calculations. In: Proc.Int. Topl. Mtg. *Advancess in Reactor Physics, Mathematics and Computation*, Paris, France, April 27-30, 1987,

Shevelev Ya. V., PoveschenkoT.S 1986 Polynomial approximation of the neutron flux in the collision probabilities method, *Atomnaya energiya*, 48, 2, 80. .

Sobol I.M, Statnikov R.B, 1981 *Vybor optimalnyh parametrov v zadachah so mnogimi kriteriiami" (Choose of the optimization parameters in the problems with many criteria)* Nauka, Moscow, 1981

Poveschenko T.S., 1999 Definition of calculation 2-D benchmark problem for VVER-1000 assembly neutron flux calculations In: Aragones, J.M (Ed.), *Mathematics and Computation Reactor Physics and Environmental Analysis in Nuclear Applications* Vol 2, September, 1999 Madrid, Spain, p. 1756