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THE RESULTS OF COMPARISON OF BENCHMARK CALCULATIONS ANALYSIS FOR VVER FUEL CELLS AND ASSEMBLIES WITH DIFFERENT METHODS

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ABSTRACT

The results of comparison of the neutron-physical analyses of the cells and the assemblies of the VVER-1000 type reactor -multiplication factor and pin power distribution are given. These results have been got with neutron transport multi-group code GEFCOP which is based on the General First Collision Probabilities Method. This code is connected with two different nuclear data libraries: the library of WIMS-code and the library of UNK-code. These codes are based on different nuclear data libraries and on the different methods to take into account spatial shielding of neutron flux; and algorithm of UNK-code is based on the more detail (about 7000 group) neutron spectra calculation analysis. The results have been compared between each other and verified with Monte-Carlo-codes results for different types of fuel assemblies.

Key Words: fuel assembly, absorption rod, spatial shielding, neutron transport calculation

1. INTRODUCTION

Necessity to increase optimum and safety conditions of exploitation of modern nuclear reactors causes stronger requirements to precision of neutron-physical analysis
Common tendency to get more precision characteristics of nuclear reactor cell is connected with increasing of accuracy of neutron-physical calculation analysis with more working out in detail. Neutron transport multigroup calculation analysis consist of two main stages: generation of group cross sections from nuclear data libraries and calculation method for solution of the neutron transport equation with the definite group cross section in the system under consideration.
In the most of modern nuclear reactor neutron transport cell codes (WIMS[1], APOLLO[2]) neutron spectra calculation analysis is carried out (as a rule) in 100 energetic groups. The methods are used are based either on the equivalence theorem or subgroup method. The both methods needs initial information about neutron spectra next to resonanse regions i.e. they need special technique for every kind of the cell (especially for unstructed macrocells of modern reactors) and they are not as universal way for group cross section generation over all energetic neutron spectra. So it seems like perspective way some alternative methods for group cross section generation based on the detail neutron spectra calculation analysis with accurate taking

into account both energetic shielding factors and spatial shielding factors. One of such advanced method is suggested in the paper [3]. It is based on the introducing of fine energetic grid in the resolved resonance region (so called "microgroups") and it allows to avoid necessity to use any self shielding correction of the group cross sections. Also UNK-code is described and module MACSEC provides the group cross sections for 89 groups for all energetic region. This code based on ENDF/B nuclear data library.

Second stage of the neutron transport calculation analysis is solution of the transport equation in considered value with definite cross sections. In the paper [4] the verification results of multi-group transport neutron-physical calculation analysis are performed for VVER-1000 type reactor cell and fuel assemblies with GEFCOP-code. This code based on the general first collision probabilities method (GFCPM). Module of combinatorial geometry SCG-5 [5] is used to describe system geometry. It allows to trake into account unstructed meshes in 2-D geometry. The results have been verified in dependence on two parameters of the solution convergence. First parameter is connected with subdivision way and it is number of calculation zones. Second one is the trajectories number and it is responsible just for the convergence of FCP when subdivision of the system is defined. The calculation analysis results have been compared with ones obtained with another codes. To get group cross section the cell code WIMS has been used (module PINCELL). This module generates 69 group cross section from UKNDL and FOND2[ ] nuclear data libraries and then averages them to necessary number of the groups.The elementary cells and different assembly kinds of the VVER-1000 type reactor have been verified. To analyse these results one could conclude that although the calculation results -K_{inf} and pin power distribution for elementary cell and the assembly with slow neutron flux change are in good agreement with the results obtained with another codes, but there is not good agreement if an assembly contains absorption rods and neutron flux has strong gradient . It seems like this discrepancy appears because of both bad spatial and spectra shielding taking into account and insufficient spatial subdivision is used. So it would be very actual to analyze the assemblies contained strong neutron flux gradient with another group sections library that would allow more accurate to take into account spatial and spectra neutron flux shielding.

Topic: Codes and Benchmarks

2. .THE REZULTS OF NEUTRON-PHYSICAL CALCULATION ANALYSIES OF CYLINDRICAL CELL

In the paper [7] benchmark calculations are performed for VVER-1000 LEU and MOX fuel assemblies. Four different assemblies were specified. These hexagonal assemblies consist of 312 fuel cells and 19 cells with tubes containing moderator or absorber rods. Geometrical and material parameters of the cell for different temperature values and moderator density values were specified. In this paper one of this state is analyzed with two different methods. First method is based on the advanced library of WIMS-code and it consist of the initial library of WIMS-code UKNDL and FOND2. Neutron transport calculations are carried out with multigroup 2-D transport code GEFCOP based on the General First Collision Probabilities Method (GFCPM). Second method is based on the group library of UNK-code. This library is based at the ENDF/B nuclear data. For preparing of group cross section the method of accurate taking into account of the energetic and spatial shielding is used. This method allows to take into account specific features of neutron spectra especially in the macrocells of modern nuclear

reactors. Then neutron transport calculation analysis has been carried out with the same transport code GEFCOP and the approximation parameters in this scheme were the same that for first one. So the comparison of different calculation methods for elementary cell allows to estimate the order of uncertainty caused of just nuclear data libraries. The geometrical and material parameters of the cell are given in the table I.

Table I.Geometrical and material parameters of the cylindrical cell.

	Fuel	Clad	Moderator
Radius, см.	0.386	0.5482	0.6694
Temperat., K degr	1.027E+03	575E+02	575E+02
Nuclear densities	235-U 8.626 E-04 238-U 2.217 E-02 O 4.606 E-02	Zr 4.259 E-02 Hf 6.597E-06 Nb 4.225E-04	H 4.843E-02 O 2.422 E-02 11-B 1.942 E-05 10-B 4.794 E-06

The equivalent cylindrical cell has been considered, eigenvalue problem has been analyzed, and "white" reflection at the boundary has been used. To check the results the transport calculation analysis with WIMS- code have been compared and the PIJ-option has been used. The calculation results are given in table II

Table II.The calculation results of K_{inf} for cylindrical cell with different methods .

Name of method	WIMS-PIJ	WIMS-GEFCOP	UNK-GEFCOP
K_{inf}	1.2635	1.2635	1.26155

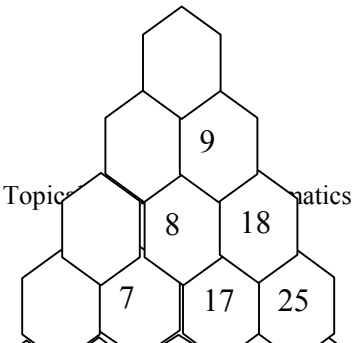
Comparison of these results shows that the uncertainty in the multiplication factor for cylindrical cell caused with using of the different libraries of nuclear data is about 0.15%

3. THE RESULTS OF NEUTRON CALCULATION ANALYSIS OF THE DIFFERENT TYPES OF VVER-FUEL ASSEMBLIES

In this part of the paper the comparison of calculation analysis has been carried out for different types of the fuel assemblies of VVER-1000 type reactor. Figure 1 is the rotation symmetry angle for typical reactor assembly. It contains 52 rods with uranium fuel and 3 guide tubes. One of variant being analyzed is just when these tubes are filled with moderator. Second one is just variant of fuel assembly when the absorption rods are in these guide tubes. Third variant is fuel assembly with 50 uranium fuel rods and 2 uranium-gadolinium fuel rods, guide tubes are filled with moderator. Also two methods have been used for calculation analysis of these fuel assembly benchmark tests.

First one is neutron transport multigroup calculation analysis with GEFCOP module. Group cross section for this method are generated with WIMS-code. It uses the combination of two nuclear data library: UKNDL and FOND2. The group cross section are generated with PINCELL module for 4 types of the cell: the cell with fuel rod (uranium and uranium-gadolinium), the central tube cell, the cell with absorption rod and the cell with guide tube filled with the

moderator (water). For the cells which don't contain the fuel material the surround configuration has been taken into account.
GEFOP is module of multigroup transport calculation. In the paper [4] the mathematical benchmark-tests for VVER-1000 fuel assemblies are described and analyzed with GEFCOP. The results accuracy depends on the accuracy of the calculation of the FCP and spatial subdivision of the considered system. The convergence of the results has been analyzed in dependence on the trajectories number for FCP calculations and spatial subdivision way. Different ways of subdivision of the elementary cell are performed in fig.2. Analysis of the results of these mathematical benchmark-test allows to estimate the uncertainties of just neutron transport method
Second method for neutron-transport calculation analysis of the benchmark-tests was UNK-GEFCOP. As it have already been mentioned UNK code uses its own method for generation of the group cross section. It based on the ENDF/B-VI files of nuclear library data, but neutron spectra calculations is carried out in detail division of energetic region without any approximations like equivalence theorem or subgroup approximation. There is fine calculation grid supposed to be introduced to any energetic group and it is about 7000 point . Purpose of using of this grid is to have possibility of direct neutron spectra calculations in the resolving resonance region without any "traditional" ways to take into account space shielding of cross sections before.
The group cross section thus generated are used in the same neutron-physical transport calculation module GEFCOP and just the same approximations for FCP and spatial subdivision are used. Thus when the comparison of these two methods for calculation analysis of an assembly benchmark is carried out the uncertainties caused with spatial shielding can be estimated.



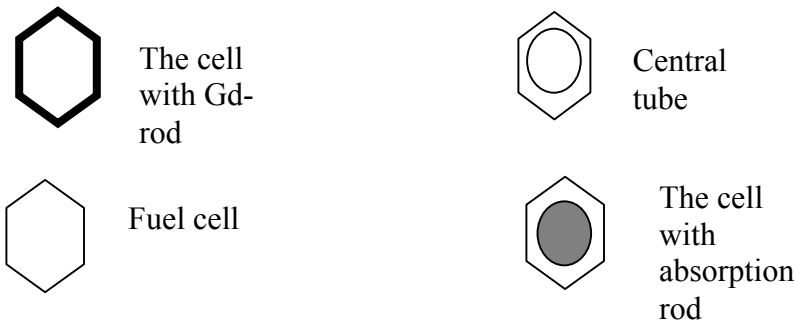
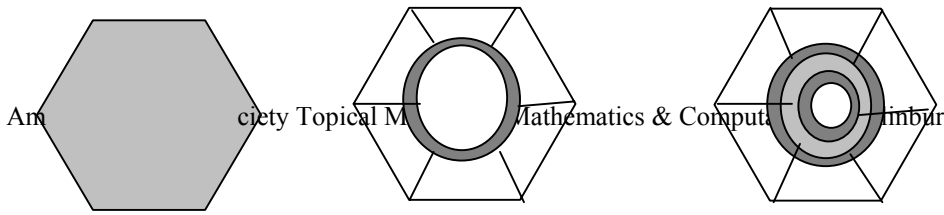
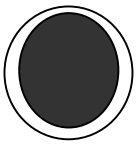


Figure 1.The VVER-1000 assembly (60 degree , rotation symmetry)





The calculation scheme number1

The calculation scheme number 2

The calculation scheme number 3

Figure 2 Different calculation scheme for cell calculation analysis..

3.1 Comparison of neutron-physical calculation analysis of the assembly with 18 rods filled up with moderator.

In this part the results of benchmark test for fuel assembly with 52 uranium fuel rods and 3 guide tubes filled with moderator are performed. In the paper [7] this variant is called as variant 1 state2 and it's specification is available. Approximation parameters (spatial subdivision and trajectories number) for convergence of solution in GEFCOP code are chosen accordingly the results of one-group benchmark-test analysis considered in [4] .There is not strong neutron flux gradients in this benchmark-test, and the way of spatial subdivision was chosen as the way number 1 on figure 2. It is 118 calculation zones because of symmetry properties are taken into account. Also for calculation of the FCP is used about 20 thousands of trajectories. The calculation results of K_{inf} are given in table III. . The comparison of calculation analysis of pin power distribution are given in Table IV. The accordance between the elementary cell of the assembly (see figure1) and the secular of the table is shown with number in brackets (see table IV).

Table III. The results of calculation analysis of K_{inf} of VVER-1000assembly with different methods

Name of the method.	UNK-GEFCOP	WIMS-GEFCOP	TVS-M
Kinf	1.2823	1.2923	1.2858

The comparison of these results shows good agreement between 3 codes , but it should be mentioned that the difference between the K_{inf} caused with different nuclear data library and different spatial shielding methods is 0.7% for this variant.

Table IV Pin power distribution for VVER 1000 - assembly with different methods
Line 1-TVS-M
Line 2 UNK-GEFCOP

Line 3 WIMS GEFCOP

(1) 1.051 1.049 1.038	(2) 1.047 1.044 1.034		(3) 1.034 1.034 1.026	(4) 1.023 1.026 1.020	(5) 1.003 1.006 1.004	(6) 0.961 0.964 0.971	(7) 0.955 0.957 0.965	(8) 0.977 0.977 0.982	(9) 1.036 1.026 1.023
(10) 1.045 1.043 1.033	(11) 1.038 1.036 1.028	(12) 1.032 1.032 1.024	(13) 1.026 1.028 1.022	(14) 1.011 1.014 1.010	(15) 0.969 0.973 0.977	(16) 0.957 0.960 0.967	(17) 0.973 0.974 0.979	(18) 1.028 1.021 1.019	
(19) 1.036 1.036 1.028	(20) 1.011 1.012 1.008	(21) 1.023 1.026 1.020		(22) 0.995 0.997 0.998	(23) 0.962 0.966 0.972	(24) 0.973 0.975 0.980	(25) 1.025 1.019 1.018		
(26) 1.027 1.028 1.021	(27) 1.019 1.021 1.016.	(28) 1.010 1.012 1.009	(29) 0.995 0.997 0.998	(30) 0.966 0.969 0.974	(31) 0.974 0.977 0.981	(32) 1.024 1.020 1.018			
	(33) 1.002 1.004 1.003	(34) 0.969 0.973 0.978	(35) 0.962 0.966 0.972	(36) 0.974 0.977 0.981	(37) 1.024 1.020 1.018				
(38) 0.992 0.995 0.996	(39) 0.961 0.964 0.970	(40) 0.957 0.960 0.967	(41) 0.973 0.975 0.980	(42) 1.024 1.020 1.018					
(43) 0.955 0.958 0.966	(44) 0.955 0.958 0.965	(45) 0.973 0.974 0.979	(46) 1.025 1.020 1.018						
(47) 0.959 0.961 0.968	(48) 0.977 0.976 0.981	(49) 1.028 1.021 1.019							
(50) 0.991 0.989 0.991	(51) 1.036 1.026 1.023								
(52) 1.059 1.044 1.038									

Analysis of these results shows that there is good agreement for all of these codes because of there are not strong flux gradients in this benchmark test. Maximum of the difference is for the pin number 52 between the WIMS-GEFCOP method and UNK code and it is about 2%. It

should be noted the difference in the K_{inf} value caused of different nuclear data libraries (WIMS-code and UNK-code) is 0.7%.

3.2 Neutron-physical calculation analysis of the fuel assembly with 18 absorption rods

In this part the results of benchmark test for fuel assembly with 52 uranium fuel rods and 3 guide tubes (in symmetry rotation angle) filled with absorption rods in these tubes are performed. In the paper [6] this variant is called as V11S2 and it's specification is available. The calculation results-K_{inf} and pin power distribution -obtained with Monte-Carlo code MCU are available. The results obtained with UNK-GEFCOP method are compared with ones obtained with MCU-code. Approximation parameters for neutron transport analysis (spatial subdivision and trajectories number for convergence of solution in GEFCOP code) are chosen accordingly the results of one-group benchmark-test analysis considered in [4] .There are strong neutron flux gradients in this benchmark-test, and the way of spatial subdivision was chosen as the way number 2 on figure 2. It is 396 calculation zones because of symmetry properties are taken into account. Also for calculation of the FCP is used about 50 thousands of trajectories. The calculation results of K_{inf} are given in table V . The comparison of calculation analysis of pin power distribution are given in Table VI. The accordance between the elementary cell of the assembly (see figure1) and the secular of the table is the same than in part 3.1

Table V The results of calculation analysis of K_{inf} of VVER-1000assembly with different methods

Name of the method.	UNK-GEFCOP	MCU
Kinf	1.0339	1.0327

Table VI Pin power distribution for VVER 1000 assembly with different methods
Line 1 UNK-GEFCOP
Line 2 Deviation (%) UNK-GEFCOP of MCU

0.8371.--1.34	0.737 2.6		0.727 5.2	0.757 .2	0.819 4.1	0.995 -0.6	1.108 -2.1	1.187 -2.1	1.271 -1.6
0.722 3.3	0.706 4.3	0.702 4.9	0.743 5.3.	0.792 5.1	0.951 0.4	1.079 -2.0	1.169 -2.5	1.260 -1.231	
0.711 5.0	0.756 3.4	0.752 4.8		0.883 3.7	1.048 -1.3	1.153 -2.5	1.252 -2.8		
0.740 5.4	0.757 5.0	0.806 4.8	0.874 3.7	1.032 -0.8	1.143 -2.6	1.247 -3.3			
	0.822 4.3	0.953 0.3	1.048 -1.3	1.143 -2.6	1.245 -3.4				
0.872 3.2	0.996 -0.7	1.079 -2.0	1.154 -2.6	1.247 -3.1					
1.048 -1.5	1.109 -2.3	1.169 -2.5	1.252 -2.8						
1.143 -2.2	1.187 -2.1	1.260 -2.3							
1.214 -1.7	1.271 -1.6								
1.297 -1.1									

Analysis of these results shows that there is good agreement for K_{inf} obtained with both codes. But there are quite strong deviations in pin power distribution value especially in the fuel cells next the cells with absorption rods because of strong flux gradient arises just here. Maximum of the deviation is in the pin number 26 and it is 5.42%. It should be noted the deviation between the results obtained with WIMS-GEFCOP code and MCU-code reaches to 10% . It seems like there are two reasons for this descreapance:1) bad method for group cross sections spatial shielding in resonance region ; 2) insufficient detail subdivision of the system especially next to the assembly boundary

3.3 Neutron-physical calculation analysis of the assembly with 18 absorption rods and 12 U-Gd fuel rods.

In this part the results of benchmark test for fuel assembly with 50 uranium fuel rods,2 uranium-gadolinium fuel rods and 3 guide tubes (in rotation symmetry 60 degree) filled with a moderator are performed. In the paper [7] this variant is called as Variant 3,state2 and it's specification is available. The calculation results-K_{inf} and pin power distribution -obtained with different codes are available.
The results obtained with UNK-GEFCOP method are compared with ones obtained with MCU-code and others. Approximation parameters for neutron transport analysis (spatial subdivision and trajectories number for convergence of solution in GEFCOP code) are chosen accordingly the results of one-group benchmark-test analysis considered in [4] . There are strong neutron flux gradients in this benchmark-test, and the way of spatial subdivision was chosen as the way

number 2 on figure 2. It is 400 calculation zones because of symmetry properties are taken into account. Also for calculation of the FCP is used about 50 thousands of trajectories. The calculation results of K_{inf} are given in table VII. The comparison of calculation analysis of pin power distribution are given in Table VIII. The accordance between the elementary cell of the assembly (see figure1) and the secular of the table is the same than in part 3.1

Table VII The results of calculation analysis of K_{inf} of VVER-1000assembly with different methods

Name of method	UNK-GEFCOP	MCU-REA	APOLLO-2
K_{inf}	1.1801	1.1757	1.1804

Table VI Pin power distribution for VVER 1000 with different methods
Line 1 UNK-GEFCOP
Line 2 Deviation (%) UNK-GEFCOP of MCU

1.069 0.84	1.057 -0.76		1.043 -0.49	1.048 -1.24	1.043 -0.96	1.013 -0.89	1.010 0.30	1.025 1.76	1.052 5.4
1.053 -1.61	1.021 -3.6	1.022 -4.3	1.044 -2.47	1.040 -1.7	1.009 -1.8	1.003 -1.1	1.020 0.68	1.056 4.0	
1.015 -4.8	0.319 0.3	1.000 -4.3		0.989 -4.0	0.979 -4.4	1.011 -1.0	1.052 3.0		
0.986 -5.3	0.990 -5.2	1.001 -2.7	0.978 -4.6	0.320 0.20	0.980 -4.0	1.045 1.8			
	1.036 -1.7	1.007 -2.8	0.982 -4.3	0.989 -4.2	1.042 1.2				
	1.038 -0.9	1.011 -1.0	1.002 -1.8	1.010 -0.5	1.045 2.2				
1.012 -0.7	1.010 -0.2	1.020 1.0	1.052 3.1						
1.014 1.5	1.025 1.9	1.056 4.4							
1.035 3.1	1.080 5.7								
1.083 5.6									

Analysis of these results shows that there is good agreement for K_{inf} obtained with different codes. But there are quite strong deviations in pin power distribution value especially in the fuel cells next the cells with uranium-gadolinium rods because of strong flux gradient arises just here. Maximum of the deviation is in the pin number 52 and it is 5.6%. It should be noted the deviation between the results obtained with WIMS-GEFCOP code and MCU-code rich to 10% for this benchmark-test.

4.CONCLUSION

The results of benchmark calculations for the cells and different types of fuel assemblies of VVER-1000 type reactor obtained with neutron transport code GEFCOP and two different nuclear data libraries are performed . Analysis of these results leads to some conclusions.

- For cylindrical cell uncertainty in the multiplication factor caused with using of the different libraries of nuclear data is about 0.15 %
- There is good agreement of pin power distribution results obtained with 3 codes for assembly with 52 fuel rods and 3 guide tubes filled with moderator. Maximum of the difference is for the pin number 52 between the WIMS-GEFCOP method and UNK code and it is about 2%.
- Difference in the K_{inf} value caused of different nuclear data libraries (WIMS-code and UNK-code) is 0.7% for the assembly with52 fuel rods and 3 guide tubes filled with moderator.
- There is good agreement for K_{inf} obtained with MCU code and UNK-GEFCOP method for assembly contained both absorption rods and uranium-gadolinium fuel rods.
- The deviation between pin power distribution obtained with Monte-Carlo code and with UNK-GEFCOP code for the assemblies with both absorption rods and uranium-gadolinium fuel rods is much more less than the deviation between these results obtained with Monte-Carlo-code and WIMS GEFCOP code.
- .There are quite strong deviations in pin power distribution value especially in the elementary fuel cells next the cells with absorption rods . Maximum of the deviation is in the pin number 26 and it is 5.42%.
- There are quite strong deviations in pin power distribution value especially in the fuel cells next the cells with uranium-gadolinium rods. Maximum of the deviation is in the pin number 52 and it is 5.6%.
- It seems two reasons for bad pin power distribution approximation: 1) bad method for group cross sections spatial shielding in resonance region ; 2) insufficient detail subdivision of the system especially next to the assembly boundary.

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