ГЛОБАЛЬНАЯ ЯДЕРНАЯ БЕЗОПАСНОСТЬ, 2018 №1(26), С. 87–95

ЭКСПЛУАТАЦИЯ ОБЪЕКТОВ АТОМНОЙ ОТРАСЛИ

УДК 621.039:621.18

ОПТИМИЗАЦИЯ НЕЙТРОННЫХ ПАРАМЕТРОВ АКТИВНОЙ ЗОНЫ ВВЭР-1200 С ИСПОЛЬЗОВАНИЕМ КОДА WIMS-ANLS

© 2018 Юйбинь Сюй

Томский государственный университет, Томск, Томская обл., Россия

В статье рассматривается использование кода WIMS-ANL для получения различных значений и максимальных значений эффективного коэффициента размножения нейтронов (Кэф) путем изменения размера канала теплоносителя топливной сборки реактора (ВВЭР-1200) и концентрации топлива UO2 (от 1 до 30%).

Ключевые слова: ВВЭР-1200, WIMS-ANL, решетчатая ячейка, топливная сборка, нейтронные характеристики, реактивность, эффективный коэффициент размножения.

Поступила в редакцию: 12.02.2018

1. Introduction

1.1. Reactor type VVER(V491)-1200

1.1.1. VVER-1200 reactor plant

The design of AES-2006 of Generation 3+ with V-491 reactor plant is an evolutionary development of the designs with the VVER-1200 water cooled and water moderated reactor proved by a long-time operation. The AES-2006 design is based on the principle of safety assurance for the personnel, population and environment. The principle meets the requirements for the standards of radioactive substance releases into the environment and their content at normal operation, at anticipated operational occurrences including the design basis conditions (i.e. design events of Category 1-4) as well as at the beyond-design basis events during the entire service life of the nuclear power plant. One of the requirements during the reactor plant and process system design elaboration was not to reach the estimated value of a severe core damage 1.0E‑6 reactor/year and for the probability of emergency radioactivity release not to exceed 1.0E‑7 reactor/year. Level I and II PSA predict that these given values are not exceeded.

The VVER-1200 (V-491) design was developed by the Organization of General Designer “Atomenergoproekt” (St.Petersburg), Organization of General Designer of reactor plant, OKB Gidropress (Podolsk), with the scientific supervision of the RRC “Kurchatov Institute” (Moscow), in line with the Russian Regulatory Documents and considering the requirements of the IAEA and the European Utilities Requirements (EUR). The design was performed using the ISO 9001:2000 Quality Assurance International Standard. The main engineering solutions for reactor plants with VVER reactors have been corroborated by the experience of operating these installations for about 1400 reactor-years (decommissioned reactors included) considering about 500 reactor-years of operating the RPs with VVER-1200.

The safety concept of VVER-1200/V-491 considers the up-to-date world tendencies in

the field of NPP safety enhancement in order to meet the requirements for the NPP safety that are continuously made more stringent. The requirements for improving the economic efficiency of the NPP were also considered.

1.1.2. Reactor core and fuel design

The reactor cores contain 163 fuel assemblies (FA). The FAs are intended for heat generation and its transfer from the fuel rod surface to coolant during the design service life without exceeding the permissible design limits of fuel rod damage. The FAs are 4570 mm high (nominal value). When the reactor is in the hot state the height of the powergenerating part of the fuel rod is 3750 mm. Each FA contains 312 fuel rods. The FA skeleton is assembled of 18 guide channels, 13 spacer grids welded to them, an instrumentation channel and a support grid. The fuel rod cladding is a zirconium alloy tube. Sintered UO_2 pellets with a 5% (4.95±0.05) maximum enrichment are stacked inside the cladding. The average linear heat rate of a fuel rod is 167.8 W/cm.

According to the cartogram up to 121 rod cluster control assemblies (RCCAs) are placed inside the core. They are intended for quick chain reaction suppression, maintaining power at assigned level and its level-to-level transition, axial power field leveling, xenon oscillation suppression. Pitch electromagnet drives with pitch position indicators are used for RCCA drive mechanisms. The drives are installed on the reactor top head. The maximum effective time of FA operation between refuelings for a 12-month fuel cycle is 8400 effective hours. The average burnup of unloaded fuel is up to 60 MWD/kg U. Annually 42 fresh FAs are loaded into the core for the basic fuel cycle.

1.1.3. Structure of fuel assembly type

The reactor core is occupied by fuel assembly design for both base TVSA and alternative TVS-2. The TVSA fuel assembly (FA) is considered as a base version of fuel assembly (FA) design and as an alternative version is TVS-2. Both versions of FA are interchangeable and are of reference character.

The core of the reactor includes 163 fuel assemblies; which are identical in design and 312 fuel elements. Each FAs comprises of the following components, top nozzle, bundle of fuel rods, bottom rod, guiding channels and spacing grid as shown in *Fig 1* and *Fig 2* [1]. The fuel rods are cylindrical and cladded with Alloy E-110.

Each rod contains fuel pellets with inner and outer diameter of 0,6cm and 0,78cm respectively. Additionally the fuel pellet are also cladded with material of 0,772cm inner diameter and an outer diameter of 0,910 cm.

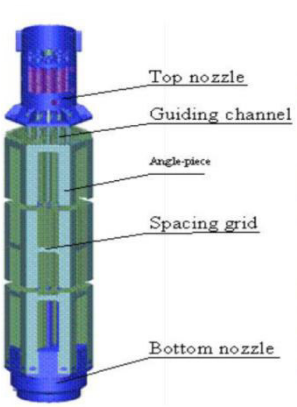

Fig. 1. – TVSA general view

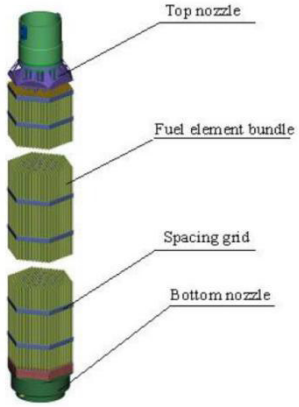

Fig. 2. – TVS-2 general view

During reactor operations, fuel rods are fully immersed in water at a nominal pressure of 15MPa preventing the water from boiling at a normal (220˚C to cover 328.9˚C) operating temperature. Fuel is low enrich (varying between 2,4 % to 4 %) UO_2.

Since boiling on the pallet surface is prohibited the temperature of cladding depends on the type of fuel used. The inserted fuel assemblies contains 312 fuel elements, 18 guide tubes, central tube and one instrumentation tube, all arranged in a triangular lattice with a pitch of 1,275. The hexagonal lattice pitch of the assembly cell is 23,6 cm.

The structure and configuration of the fuel assembly in the core are kept unchanged but rather refueled at the end of its fuel cycle. Water is mostly used as reflector. The huge size of the FAs contribute to the increase of multiplication factor and the short migration length.

The reactor VVER-1200 are required for large scale power generation and are mostly enhanced by increasing the size or length of fuel assembly or advancing the fuel (UO_2) without changing the volume of the core. After modernization, power of the reactor can only be increased by 2-5% of nominal reactor power. Period of refueling of FAs ranges from 12 – 18 months. The main characteristics of this prototype reactor are listed in the table 1 below.

Tab. 1. – Main parameters of the VVER-1200 reactor

Main parameter, unit		*Values*
Nominal thermal capacity, MW		3200
Rated Electrical capacity, MW		1170
Fuel assembly quantity		163
Cladding material(Alloy)		Zr+2.5%Nb
Fuel mass in the rod(kg)		1,575
Fuel rod effective height (cm)		353
Fuel material		UO_2 and $UO_2 + Gd_2O_3$
Density of fuel, g/cm^3		10,5
Fuel temperature, K		500
Fuel pallet diameter, cm		
	outer	0,78
	innerr	0,06
Cladding diameter, cm		0,910
Moderator-coolant		H_2O
	Pressure, MPa	14
	Temperature, C	329
Fuel assembly form		Hexagonal
Lattice, cm		1,35
Type of Lattice		triangle

1.1.4. Configuration of the core and FAs of VVER-1200

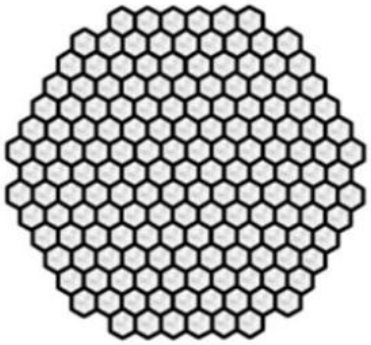

Fig. 3. – Core (163 FAs)

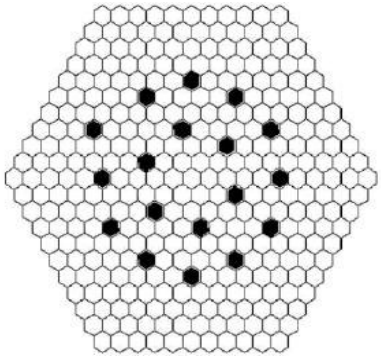

Fig. 4. – FAs (312 FE)

In modern VVER-1200 with uranium enrichment 3-5% at 2-3 years of campaign with partial overload has burnup ranging from 40-50 GWd/KgU and it is even more in intense fuel elements as compared to the burnup of PWR reactors which ranges from 45 to 50 GWd/KgU. Average burnup of unloaded fuel in PWRs is 52,8 GWd/KgU.

Reactivity margin for VVER-1200 reactors is approximately equal to 0,25 Δk/k. The loss of reactivity due to Xe^{135} poisoning in the reactor core is equivalent to % but reactivity worth, also known as Xenon load which is proportional to the Xenon concentration varies with the steady power of the reactor for LWR.

Time required for xenon to achieve it maximum equilibrium concentration is between 30-40hr, with specific yield of $(\gamma) \approx 0{,}03$ % directly from fission and a total fission yield of $(\gamma) \approx 6{,}6\%$. For most thermal reactors the value of optimum lattice lies between 0,5 to 0,6R.

The experimental values obtained for reactivity coefficient of PWRs reactors are summarized in the table 1.2 below. [10]. According to the estimation, coefficients in PWRs are mostly negative.

1.2. WIMS-ANL code

1.2.1. General code properties

The WIMS-ANL (Winfrith Improved Multigroup Scheme) are codes extensively used for power and lattice physics analysis. This developed program was initially aimed at using the transport theory to calculate the neutron flux as a function of energy spatial location in one dimensional cell. The program uses its own 69 group constant library and the ENDF/BV library prepared for various materials, power and temperatures bases on the ENDF format [2].

There are two main transport option known as DSN (discrete ordinates) and PERSUS (collision probability). Transport solutions are performed using any specified intermediate group structure up to the number of library groups. The main transport is mostly preceded by one or two STECTROX flux spectra calculation(s) for few spatial regions in the few region library group structure after which calculation of spectra for a spatial region in the full number of energy groups of its library are performed using the spectra to condense the basic cross sections into few groups [2].

After completion of the main transport solutions, the intermediate group cross sections are collapsed to the broad group structure ($\leq$ 20 groups) and may be written in the microscopic or macroscopic ISOTXS format for use in subsequent transport or diffusion theory codes. The microscopic ISOTXS cross-sections contain the full P_o and P_1 scattering matrices for transport calculations, but their primary use will be in Multigroup diffusion theory analyses [2]. The cross sections are also burnup and spatially dependent. The cross

section and energy group of the 69 library groups are shown in Appendix B.

Main purposes of this code developing algorithms for the WIMS code, creating main data and prelude data to calculate spectra for few spatial region and rated regions in homogeneous medium, creating libraries for theses program. WIMS code and its 69 group libraries are found to be one of the adequate predictor of cell reactivity, burnup process and flux spectrum cell modelling. The input data model includes prelude data (the two transport solutions i.e. PERSUS & DSN) and main data (geometry, composition, cell characteristics, burnup and reaction rate edit).

1.2.2. The general scheme of WIMS-ANL simulation method

To perform neutron calculation in the core of the reactor, simulation of the reactor is required. In this research work, the cell calculation which stimulates the fuel assembly in the reactor core was performed and the output was used in the core calculation which apparently determines the neutronic parameters of the reactor. Firstly, the simulation of the fuel assembly, the WIMS-ANL codes were used. It is necessary to remember that the occurring process in the simulation corresponds to the actual physical process. A neutron will be absorbed by a nuclei to sustain the fission chain reaction by dividing itself and releasing energy. These happens at the end of the simulation, the particle that appear in simulation is known as the neutron flux (neutron clusters). Internally, the WIMS code also generates region-averaged cross sections in an intermediate group structure that can utilize maximum number of fine groups in the library. Presently, fine-group libraries with 69 and 172 groups are used in the ANL RERTR Program [2].

The WIMS-ANL codes uses the transport theory to calculate the neutron flux as a function of energy and positions in the cell. It begins by performing spectra calculations for few regions in the full number of energy groups of its 69 library and uses these spectra to condense the basic cross section into few groups. The obtained flux values are then expanded by using the formal spectra calculations in order for the reaction rate at reach point to be calculated in each library structure. Different geometries such as (the elementary ones are homogenous, annulus, rod clusters in cylindrical geometry and finite cylinders in r-z geometry) are evaluated. In addition to the primary cell calculations, the program is used to carryout burnup calculations using time steps intervals or power value.

Moreover, the program reads the basic macroscopic cross section from its library tape. It then calculates the macro cross sections for each material, with automatically calculated resonance shielding the preliminary spectrum is evaluated using the collision probability methods. In this work, fuel assemblies with different enrichments has been modelled using the WDSN (transport calculations are performed in one dimension) and the main transport option for finding the problem (homogenous) in the finite geometric medium. In modeling it is recommended to convert the hexagonal structure of the fuel assembly (having a triangular lattice pitch) to circular fuel assembly model since WIMS code can only perform cell calculation in circular model. In converting the hexagonal structure to the circular model the volume remains unchanged. The *fig 5* and *fig 6* below shows the hexagonal model and the circular model used for cell computation respectively.

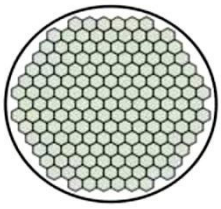

Fig. 5. – Hexagonal model

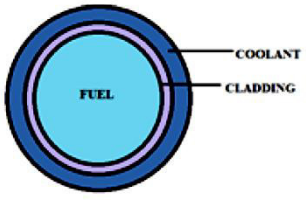

Fig. 6. – Circular model for a unit cell

The WIMS code requires the following information;

– the material of each part of the cell,
– the geometry of the fuel assembly,
– burnup parameters,
– buckling value of the call,
– temperature and power used and other information depending on the method used for the calculation. At the end of the simulation, the values obtained for effective multiplication factor in each medium were used in evaluating the optimum sizes for the four regions of the lattice cell and were further used for optimizing the neutronic parameter associated with the dynamics of the reactor.

1.2.3. Simulating the VVER-1200 reactor core using WIMS-ANL codes

Analysis was performed on the neutronic characteristics associated with the reactor VVER-1200 design using the WIMS-ANL code. In simulating, two main inputs were used i.e. the main input data and the prelude input data.

The prelude input option comprises of method of solution, which contains the SEQUENCE card which defined the main transport routine used in the lattice calculation, and the CELL card was used in the selecting the cell type which in this case was a single homogenous cell [1], 2) accuracy of solution which contains the NGROUP card which set the number of main transport group that was used. Note the larger the number of group selected usually results in a more accurate flux and cell reactivity calculation and the NMESH card was set to calculate the number of mesh points in the main transportation calculations.

Description of the core geometry (NREGION) and compositions (NMATERIAL) were also defined in the prelude data. The NREGION card was used to set the number of zones in the given lattice whilst the material card defined the number of material component and lastly the NREACT card was used in some cases [1] for the finite medium.

The secondary input option for a one dimensional geometry the ANNULUS card was used to define the unit cell and the NMATERIAL card in specified the density/composition, temperature and spectrum type of the material in the set lattice, other input used in the main data were the POWERC (calculation of fuel depletion over a time step) card, reaction rate card and the ISOTXS (for writing micro and macro cross section in the output data file), the BEGINC card which ends the main data input and the PRTOPT card set at 1 to edit result in the full output [1].

2. Description of research

In this study the determination of the neutronic characteristics associated with the dynamics if the reactor which leads to the increase or decrease of the campaign length has been evaluated. Parameters such as reactivity and reactivity feedback were calculated. The reactivity associated with the xenon transients arising from power level change were also calculated. Calculations were also carried out on energy contribution by each fuel composition in the cell. All the parameters were performed for different uranium enrichment (from the very low enrichment to the highly enriched uranium) in finite or infinite medium as a function of buckling. Insertion the fuel assemblies (section 1.1.1) into the core of the reactor, it is important to know that the fuel elements contains fresh fuel isotopic compositions yet to be irradiated by thermal neutrons.

In order to get the accurate optimum lattice of the unit cell and have an overall idea about the optimization of the fuel consumption in the reactor, contribution of every individual isotope was calculated based on the enrichment. The concentrations of every individual

isotope of fuel such as ^{235}U, ^{238}U, O_2 were comprehensively calculated based on the enrichment given and their respective reactivity were analyzed based on *Keff* values obtained from the WIMS-ANL. Below is a graph which indicate the effect of reactivity on the fuel enrichment as it varies.

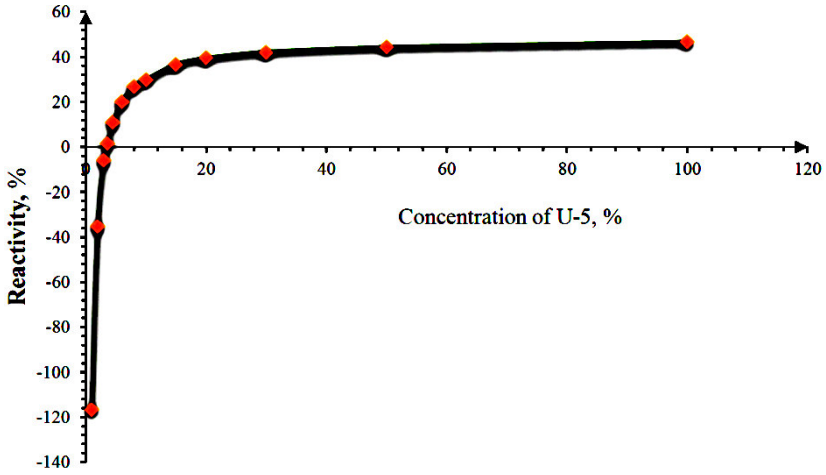

Fig. 7. – Reactivity of VVER-1200 vs. fuel enrichment

The result of the calculation shows that the reactivity of the reactor increases with an increase in fuel enrichment. This is due to the fact that effective multiplication factor of neutron generation (neutrons in the multiplying medium) increases as the enrichment of fissile isotope increases. As the number of fissile isotopes in the fuel increase the kinetic energy of the neutrons causing fission, thus the higher neutron population (neutron flux) which is directly proportion to the energy (E) of the reactor. Hence the reactivity which directly reflects the responsiveness of a reactor to the change in neutron balance also increases.

3. Calculation of optimum size using WIMS-ANL

From the neutronic point of view, the optimum sizes occurred where K_{eff} is maximum in the cold zero power reactor condition (i.e. T = 20˚C). The VVER-1200 reactor core was firstly simulated, using the WIMS-ANL program with various allowable values for each annulus (ANNULUS 1: region of gap in fuel pellet, annulus 2: fuel region, annulus 3: cladding and annulus 4 annular cooling thickness) of the lattice cell.

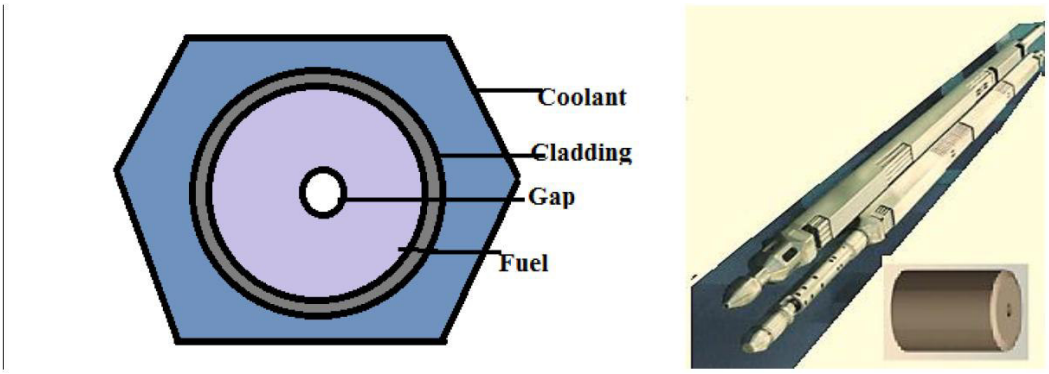

Fig. 8. – lattice cell structure and a fuel assembly

After simulation, the results of the experiment show that the maximum K_{eff} for ANNULUS 4 (with radius ranging from 0,45 cm to 1cm) for VVER-1200 with 2% to 4,5% fuel enrichment were found at radius 0,55cm. The same procedure is repeated for annulus 1(with radius ranging from 0,01 to 0,15cm) , 2(radius between 0,2 to 0,45cm) and 3 (0,4 to 0,75 cm) and their obtained values were proposed to be 0,01 , 0,45 and 0.5425 respectively. Figure 2.1 below shows the optimum radius of coolant region.

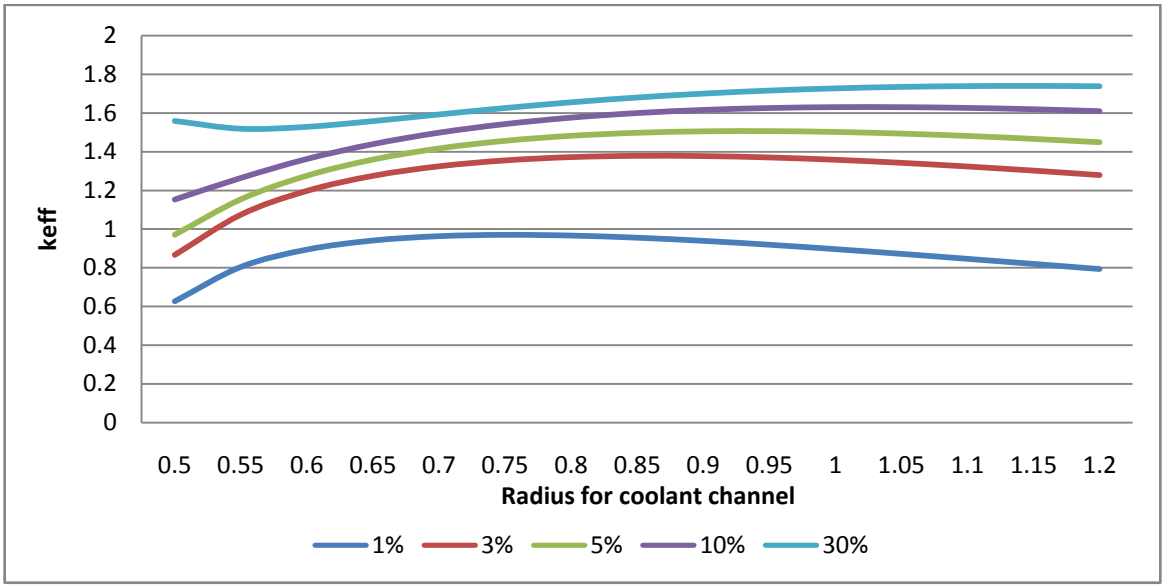

Fig. 9 – Dependence of Keff on coolant channel radius

Below is a tabulated values for the optimum sizes of coolant/moderator. Similar values were obtained in both finite and infinite medium.

Tab. 2. – optimum sizes of coolant for each enrichment

Enrichment%	*Max keff*	*Optimum size*
1%	0.9705	0.75
3%	1.3788	0.85
5%	1.5062	0.95
10%	1.6307	1
30%	1.7395	1.15

The optimized configurations were used as the new input for the subsequent calculations. The optimum values obtained were reintroduced in the lattice cell and examine to it effect and the neutronic parameters associated with the optimized core were then investigated.

4. Conclusion

Calculated by WIMS-ANL, the Keff changes at different fuel concentrations and at different cooling water channel sizes, and the maximum appears. The higher the fuel

concentration, the higher the keff value. As the size of the sink changes, the keff value first increases and then decreases.

REFERENCES

[1] J.G Tyror & R. I. Vanghan. An introduction to Neutron kinetics of nuclear power reactor. Second edition: point reactor kinetics. 2011, p. 182. (in English)
[2] W.L. Woodruff, Argonne National Laboratory, USA and L. S. Leo Pando: "Upgrades to the WIMS-ANL Code and Library. Philippine Nuclear Research Institute, Sao Paulo, Brazil, 1998. (in English)
[3] Yoshiaki Oka, Katsuo Suzuki (edited). Nuclear reactor kinematics and plant control. Second edition. Tokyo, Japan, Springer, 2008, pp. 501–519. (in English)
[4] D.M. Titova, et al. Approaches to optimization of the core reactivity coefficient for the 'master' heat supply reactor. Obinisky institute of nuclear power engineering, national research university (MEPHI), Russia, 2015. (in English)
[5] Bengt pershagen. Light water reactor safety. Substantially revised and updated edition: Studsvik AB, Nykoping. Springer, 1988, pp. 466. (in English)
[6] Teresa Kulikowska. Reactor lattice transport calculation. Institute of atomic energy, Swierk, Poland, 2000, pp. 19–21. (in English)
[7] Carl Bergl., et. On measuring and monitoring of reactivity in subcritical reactor systems with Monte Carlo neutronics. KTH School of engineering, Department of physics, Sweden, 2014, pp. 15. (in English)
[8] Dan Gabrial Cancuci. Handbook of nuclear engineering. Vol. 1: Nuclear engineering fundamentals. Springer, 2010, pp 1172. (in English)
[9] Amir Hosein Faidaei, Saeed Serayeshi. Some neutronic calculations for the VVER-1000 using WIMS and Citation code. Anirkabir University of Technology, Iran, 2009. (in English)
[10] Amir Hosein Faidaei, Saeed Serayeshi. Control rod worth calculation for VVER-1000 using SRAC and MC NPS code. Institute nuclear science and Technology, Vietnam, 2012. (in English)
[11] L. Thilagam, C. Sunil Sunny & K.V. Subbaiah. Doppler coefficient of reactivity calculation for different enrichment of uranium oxide. Safety research. Kalpakkam, India, 2000. (in English)
[12] Gert. Jan Auwerda. Investigation of neutrons and heat transfer in gas core reactor, 2009. (in English)
[13] Georgy L. Khorasanov, Anatoly P. Ivanov, Anatoly I. Blokhin. Neutronic and Physical Characteristics of the Ads CoreWith A Lead-208 Coolant. Institute for Physics and Power Engineering, 2007. (in English)

Optimizing the Neutronic Parameters for VVER-1200 Reactor Core Using WIMS-ANLS Code

Yubin Xu

National Research Tomsk Polytechnic University, Institute of Physics
Lenin St., 2, Tomsk region, Russia 634050
e-mail: 512685424@qq.com

Abstract – In this paper, the WIMS-ANL is used to obtain different keff values and maximum keff values by varying the size of the cooling sink of the fuel assembly of the reactor (VVER-1200) and the fuel UO2 concentration (1 to 30%).

Keywords: VVER-1200, WIMS-ANL; lattice cell, fuel assembly, neutronic characteristics, reactivity, effective multiplication factor.