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AN IMPROVED METHOD FOR AUTOMATED CONTROL OF THE WWER-1000 POWER MANEUVERING

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A multizone mathematical model for automated control of the WWER-1000 power maneuvering has been improved by means of considering the power release at fission of both ^{235}U and ^{239}Pu nuclei, as well as by using simultaneous control actions of changing the boric acid solution concentration in the reactor coolant and the position of control rods of the reactor control system. This distributed model allows us to control the change of reactor technological parameters in specified core sectors, core axial segments, as well as accounting for fuel assembly groups. A new method for automated control of the WWER-1000 power maneuvering based on using three control loops has been proposed, thereby two reactor power control programs have been improved.

INTRODUCTION

When the share of nuclear energy in the total electricity production lies in the range 25...50% and the share of power plants operated in the load-following mode decreases, the electricity production in a consolidated power system (CPS) does not correspond to the electricity consumption. This unfavourable situation becomes critical if the nuclear share in the total electricity generation exceeds 50%. Due to the lack of load following units in the CPS of Ukraine, in order to insure the electricity quality corresponding to standard requirements of the European Union, Ukrainian nuclear power units participating in peak load and frequency regulation should be considered [1].

According to the evaluation of prospects for nuclear energy in Ukraine, the basis of national NPPs will be formed by WWER – type reactors operated at variable loading [2]. When operating a WWER-1000 reactor under variable loading, an optimal choice of the reactor power control method is very important as this method influences greatly on the power equipment behavior characteristics [3, 4]. Hence one of important directions for improvement of the WWER-1000 power control system consists of developing a method for automated control of the reactor power maneuvering characterized by an increased stability of the neutron field in the core, under normal operating conditions [5].

Presently no automated system for control of the WWER-1000 power maneuvering in the range 100…80% of the nominal reactor power N_0 is known and, according to the schedule of WWER-1000 operation, the reactor power maneuvering is manually controlled by an operator. This is not a very eligible choice because a continuous manual control of the reactor power maneuvering leads to a considerable probability of "human factor" accidents.

Among different programs which can be applied to controlling the WWER-1000 power, these main ones will be considered hereinafter [6]:

– the core averaged coolant temperature is constant: $\langle t_w \rangle$ = const (program 1);

– the second circuit inlet steam pressure is constant: p_2 = const (program 2).

In order to minimize the probability of xenon oscillations in large-core thermal reactors like WWER-1000, it is necessary to decrease the space-time nonuniformity of the neutron field in the core [7, 8].

For the case of $\langle t_w \rangle$ = const, when changing the reactor power, both core inlet and outlet coolant temperatures change, so the temperature effect of reactivity is important for both upper and lower parts of the reactor core. In this case the probability of xenon oscillations is high because the sign of temperature changes for the core outlet is opposite to the same for the core inlet. Also the design of control rods does not allow us to control the lower half of the core independently from the upper half.

For the case of p_2 = const, when lowering the reactor power, both core inlet and outlet coolant temperatures are decreased, so the core average coolant temperature is decreased and, due to a negative coolant temperature reactivity coefficient, this effect requires using control rods leading to an increase of the axial non-uniformity of power release defined by axial offset (AO). Hence in the case of p_2 = const the probability of xenon oscillations at reactor power maneuvering is high also.

In order to achieve a high efficiency of the WWER-1000 power maneuvering control, internal physical properties of the core defining transient processes influencing on coolant temperature, neutron flux density, concentration of fission product poisons, etc. should be taken into account precisely. Thus one of main features of an advanced method for automated control of WWER-1000 power maneuvering is using a maximally detailed model of the control object properties, as well as considering the influence of a power control program on these properties, first of all the stability of the reactor power control [7].

The main aim of the paper is developing a complete and detailed model of the neutron-physical processes in the WWER-1000 core, for the purpose of creating the grounds for an innovative automated system controlling the reactor power maneuvering in the range 100…80% of N_0 with high quality, from the point of view of the automatic control theory.

When modelling WWER-1000 power control programs, the shortcoming of existing mathematical models for calculation of reactor technological parameters is that the reactions of ^{239}Pu generation and fission are not considered – see Eq. (1) explaining the mechanism of ^{239}Pu generation [8, 9]:

$$ {}_0^1n + {}_{92}^{238}\mathrm{U} \rightarrow {}_{92}^{239}\mathrm{U} \rightarrow {}_{93}^{239}\mathrm{Np} + e^- + \bar{\nu}_e , $$

$$ {}_{93}^{239}\mathrm{Np} \rightarrow {}_{94}^{239}\mathrm{Pu} + e^- + \bar{\nu}_e . \qquad (1) $$

But the isotopes of ^{135}I and ^{135}Xe bearing xenon oscillations in the core are generated from fission both ^{235}U and ^{239}Pu (Tabl. 1) [4].

Table 1

The probability of ^{135}I and ^{135}Xe generation when ^{235}U and ^{239}Pu are divided, %

Isotope	^{235}U	^{239}Pu
^{135}I	6.29	6.54
^{135}Xe	0.258	1.08

For the first time, a multizone mathematical model of the WWER-1000 core intended for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0, was proposed in [8, 9]. Though this multizone model developed for creating a corresponding simulation model using a specialized software environment (e.g., SIMULINK) could be considered already as a distributed one, where each zone (unit cell) was described using a lumped parameters model, this model was not fit for solving the reactor power control problems because the transient processes in the control object could not be described precisely due to neglecting the difference in properties between ^{235}U and ^{239}Pu, and their fission products also, as well as neglecting the difference in dynamical properties between fuel assembly groups corresponding to fuel campaign years.

Thus, having analyzed the weaknesses of major known models developed for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0 [8, 9], the following directions of improving the simulation of core processes were accepted:

– accounting for production of ^{239}Pu and its fission products;

– introducing a more complete and detailed distributed model of the WWER-1000 core.

AN IMPROVED MATHEMATICAL MODEL OF THE CONTROL OBJECT

The proposed multizone model of the core for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0, where each zone is described using a lumped parameters model, taking into account the creation of neutrons and fission products from both ^{235}U and ^{239}Pu, has got the following advantages, compared to the preceding works [3, 8, 9]:

– compared to [3], where a method for control of the WWER-1000 power maneuvering in the range 100…80% of N_0, based on keeping the core inlet coolant temperature constant and insuring a maximum stability of the core AO for the “Advanced algorithm” of disposition of regulating units in the core, was proposed: for the first time, a three-loop control model using a full and adequate model simulating the core transient processes and delivering an extremely high quality of power control, which yields an improved axial stability of power release in the core during continuous power maneuvering under normal operating conditions, has been developed;

– compared to [8, 9], where a method for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0, based on using a multizone mathematical model of the core where each zone is described using a lumped parameters model, was developed: the three-loop control model was proposed and the model of neutron-physical processes in the core became much more complete due to not only accounting for the creation of neutrons and fission products from both ^{235}U and ^{239}Pu, but thanks to introducing a much more detailed grid of unit cells for the core also.

In this investigation the control object is a nuclear power unit with a WWER-1000 reactor, so the mathematical model of the control object consists of [8]:

– model of the steam generator;

– model of the coolant circulation between the reactor and the steam generator;

– model of the turbogenerator;

– model of the reactor core taking into account the space-time distribution of the control object technological parameters among unit cells formed by 6 specified core sectors (each sector contains one-sixth of all fuel assemblies, as well as one-sixth of all regulating units used for power maneuvering) [2], 10 core axial segments, as well as considering, in any core sector, 4 fuel assembly groups corresponding to fuel campaign years. Thus the distributed model of the reactor core is a multizone model where each zone (unit cell) is described using a lumped parameters model. Thus, each unit cell of the core is marked by “y” (1...10), “x” (1...6) and “z” (1...4) indices denoting axial segment, core sector and fuel assembly group numbers, respectively. For simplicity reasons, hereinafter some cell indices can be missed.

The following assumptions were accepted also:

– the considered start moment was 284.72 eff. days of the 22th campaign of Unit 5, Zaporizhzhya NPP;

– fuel assembly group 1, 2, 3, and 4 includes core cells (2, 3, 4, 5, 9, 13, 55), (11, 19, 22, 30, 31, 32, 41), (10, 12, 18, 20, 21, 54, 68), and (1, 6, 8, 29, 42, 43), respectively;

– for any fuel assembly group (one group included 7–6 fuel assemblies), technological parameters were set as arithmetic means for corresponding fuel assemblies. For example, the average burnup for fuel assemblies included in group 1, 2, 3, and 4 was 12.5, 26.6, 38.5, and 45.9 (MW·d)/kg, respectively;

– a required value of AO is maintained at the expense of changing the position of control rods included in the 9th regulating group, while control rods of all other groups are completely removed from the core. So each specified core sector contains one core cell where control rods can move. The numbers of core cells with control rods of the 9th regulating group are 11, 38, 47, 126, 153, and 117 (360-degree symmetry) [10].

– model of the nuclear reaction kinetics accounting for the change of the core isotope composition due to the fission of both ^{235}U and ^{239}Pu:

$$\frac{d\Phi_i}{d\tau} = \frac{\left(\rho_i(\tau) - \beta_5 - \beta_9\right)\cdot\Phi_i(\tau)}{l} + \sum_{j=1}^{6}\lambda_{j,5}\cdot C_{i,j,5}(\tau) + \\ + \sum_{j=1}^{6}\lambda_{j,9}\cdot C_{i,j,9}(\tau); \quad (2)$$

$$\frac{dC_{i,j,5}}{d\tau} = \frac{\beta_{j,5}\cdot\Phi_i(\tau)}{l} - \lambda_{j,5}\cdot C_{i,j,5}(\tau); \quad (3)$$

$$\frac{dC_{i,j,9}}{d\tau} = \frac{\beta_{j,9}\cdot\Phi_i(\tau)}{l} - \lambda_{j,9}\cdot C_{i,j,9}(\tau); \quad (4)$$

$$\beta_5 \equiv \sum_{j=1}^{6}\beta_{j,5},\ \beta_9 \equiv \sum_{j=1}^{6}\beta_{j,9}, \quad (5)$$

where Φ_i is neutron flux density averaged in the i-th unit cell of the core, $\text{cm}^{-2}\cdot\text{s}^{-1}$; τ is time, s; $\rho_i(\tau)$ is reactivity in a unit cell; β_5, β_9 is delayed-neutron fraction for ^{235}U and ^{239}Pu, respectively; l is neutron lifetime, s; $\lambda_{j,5}, \lambda_{j,9}$ is radioactive decay constant considering the j-th group of delayed-neutron emitters for ^{235}U and ^{239}Pu fission fragments, respectively, s^{-1}; $C_{i,j,5}(\tau)$, $C_{i,j,9}(\tau)$ is flux density of neutrons bound in delayed-neutron emitters belonging to the j-th group of ^{235}U and ^{239}Pu fission fragments, averaged in the i-th unit cell of the core, respectively, $\text{cm}^{-2}\cdot\text{s}^{-1}$; $\beta_{j,5}, \beta_{j,9}$ is delayed-neutron fraction considering the j-th group of delayed-neutron emitters for ^{235}U and ^{239}Pu fission fragments, respectively.

It should be added that neglecting the flow-over of neutrons between unit cells is an intrinsic shortcoming of the proposed multizone model compared to the models used in known 3D diffusion codes.

Taking into account Eq. (1), the ^{239}Pu production by irradiation of ^{238}U is described as

$$\frac{dN_{i,8}}{d\tau} = -N_{i,8}\cdot\sigma_{f,8}\cdot\Phi_i - N_{i,8}\cdot\sigma_{c,8}\cdot\Phi_i; \quad (6)$$

$$\frac{dN_{i,\text{U}-9}}{d\tau} = N_{i,8}\cdot\sigma_{c,8}\cdot\Phi_i - \lambda_{\text{U}-9}\cdot N_{i,\text{U}-9}; \quad (7)$$

$$\frac{dN_{i,\text{Np}}}{d\tau} = \lambda_{\text{U}-9}\cdot N_{i,\text{U}-9} - \lambda_{\text{Np}}\cdot N_{i,\text{Np}}; \quad (8)$$

$$\frac{dN_{i,9}}{d\tau} = \lambda_{\text{Np}}\cdot N_{i,\text{Np}} - N_{i,9}\cdot\sigma_{f,9}\cdot\Phi_i - N_{i,9}\cdot\sigma_{c,9}\cdot\Phi_i, \quad (9)$$

where $N_{i,8}$, $N_{i,\text{U}-9}$, $N_{i,\text{Np}}$, $N_{i,9}$ is concentration of ^{238}U, ^{239}U, ^{239}Np, and ^{239}Pu, respectively, averaged in the i-th unit cell of the core, cm^{-3}; $\sigma_{f,8}$, $\sigma_{f,9}$ is microscopic fission cross-section for ^{238}U and ^{239}Pu, respectively, cm^2; $\sigma_{c,8}$, $\sigma_{c,9}$ is microscopic radiative capture cross-section for ^{238}U and ^{239}Pu, respectively, cm^2; $\lambda_{\text{U}-9}$, λ_{Np} is radioactive decay constant for ^{239}U and ^{239}Pu, respectively, s^{-1}.

The differential equations describing the rate of ^{135}Xe generation due to fission of ^{235}U and ^{239}Pu are written as

$$\frac{dN_{i,\text{I},5}}{d\tau} = P_{\text{I},5}\cdot\Phi_i\cdot\sigma_{f,5}\cdot N_{i,5} - \lambda_\text{I}\cdot N_{i,\text{I},5}; \quad (10)$$

$$\frac{dN_{i,\text{Xe},5}}{d\tau} = \lambda_\text{I}\cdot N_{i,\text{I},5} - \lambda_\text{Xe}\cdot N_{i,\text{Xe},5} - \\ - \Phi_i\cdot\sigma_{a,\text{Xe}}\cdot N_{i,\text{Xe},5}; \quad (11)$$

$$\frac{dN_{i,\text{I},9}}{d\tau} = P_{\text{I},9}\cdot\Phi_i\cdot\sigma_{f,9}\cdot N_{i,9} - \lambda_\text{I}\cdot N_{i,\text{I},9}; \quad (12)$$

$$\frac{dN_{i,\text{Xe},9}}{d\tau} = P_{\text{Xe},9}\cdot\Phi_i\cdot\sigma_{f,9}\cdot N_{i,9} + \lambda_\text{I}\cdot N_{i,\text{I},9} - \\ - \lambda_\text{Xe}\cdot N_{i,\text{Xe},9} - \Phi_i\cdot\sigma_{a,\text{Xe}}\cdot N_{i,\text{Xe},9}, \quad (13)$$

where $N_{i,\text{I},5}, N_{i,\text{I},9}$ is concentration of ^{135}I produced by fission of ^{235}U and ^{239}Pu, respectively, averaged in the i-th unit cell of the core, cm^{-3}; $N_{i,\text{Xe},5}, N_{i,\text{Xe},9}$ is concentration of ^{135}Xe produced by fission of ^{235}U and ^{239}Pu, respectively, averaged in the i-th unit cell of the core, cm^{-3}; $P_{\text{I},5}$, $P_{\text{I},9}$ is probability of producing ^{135}I due to fission of ^{235}U and ^{239}Pu, respectively; $P_{\text{Xe},5}, P_{\text{Xe},9}$ is probability of producing ^{135}Xe due to fission of ^{235}U and ^{239}Pu, respectively ($P_{\text{Xe},5}$ is neglected – see Tabl. 1); $\sigma_{a,\text{Xe}}$, $\sigma_{f,5}$ is microscopic absorption cross-section for ^{135}Xe and fission cross-section for ^{235}U, respectively, cm^2; $N_{i,5}$ is concentration of ^{235}U averaged in the i-th unit cell of the core, cm^{-3}; λ_I, λ_Xe is radioactive decay constant for ^{135}I and ^{135}Xe, respectively, s^{-1}.

The heat generation model for a unit cell of the core considering fission of both ^{235}U and ^{239}Pu includes the following equation:

$$Q_i(\tau) = \Phi_i(\tau)\cdot V_i\cdot(\Sigma_{f,5}\cdot E_{f,5} + \Sigma_{f,9}\cdot E_{f,9}), \quad (14)$$

where V_i is the unit cell volume; $\Sigma_{f,5}, \Sigma_{f,9}$ is macroscopic fission cross-section for ^{235}U and ^{239}Pu, respectively, cm^{-1}; $E_{f,5}$, $E_{f,9}$ is nucleus fission energy for ^{235}U and ^{239}Pu, respectively, J.

The heat transfer model for a unit cell of the core includes the following equations:

$$Q_i(\tau) = c_{p,f}\cdot m_{i,f}\cdot\frac{dt_{i,f}}{d\tau} + \alpha\cdot F_i(t_{i,f} - t_{i,w}); \quad (15)$$

$$\alpha\cdot F_i(t_{i,f} - t_{i,w}) = c_{p,w}\cdot m_{i,w}\cdot\frac{dt_{i,w}}{d\tau} + \\ + \frac{2\cdot c_{p,w}\cdot m_{i,w}}{\tau_0}\cdot(t_{i,w} - t_{i,w,in}), \quad (16)$$

where $c_{p,f}$, $c_{p,w}$ is fuel and coolant specific heat, respectively, J/(kg·K); $m_{i,f}$, $m_{i,w}$ is fuel and coolant mass in a unit cell, respectively, kg; $t_{i,f}$, $t_{i,w}$ is fuel and coolant average temperature in a unit cell, respectively, °C; $t_{i,w,in}$ is coolant inlet temperature in a unit cell, °C; α is coefficient of heat transfer from fuel rods to coolant, $\text{W/(m}^2\cdot\text{K)}$; F_i is heat transfer surface area in a unit cell, m^2; τ_0 is coolant passage time in a unit cell, s.

The reactivity deviation in a unit cell is found as

$$\delta\rho_i = \delta\rho_{i,r} + \delta\rho_{i,b} + \delta\rho_{i,N} + \delta\rho_{i,\text{Xe}} + \delta\rho_{i,t}, \quad (17)$$

where $\delta\rho_{i,r}$, $\delta\rho_{i,b}$, $\delta\rho_{i,N}$, $\delta\rho_{i,\text{Xe}}$, $\delta\rho_{i,t}$ is reactivity deviation due to deviation of the position of control rods, concentration of boric acid in the reactor circuit coolant, reactor power, concentration of xenon in the core, reactor circuit coolant temperature, respectively.

The reactivity deviation due to a deviation of the position of control rods in a unit cell is calculated as

$$\delta\rho_{i,r} = \frac{\partial\rho_i}{\partial h_{i,r}}\delta h_{i,r}, \qquad (18)$$

where $\frac{\partial\rho_i}{\partial h_{i,r}}$ is control rod position coefficient of reactivity; $\delta h_{i,r}$ is control rod position deviation.

The reactivity deviation due to a deviation of the concentration of boric acid in the reactor circuit coolant, for a unit cell is calculated as

$$\delta\rho_{i,b} = \frac{\partial\rho_i}{\partial C_{i,b}}\delta C_{i,b}, \qquad (19)$$

where $\frac{\partial\rho_i}{\partial C_{i,b}}$ is boric acid concentration coefficient of reactivity; $\delta C_{i,b}$ is boric acid concentration deviation.

When boric acid solution is inserted, the boric acid concentration deviation is found from the equation:

$$T_4\frac{\partial C_{i,b}}{\partial\tau} + \delta C_{i,b} = k_4\cdot\delta G_{i,b}, \qquad (20)$$

where T_4, k_4 is time and transfer constant, respectively, s; $\delta G_{i,b}$ is boric acid mass flow deviation, kg/s.

When desalted water is inserted, the boric acid concentration deviation is found from the equation:

$$T_5\frac{\partial C_{i,b}}{\partial\tau} + \delta C_{i,b} = k_5\cdot\delta G_{i,w}, \qquad (21)$$

where T_5, k_5 is time and transfer constant, respectively, s; $\delta G_{i,w}$ is desalted water mass flow deviation, kg/s.

The reactivity deviation due to a deviation of the reactor power, for a unit cell is calculated as

$$\delta\rho_{i,N} = \frac{\partial\rho_i}{\partial N}\delta N, \qquad (22)$$

where $\frac{\partial\rho_i}{\partial N}$ is reactor power coefficient of reactivity; δN is reactor power deviation.

The reactivity deviation corresponding to a deviation of the ^{135}Xe concentration, for a unit cell is calculated as

$$\delta\rho_{i,\mathrm{Xe}} = \frac{\partial\rho}{\partial N_{\mathrm{Xe}}}\delta N_{i,\mathrm{Xe}}, \qquad (23)$$

where $\frac{\partial\rho}{\partial N_{\mathrm{Xe}}}$ is ^{135}Xe concentration coefficient of reactivity.

Al last, the reactivity deviation due to a deviation of the reactor circuit coolant temperature, for a unit cell is calculated as

$$\delta\rho_{i,t} = \frac{\partial\rho_i}{\partial t_w}\delta t_{i,w}, \qquad (24)$$

where $\frac{\partial\rho_i}{\partial t_w}$ is coolant temperature coefficient of reactivity; $\delta t_{i,w}$ is coolant temperature deviation.

The numerical values of main model parameters were set according to [10–13]. For instance, considering 6 groups of delayed-neutron emitters for ^{235}U and ^{239}Pu fission fragments, the accepted values of delayed-neutron fractions are shown in Tabl. 2 [11].

Table 2

The delayed-neutron fractions for ^{235}U and ^{239}Pu, 10^{-3}

Fraction	^{235}U	Fraction	^{239}Pu
$\beta_{1,5}$	0.21	$\beta_{1,9}$	0.072
$\beta_{2,5}$	1.4	$\beta_{2,9}$	0.626
$\beta_{3,5}$	1.26	$\beta_{3,9}$	0.444
$\beta_{4,5}$	2.52	$\beta_{4,9}$	0.685
$\beta_{5,5}$	0.74	$\beta_{5,9}$	0.18
$\beta_{6,5}$	0.27	$\beta_{6,9}$	0.093

The accepted core-averaged values of boric acid concentration, reactor power and coolant temperature coefficients of reactivity are shown in Tabl. 3 [10].

Table 3

Core-averaged coefficients of reactivity

$\partial\rho/\partial C_b$, 1/g/kg	-0.0158
$\partial\rho/\partial N$, 1/MW	$-1.16\cdot10^{-6}$
$\partial\rho_i/\partial t_w$, 1/°C	$-6.7\cdot10^{-5}$

THE METHOD FOR AUTOMATED CONTROL OF THE WWER-1000 POWER MANEUVERING

In order to insure a stable state of the WWER-1000 core at its power maneuvering, a constant value of AO must be maintained and the change of the linear heat rate axial profile must be controlled also, as this change badly influences on the core state due to its internal feedbacks [6]. Thus, for improved automated control of the WWER-1000 power maneuvering, a new method using three control loops has been proposed. These control loops have such functions:

– the first control loop maintains a scheduled change of the reactor power at the expense of regulating the concentration of boric acid in the reactor circuit coolant;

– the second control loop maintains a required value of AO at the expense of changing the position of control rods;

– the third control loop maintains the core averaged coolant temperature constant (program 1) or the second circuit inlet steam pressure constant (program 2), at the expense of changing the position of main valves of the turbogenerator.

The principles of the proposed method for improved controlling the WWER-1000 power maneuvering are:

– the core AO must be regulated by control rods, while the reactor power must be maintained by the regulator of the concentration of boric acid in the reactor coolant;

– the effect of xenon-poisoning cycle must be used, in order to decrease the boric acid concentration change at reactor power maneuvering;

– the regulators take into account the non-linear properties of the control object and the participation of operators in the reactor power maneuvering procedure is not required.

The proposed method is applicable to any existing program of controlling the WWER-1000 power under variable loading-mode conditions.

RESULTS

The simulation models based on Eq. (2)–(24) have allowed us to study the details of the processes in the core at reactor power maneuvering and improve their regulation quality. For example, the model simulating a unit cell of the WWER-1000 core includes 26 differential equations, 3 input parameters: $(h_i; C_{i,b}; t_{i,w,in})$ and 4 output parameters: $(\Phi_i; Q_i; t_{i,w,out}; t_{i,f})$. As a result, using the Simulink suite of MATLAB, a distributed model of the WWER-1000 core allowing us to take into account the distributed processes in the core at its power maneuvering was created (Fig. 1) [6].

The improved method for automated control of the WWER-1000 power under variable loading conditions using 3 control loops and allowing us to improve the known programs for controlling the WWER-1000 power with a constant core average coolant temperature $\langle t_w \rangle = \text{const}$ and a constant second circuit inlet steam pressure $\langle p_2 \rangle = \text{const}$ is based on this distributed simulation model of the WWER-1000 core. The schematic diagram of these improved reactor power control programs is shown in Fig. 2.

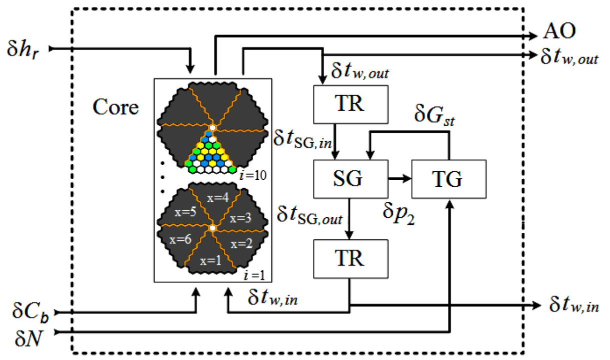

Fig. 1. The structure of the distributed model of the WWER-1000 core realized in Simulink: SG is steam generator; TR is turbogenerator; TR is a transportation lag element

Considering spasmodic changes of main technological parameters of the power unit with a WWER-1000 reactor, the results of four experiments carried out at the South-Ukraine NPP, Unit 3 were used in order to investigate the reactor dynamic behavior [7]. Changes of the position of turbine regulating valves as well as control rods have been taken into account. During the experiments a perturbation was made by continuous movement of the regulating group of control rods, moving near 10% of the core height down. The divergence between the model and experimental [7] data was estimated by calculating the average $\langle \delta \rangle$ and maximum $\delta^{\max}$ relative error of modelling (Fig. 3):

$$\langle \delta \rangle \approx 9.4 \cdot 10^{-2}\%, \quad \delta^{\max} \approx 1.5 \cdot 10^{-1}\%. \qquad (25)$$

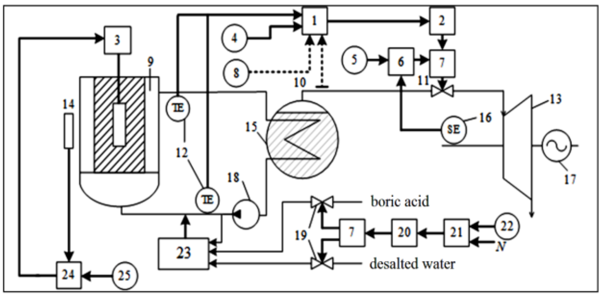

Fig. 2. The schematic diagram of improved reactor power control programs: 1 is $\langle t_w \rangle$ (program 1) or p_2 (program 2) regulator; 2 is turbine control mechanism; 3 is safety-rod actuator; 4 is $\langle t_w \rangle$ selector; 5 is turbine rotating frequency selector; 6 is turbine rotating frequency regulator; 7 is servomotor; 8 is p_2 selector; 9 is reactor; 10 is p_2 primary detector; 11 are turbine regulating valves; 12 are reactor coolant temperature sensors; 13 is turbine; 14 is ion chamber; 15 is steam generator; 16 is turbine rotating frequency sensor; 17 is electric generator; 18 is reactor coolant pump; 19 are boric acid and desalted water regulating valves; 20 is boric acid and desalted water supply control mechanism; 21 is reactor unit power regulator; 22 is electric generator power selector; 23 is boost pump tank; 24, 25 is AO regulator and selector, respectively

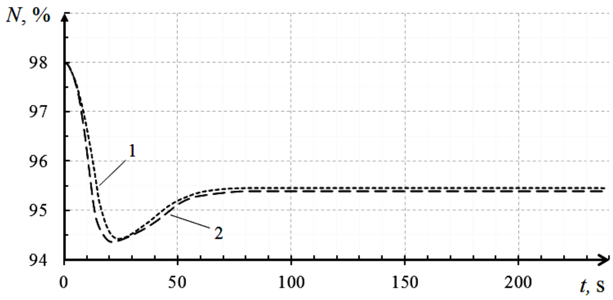

Fig. 3. Influence of the regulating group position on the WWER-1000 power: 1, 2 is experimental [7] and model curve, respectively

It should be noticed that the average and maximum relative error of modeling achieved in [6] was

$$\langle \delta \rangle \approx 2.4 \cdot 10^{-1}\%, \quad \delta^{\max} \approx 9.2 \cdot 10^{-1}\%. \qquad (26)$$

The relative errors of modelling the influence of the reactor outlet coolant temperature on the electric generator power were small also, so a conclusion was made that the proposed simulation model allowed us to improve considerably the accuracy of controlling the WWER-1000 power maneuvering.

Program 1

For program 1 ($\langle t_w \rangle = \text{const}$), the amplitude of changing the regulating group position at WWER-1000 power maneuvering according to the daily loading cycle $100\% \to 80\% \to 100\%$, using

– improved automated control system (improved ACS);

– known automated control system proposed in [6] (known ACS);

– traditional automated control system [5] (traditional ACS) is shown in Fig. 4.

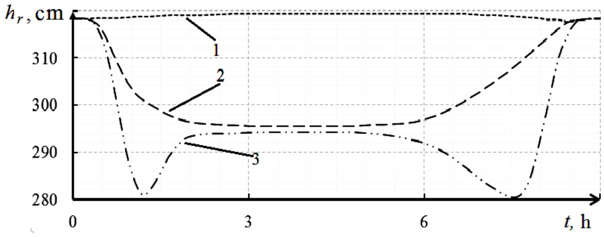

Fig. 4. The change of the regulating group position at WWER-1000 daily power maneuvering for program 1: 1, 2, 3 is improved, known and traditional ACS, respectively

It can be seen that, for the program keeping the core averaged coolant temperature constant, using the improved ACS at WWER-1000 daily power maneuvering has resulted in a considerably decreased amplitude of moving the control rods comparing to both the known and traditional ACS.

The generator power N_g and boron acid concentration C_b change at daily WWER-1000 power maneuvering for program 1 is shown in Figs. 5 and 6, respectively.

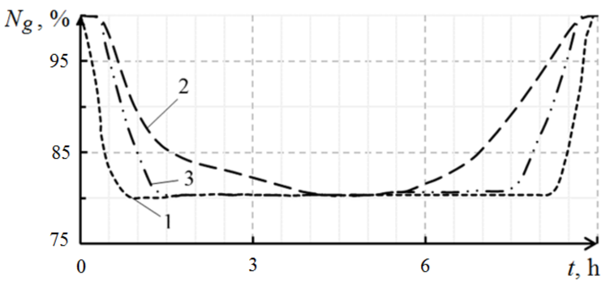

Fig. 5. The generator power change for program 1: 1, 2, 3 is improved, known and traditional ACS, respectively

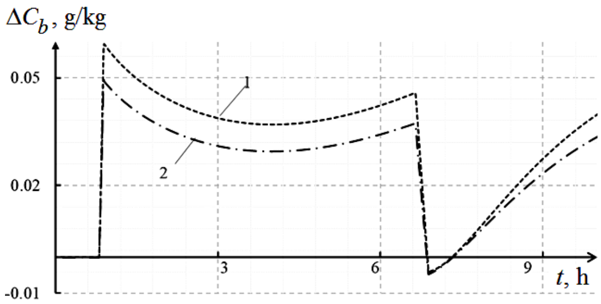

Fig. 6. The boron acid concentration change at reactor power maneuvering: 1, 2 is program 1 and 2, respectively

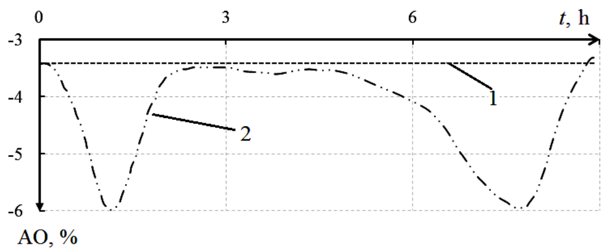

Fig. 7. The change of AO at WWER-1000 daily power maneuvering for program 1: 1, 2 is improved and traditional ACS, respectively

For program 1 also, the amplitude of changing axial offset at WWER-1000 power maneuvering according to the daily loading cycle $100\% \rightarrow 80\% \rightarrow 100\%$, using the improved and traditional ACS is shown in Fig. 7.

As is known, the lumped parameters model of neutron kinetics is applicable to solving the reactor power control tasks in case of sufficiently small volumes of core cells and a sufficiently big number of cells described by the lumped parameters model. So, lowering the size of core cells by introducing a grid formed by axial segments, core sectors and fuel assembly groups has increased the modelling correctness by means of taking into account the core internal distributed properties including the transient processes due to presence of ^{135}Xe.

Thus using the improved ACS at WWER-1000 daily power maneuvering for the program keeping the average coolant temperature constant, the change of AO is considerably lower comparing to the traditional ACS and the stability of power release in the core at its power maneuvering under normal operating conditions has been considerably improved for program 1: the maximum absolute value of AO decreases by 43% (from 6.0 to 3.4%) – see Fig. 7.

Program 2

For program 2 (p_2 = const), the amplitude of changing the regulating group position at WWER-1000 power maneuvering according to the daily loading cycle $100\% \rightarrow 80\% \rightarrow 100\%$, using the improved and traditional ACS is shown in Fig. 8.

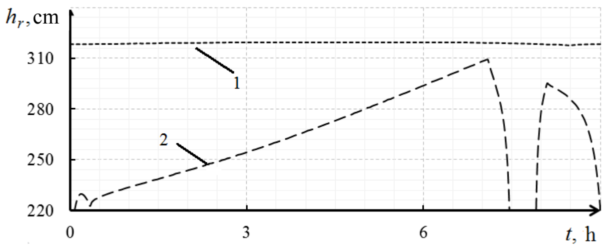

Fig. 8. The change of the regulating group position at WWER-1000 daily power maneuvering for program 2: 1, 2 is improved and traditional ACS, respectively

It can be seen that, for the program keeping the second circuit inlet steam pressure constant, using the proposed ACS at WWER-1000 daily power maneuvering has resulted in a considerably decreased amplitude of moving the control rods comparing to the traditional ACS.

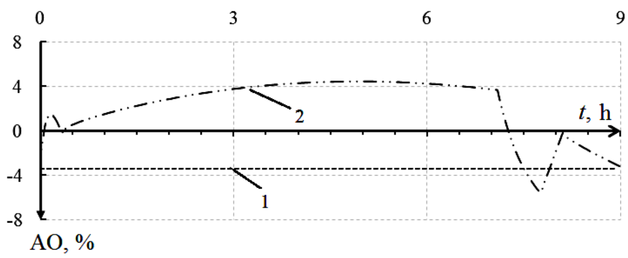

Fig. 9. The change of AO at WWER-1000 daily power maneuvering for program 2: 1, 2 is improved and traditional ACS, respectively

At last, for program 2, the amplitude of changing axial offset at WWER-1000 power maneuvering according to the daily loading cycle $100\% \rightarrow 80\% \rightarrow 100\%$, using the improved and traditional ACS is shown in Fig. 9.

Hence using the improved ACS at WWER-1000 daily power maneuvering for the program keeping the second circuit inlet steam pressure constant, the change of AO is considerably lower comparing to the traditional ACS, and the stability of power release in the core at its power maneuvering under normal operating conditions has been considerably improved for program 2 also: the maximum absolute value of AO decreases by 39% (from 5.6 to 3.4 %) – see Fig. 9.

The generator power change at reactor power maneuvering for program 2 was the same as for program 1 (see Fig. 5).

The boron acid concentration change at reactor power maneuvering for program 2 is shown in Fig 6.

It should be noticed that a number of 3D kinetic codes describing the diffusion of neutrons in the core based on the few-group approach, e. g. DYN3D, NESTLE, etc. are widely used presently. Compared to the proposed multizone model of the WWER-1000 core where each zone is described on the basis of a lumped parameters model, such 3D diffusion codes describe the core neutron flux more accurately. But, aiming to create a method for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0, the proposed model is preferable because:

– an application of 3D diffusion codes to automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0 is much more labor-consuming, though it's possible principally, on the basis of the proposed approach;

– when developing an automated control system for control of the WWER-1000 power maneuvering in the range 100…80% of N_0, modelling the dynamic behavior of a nuclear power unit as a whole includes using lumped parameters models for such important elements of a power unit as steam generators, transport sections between the reactor and steam generators, turbogenerators, etc. Thus, when modelling the dynamic behavior of a whole power unit, it is not reasonable to use an extremely precise code for one element of the unit only;

– the proposed model and method for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0 have already delivered a very high quality of reactor power regulation.

CONCLUSIONS

1. The mathematical model for automated control of the WWER-1000 power maneuvering has been improved by means of considering separate groups of WWER-1000 fuel assemblies, accounting for the power release at fission of both ^{235}U and ^{239}Pu nuclei, and using simultaneous control actions of influencing on the core power by changing both the concentration of boric acid solution in the reactor coolant and the position of control rods of the reactor control system. This improved distributed model of the reactor core considering lumped parameters core zones (unit cells) allows us to calculate the technological parameters of the control object more precisely, and therefore control their change in specified core sectors, axial segments, as well as fuel assembly groups corresponding to fuel campaign years.

2. Having developed the improved multizone simulation model of the WWER-1000 core and considering the reactor as a control object, the relative error of modelling the static and dynamic reactor properties has been considerably decreased comparing with the known model [12], to be exact:

– for the neutron reactor power, 2.6 times (from $2.4 \cdot 10^{-1}$ to $9.4 \cdot 10^{-2}\%$);

– for the core outlet coolant temperature, by 10 % (from $1.1 \cdot 10^{-1}$ to $1 \cdot 10^{-1}\%$);

– for the electric generator power, 1.8 times (from $1.7 \cdot 10^{-1}$ to $9.6 \cdot 10^{-2}\%$).

3. A new method for automated control of the WWER-1000 power maneuvering based on using three control loops has been proposed, where

– the first control loop maintains a scheduled change of the reactor power at the expense of regulating the concentration of boric acid in the reactor circuit coolant;

– the second control loop maintains a required value of AO at the expense of changing the position of control rods;

– the third control loop maintains a required temperature regime of the reactor circuit coolant at the expense of changing the position of main valves of the turbogenerator.

This new method for automated control of the WWER-1000 power maneuvering delivers an improved stability of the power release in the core which is described by decreased average and maximum values of the axial offset during reactor power maneuvering. Namely, for the coolant temperature regime keeping:

– the core averaged coolant temperature constant, AO maximum module decreases by 43% (from 6.0 to 3.4%);

– the second circuit inlet steam pressure constant, AO maximum module decreases by 39% (from 5.6 to 3.4%).

4. The proposed method is applicable to any existing program of controlling the WWER-1000 power under variable loading-mode conditions. Though well-known 3D diffusion codes describe the neutron flux in the WWER-1000 core more accurately than the proposed multizone model, this multizone model is fit for automated control of the WWER-1000 power maneuvering in the range 100…80% of N_0, as it delivers a very high quality of reactor power regulation while it is rather simple.

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УСОВЕРШЕНСТВОВАННЫЙ МЕТОД АВТОМАТИЗИРОВАННОГО УПРАВЛЕНИЯ ИЗМЕНЕНИЕМ МОЩНОСТИ ВВЭР-1000

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Многозонная математическая модель процессов в активной зоне (АКЗ) реактора типа ВВЭР-1000, разработанная для автоматизированного управления изменением мощности реактора, усовершенствована за счет учета энерговыделения при делении не только ядер ^{235}U, но и ^{239}Pu, а также путем применения одновременных управляющих воздействий по каналам изменения концентрации раствора борной кислоты в теплоносителе 1-го контура и положения управляющих стержней системы управления и защиты реактора. Предложенная распределенная модель процессов в АКЗ позволяет контролировать изменение технологических параметров в выделенных секторах симметрии и аксиальных сегментах АКЗ, для групп ТВС, соответствующих годам топливного цикла. Новый метод автоматизированного управления изменением мощности реактора типа ВВЭР-1000, основанный на применении трех контуров управления, позволил усовершенствовать две известные программы управления мощностью реактора.

ВДОСКОНАЛЕНИЙ МЕТОД АВТОМАТИЗОВАНОГО УПРАВЛІННЯ ЗМІНОЮ ПОТУЖНІСТІ ВВЕР-1000

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Багатозонна математична модель процесів в активній зоні (АКЗ) реактора типу ВВЕР-1000, яка розроблена для автоматизованого управління зміною потужності реактора, вдосконалена за рахунок обліку енерговиділення при розподілі не тільки ядер ^{235}U, а й ^{239}Pu, а також шляхом застосування одночасних дій, що управляють по каналах зміни концентрації розчину борної кислоти в теплоносії 1-го контуру і положення керуючих стрижнів системи управління та захисту реактора. Запропонована розподілена модель процесів у АКЗ, яка дозволяє контролювати зміну технологічних параметрів у виділених секторах симетрії і аксіальних сегментах АКЗ, для груп ТВЗ, що відповідають рокам паливного циклу. Новий метод автоматизованого управління зміною потужності реактора типу ВВЕР-1000, заснований на застосуванні трьох контурів управління, дозволив вдосконалити дві відомі програми управління потужністю реактора.