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T. Foshch[1], M. Maksimov[1], S. Pelykh[1], O. Maksimova[2]

[1] Odesa National Polytechnic University, Odesa, Ukraine
[2] Odesa National Academy of Food Technologies, Odesa, Ukraine

Models and Methods for Automated Control of Power Change at VVER-1000 Nuclear Power Unit

This work is devoted to the development of automated control models and methods of power change at VVER-1000 nuclear power unit to provide the most stable axial offset in the load-following mode. Improved multi-zone mathematical model of VVER-1000 allows taking into account the energy release of ^{235}U nuclei fission as well as ^{239}Pu and includes a sub-model with distributed parameters.

The automated control method of power change at VVER-1000 nuclear power unit that uses three control loops was proposed for the first time. The first loop maintains change of reactor power by controlling the boric acid concentration in the primary coolant. The second control loop maintains the required value of axial offset by controlling the position of 9th group control rods, and the third one maintains coolant temperature mode or steam pressure mode by controlling the main valve positions in the turbine generator.

Keywords: method of automated control, control models and methods, NPP, mathematical model, ^{235}U, ^{239}Pu, control loop, automatic control system, VVER-1000.

Т. В. Фощ, М. В. Максимов, С. Н. Пелих, О. Б. Максимова

Моделі та методи автоматизованого керування зміною потужності енергоблока з ВВЕР-1000

Розглянуто розробку моделей та методів керування зміною потужності ядерної енергетичної установки з ВВЕР-1000, які дають змогу отримати найбільш стабільний аксіальний офсет (АО). Вдосконалено багатозонну математичну модель реактора типу ВВЕР-1000, яка відрізняється від відомих урахуванням виділення енергії в разі поділу ядер як ^{235}U, так і ^{239}Pu, та включає субмодель з розподіленими параметрами. Вперше запропоновано метод автоматизованого керування зміною потужності ЯЕУ з ВВЕР-1000, в якому застосовано три контури керування, один з яких підтримує регламентну зміну потужності реактора за рахунок регулювання концентрації борної кислоти в теплоносії, другий підтримує необхідне значення АО зміненням положення стрижнів СУЗ, а третій — температурний режим теплоносія регулюванням положення головних клапанів турбогенератора.

Ключові слова: метод автоматизованого керування, моделі та методи керування, ядерна енергетична установка (ЯЕУ), математична модель, ^{235}U, ^{239}Pu, контур керування, система автоматичного керування, ВВЕР-1000.

Nuclear power plants (NPPs) are currently used in Ukraine to cover only the basic load of the combined energy system (CES) in the country. A sufficient condition for satisfactory coverage of peak and semi-peak loads of CES is about 20 % share of electricity generation at hydroelectric power stations, but this share currently makes up only 5 %. The fact is that the share of power generation at NPPs in the country can reach 60 % and it is expedient to use NPP, which will allow matching the consumption and generation of electricity in order to solve the issue related to insufficient maneuver capacity in CES. Therefore, the relevant task is to adapt operating Ukrainian NPPs to maneuverable operating conditions by creating methods of the automated control system for power change at NPP units.

It is important to emphasize that ensuring stable energy release in the reactor core while power maneuvering in the core of the water-cooled water moderated power reactor (VVER-1000) is rather a complicated problem, which is solved by achieving maximum stability of axial offset with time [1]. Thus, it is necessary to take into account the change in the reactor parameters, which affect axial offset, while creating the methods of automated control at NPP unit [2, 3].

Background

The analyses of the existing control programs of power change at VVER-1000 nuclear power units and process characteristics, advantages, disadvantages of each program have demonstrated that all the existing control programs cause perturbation, which can lead to loss of reactor stability due to changes in the field of energy release [4]. As a perturbation, there can be process parameters such as core inlet coolant temperature or secondary steam pressure. In addition, the reactor properties were not taken into account in any control program due to intrinsic perturbations [5, 6].

M. V. Maksimov, V. A. Ivanov, M. P. Shalman, V. P. Severin, V. I. Plyutinsky, M. O. Duel, V. I. Rotach, V. O. Podsbyyakin and many others made significant contributions to the theory and practice of control power methods [7—9]. The analysis of slow xenon transient processes characterized by additional feedback, which showed that the redistribution of energy release in the core, and processes in the theory of reactors were described by A. D. Galanin, V. O. Orlov, L. N. Usachov, C. M. Feinberg, S. B. Shikhov, R. A. Bat, B. Davison, S. Chandrasekara, K. Case, J. Lener, R. Wing.

VVER-1000 control programs. Three static control programs of VVER-1000 were used in this work:

1) the control program of the power unit with constant average temperature of the primary coolant (t_{av} = const), in which the main characteristics are presented in Fig. 1;

2) the control program of the power unit with constant pressure in the secondary system (p_{II} = const), in which the main characteristics are presented in Fig. 2;

3) the control program of the power unit with constant coolant temperature at the core inlet (t_{in} = const), in which the main characteristics are presented in Fig. 3.

In order to understand the operation of the control programs, they are described below.

The control program of the power unit with constant average temperature of the primary coolant operates according to the principle below.

In order to increase/decrease power by the setting device, the operating personnel of the power unit changes the set value, while depending on error signal, the unit power control generates a control command that is transmitted to turbine

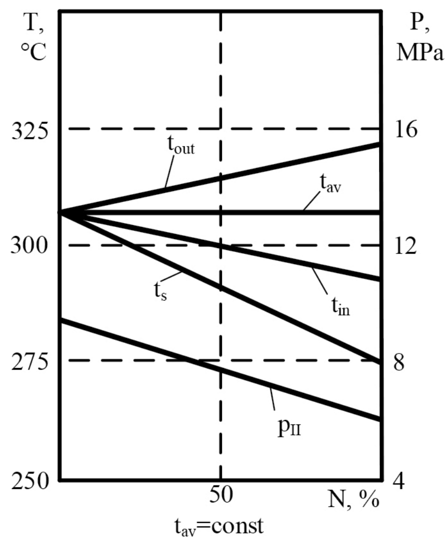

Fig. 1. Characteristics of power unit with t_{av}=const

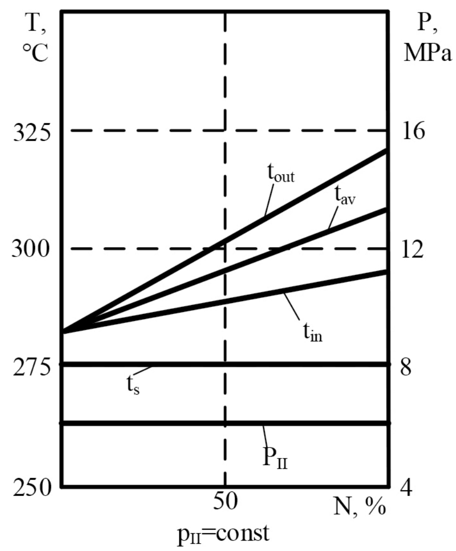

Fig. 2. Characteristics of power unit with p_{II}=const

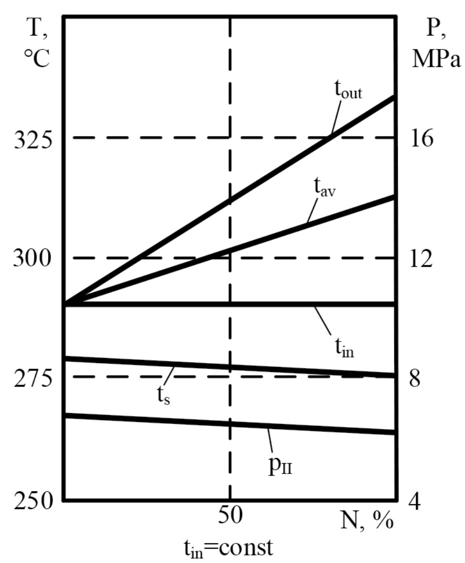

Fig. 3. Characteristics of the power unit with t_{in}=const

control mechanism. The control mechanism of the turbine opens/closes turbine control valves by means of servomotor. Consequently, electric power of the generator will change until error signal equals to zero.

In opening/closure of turbine control valves, the steam pressure before the turbine and in the steam generator increases/decreases, this consequently leads to decrease/increase of steam pressure and saturation temperature in the steam generator, in other words, the amount of heat released from the secondary system decreases/increases. These processes previously led to the fact that the primary coolant temperature at the steam generator outlet decreases/increases. At the same time, the average temperature of the coolant also decreases/increases.

The control of neutron power produced by the automatic power control (APC) consists of "the correcting controller" of the average temperature of the primary coolant and neutron power controller. An error signal is generated by setting device and sensors of average temperature in the primary coolant, thus, the control of the average temperature in the primary coolant generates a correction signal to the neutron power controller. The neutron power controller further changes control rod position, this leads to maintenance of constant average temperature in the primary coolant during transfer from one power level to another.

The functioning of the control program with pressure constant in the secondary system is similar to operation of the control program with average temperature constant of the coolant in the core, except the correction signal generated to the neutron power controller by the error signal of the primary measuring transducer of the steam pressure in the secondary system and the setting device of the steam pressure in the secondary system.

The functioning of the control program with temperature constant of the coolant at the core inlet is also similar to operation of the control program with average temperature constant of the primary coolant.

Some advantages and disadvantages of NPP power control programs [10] are presented in Table 1.

Power unit model. The unit model is described as unit ($\boldsymbol{R}$; $\boldsymbol{SG}$; $\boldsymbol{TG}$; $\boldsymbol{PL}$; $\delta\boldsymbol{h}$; $\delta\boldsymbol{C_b}$; N_e)=$\boldsymbol{unit}$(AO; $t_{r,w,out}$; $t_{r,w,in}$; $t_{r,w,av}$; p_{st}) that includes the following components: steam generator model ($\boldsymbol{SG}$)($\boldsymbol{t_{sg,w,in}}$; $\boldsymbol{G_{sg,w,out}}$)=$\boldsymbol{SG}$ ($t_{sg,w,out}$; p_{st}); a model that considers transport delay of the coolant circulating in the pipelines from the reactor to SG and vice versa ($\boldsymbol{t_{sg,w,out}}$; $\boldsymbol{t_{r,w,out}}$)=$\boldsymbol{PL}$ ($t_{sg,w,in}$; $t_{r,w,in}$); turbine generator model ($\boldsymbol{TG}$) ($\boldsymbol{p_{st}}$;$\boldsymbol{N_e}$)=$\boldsymbol{TG}$(G_{st}; N_t); reactor model ($\delta\boldsymbol{C_b}$; $\boldsymbol{t_{r,w,in}}$; $\delta\boldsymbol{h}$; $\boldsymbol{i}$)=$\boldsymbol{R}$(AO; Q_i; $t_{i,w,in}$), which allows calculation of changes in the indicated process parameters of the control object (CO) in the core symmetry sectors, the high-level layers of the core and groups of fuel cells in each symmetry sector.

In the above functions, bold variables mean input parameters and italic variables are output parameters with the following marking: δh — deviation of control rod position (CR), cm; δC_b — deviation of boric acid concentration in the coolant, g/kg; N_e — unit electric power, MW; AO — axial offset, %; $t_{r,w,out}$ — primary coolant temperature at the reactor outlet, °C; $t_{r,w,in}$ — primary coolant temperature at the reactor inlet, °C; $t_{r,w,av}$ — average temperature of the primary coolant, °C; p_{st} — pressure of the saturated steam at the steam turbine inlet, MPa; $t_{sg,w,in}$ — primary coolant temperature at the steam generator inlet, °C; G_{st} — steam consumption, kg/s; $t_{sg,w,out}$ — primary coolant temperature at the steam generator outlet, °C; N_t — turbine thermal power, MW; i — process parameter, respectively, in the elementary cell (EC) of the core, by the symmetry sectors (x), high-level layers of the core (y), groups of fuel cells in each symmetry sector (z); Q_i — specific energy release in fuel in

Table 1. Advantages and disadvantages of NPP power control programs

Control program	Advantages	Disadvantages
t_{av}=const	Favorable conditions for the primary equipment operation (no thermal expansion); the possibility of using the temperature effect of reactivity for reactor control; speed, ease of automation	Steam pressure increase in the steam generator from the secondary system with power decrease at the power unit (deterioration of strength characteristics); occurrence of xenon oscillations; large number of unbalanced water; insertion and removal of solid absorber leads to a curvature of the energy release field; considerable voltage is initiated due to power jump when the control rods move upwards at the traffic boundary in fuel cladding shells located near control rods.
p_{II}=const	Favorable conditions for the operation of steam generating equipment of the secondary system (unchanged thermos-physical characteristics); the possibility of using higher steam parameters at the turbine inlet under rated conditions; speed, ease of automation	The need for a pressurizer of increased dimensions; increased temperature stresses in the reactor vessel, fuel rod shells; In order to compensate the change in reactivity due to the temperature effect, the impact of the control system on the control rods leads to a curvature of the energy release field; large number of unbalanced water; occurrence of xenon oscillations; considerable voltage is initiated due to a power jump in fuel cladding shells located near control relay when the control rods move upwards at the traffic boundary
t_{in}=const	Minimization of the impact on the control rods; minimum consumption of pure distillate and boron concentrate; reduced amount of unbalanced water; improvement of fuel operating conditions; the possibility of using the temperature effect of reactivity for reactor control; the parameters of the energy release field do not change in height with power changes, the absence of xenon oscillations	A small regulation range is limited by permissible pressure in the steam generator; the temperature at the coolant inlet to the reactor, which increases with increasing pressure in MSV, is strictly limited by the Table of permissible modes; there are cyclic changes in the secondary parameters; low speed

the elementary cell of the core,%; $t_{i,w,out}$ — primary coolant temperature at the outlet of the elementary cell in the core, °C.

A multi-zone reactor model is used for further development of a distributed reactor structural model in space, each zone of which is described by a model with lumped parameters. The following assumptions are made in creating a distributed structural reactor model:

each layer in the core is divided into six identical segments;

each segment is divided into four areas, which simulate the fuel cycle depending on the year of operation: from 1 to 4th year, respectively.

Model of nuclear reaction kinetics. The model of the nuclear reaction kinetics accounting for the change of the core isotope composition due to fission of both ^{235}U and ^{239}Pu, which are formed in the core, is described by the following differential equations:

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

$$\frac{dC_{i,j,5}}{d\tau} = \frac{\beta_{j,5}\cdot\Phi_i(\tau)}{l} - \lambda_{j,5}\cdot C_{i,j,5}(\tau), \quad (2)$$

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

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

where Φ_i is neutron flux density averaged in the i-th unit cell of the core, sm^{-2}·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; $\lambda_{j,59}$, $\lambda_{j,9}$ 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 neutron flux density 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, sm^{-2}·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.

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

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

$$\frac{dN_{i,\mathrm{U-9}}}{d\tau} = N_{i,8}\cdot\sigma_{c,8}\cdot\Phi_i - \lambda_{\mathrm{U-9}}\cdot N_{i,\mathrm{U-9}};$$

$$\frac{dN_{i,\mathrm{Np}}}{d\tau} = \lambda_{\mathrm{U-9}}\cdot N_{i,\mathrm{U-9}} - \lambda_{\mathrm{Np}}\cdot N_{i,\mathrm{Np}};$$

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

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

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

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

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

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

$$-\lambda_{\text{Xe}} \cdot N_{i,\text{Xe},9} - \Phi_i \cdot \sigma_{a,\text{Xe}} \cdot N_{i,\text{Xe},9}, \quad (12)$$

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}; $N_{i,\text{Xe},5}$, $N_{i,\,Xe,9}$ 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); $\sigma_{a,\text{Xe}}$, $\sigma_{f,5}$ 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}.

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

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

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.

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

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

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

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, W/(m^2·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.

Reactivity model. The reactivity deviation in a unit cell is

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

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 the deviation of the position of CR, 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 deviation of control rod position in a unit cell is calculated as

$$\delta\rho_{i,r} = \frac{\partial\rho_i}{\partial h_{i,r}}\delta h_{i,r},$$

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 boric acid concentration in the reactor coolant for a unit cell is calculated as

$$\delta\rho_{i,b} = \frac{\partial\rho_i}{\partial C_{i,b}}\delta C_{i,b},$$

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:

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

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 demineralized water is injected, the boric acid concentration deviation is:

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

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

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

$$\delta\rho_{i,N} = \frac{\partial\rho_i}{\partial N}\delta N,$$

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 ^{135}Xe concentration for a unit cell is calculated as follows:

$$\delta\rho_{i,\text{Xe}} = \frac{\partial\rho}{\partial N_{\text{Xe}}}\delta N_{i,\text{Xe}},$$

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

Finally, the reactivity deviation due to a deviation of the reactor coolant temperature, for a unit cell is calculated as follows:

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

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

In addition, the distribution of the core in the elementary cell (i) is applied to equations (1), (2) by allocating the layers in height (y), the symmetry sectors of the core (x) and groups of fuel cells (z) within each sector.

The following equations (3)—(7) were used to calculate the output parameters of the model such as AO, Φ_i, Q_i, $t_{i,w,out}$, $t_{i,f}$:

$$AO = \frac{\left(\sum_{y=6}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(Q_{y,x,z,top}\right)\right) - \left(\sum_{y=1}^{5}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(Q_{y,x,z,bot}\right)\right)}{\left(\sum_{y=6}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(Q_{y,x,z,top}\right)\right) + \left(\sum_{y=1}^{5}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(Q_{y,x,z,bot}\right)\right)} \cdot 100\%, \quad (3)$$

$$\Phi_i = \sum_{y=1}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(\Phi_{y,x,z}\right), \quad (4)$$

$$Q_i = \sum_{y=1}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(Q_{y,x,z}\right), \quad (5)$$

$$t_{i,w,out} = \frac{\sum_{y=6}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(t_{y,x,z,w,out}\right)}{\left(\sum_{y=10}^{10} y\right) + \left(\sum_{x=1}^{6} x\right) + \left(\sum_{z=1}^{4} z\right)}, \quad (6)$$

$$t_{i,f} = \frac{\sum_{y=6}^{10}\sum_{x=1}^{6}\sum_{z=1}^{4}\left(t_{y,x,z,f}\right)}{\left(\sum_{y=10}^{10} y\right) + \left(\sum_{x=1}^{6} x\right) + \left(\sum_{z=1}^{4} z\right)}. \quad (7)$$

As a result of solving the third issue, further development of the simulation model of VVER-1000 reactor as an control object in the Simulink environment was obtained that provides determination of the specified parameters by the sectors of symmetry and high-level layers of the core, as well as groups of fuel cells of each symmetry sector, which takes into account the change in process parameters of the core from division ^{239}Pu and ^{235}U cores.

The automated control method of the change in VVER-1000 power unit

In order to ensure the stable state of the core of VVER-1000 reactor, it is necessary to maintain a constant value of axial offset and, at the same time, control the change in the field of energy separation, which may adversely affect the reactor as a whole due to the intrinsic properties of the core [11, 12]. Therefore, for the first time, the method of automated control of the change in power of VVER-1000 nuclear power unit uses three control loops:

the first loop supports regular change in reactor power due to the maintenance of boric acid concentration in the coolant;

the second loop maintains the necessary value of axial offset by changing the position of the pivots;

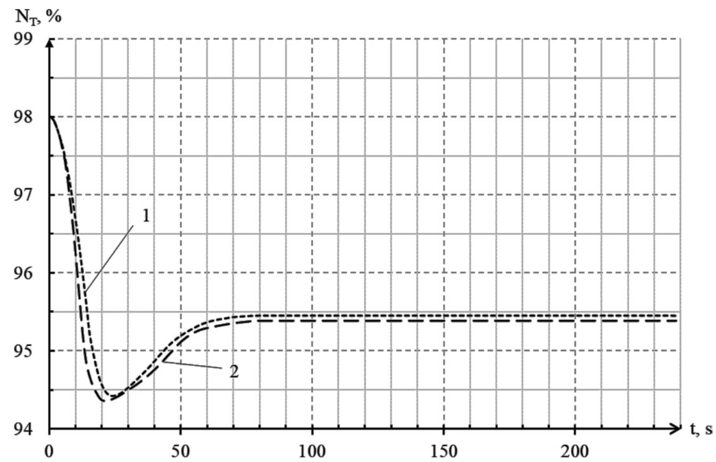

Fig. 4. Influence of the control rod group inserted in the core to change the neutron capacity (N_T) of the unit:
1, 2 —experimental and analytical curve, respectively

the third loop supports the temperature mode in the coolant by controlling the position of the main valves in the turbine generator.

The provisions of the proposed method of automated control of power change at VVER-1000 nuclear power unit are as follows:

the axial offset controller should influence the control rod, and the controller of boric acid concentration should maintain the reactor power;

for the full use of the iodine well effect, the concentration of boric acid should remain unchanged at a reduced power value, that is, the controller should not react to the change in power;

when returning the power unit to its maximum power, the concentration of boric acid should be different from the initial value that was before the maneuver to compensate for changes in ^{135}Xe and ^{135}I concentrations caused by the change in power;

the controller should take into account nonlinear properties of feedback;

operator's participation is not required during controller operation.

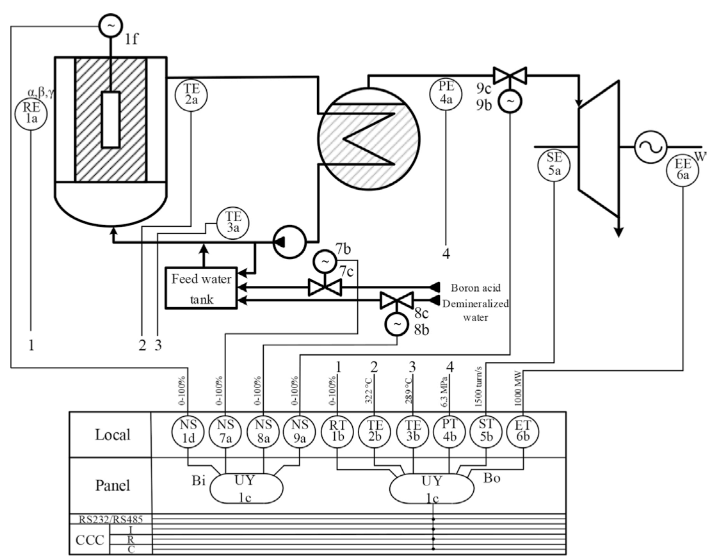

Fig. 5. Enhanced functional diagram of automation of control programs 1 and 2

In order to determine the dynamic characteristics of the main process parameters of the power unit in case of perturbing disturbances, the results of four experiments carried out at South Ukraine NPP unit 3 were used, in which control valve position in the turbine and the control rod group was changed. Since the model adequacy cannot be checked by statistical methods, the discrepancy was obtained in simulation of the data with experimental data (in the course of the experiment, perturbation was caused by the control rod group, control rods moved continuously approximately by 10 % of the height of the core downwards). According to Fig. 4, the average and maximum relative error of simulation is $\delta = 9.44 \cdot 10^{-4}$ and $\delta_{max} = 1.5 \cdot 10^{-3}$.

As a result of using the proposed method of automated control of power change at VVER-1000 nuclear power unit, the functional scheme of automation of two control programs was improved: the control program of the unit with constant average temperature of the primary coolant and the control program of the unit with constant secondary pressure, which are demonstrated in Fig. 5.

Discussion

Comparison of changes in the position of the control rod group, the power unit and axial offset during the daily maneuver of program 1 is presented in Fig. 6—8.

The transient process of changing the generator power during day-to-day program maneuvers 2 when using the proposed automated control system (ACS) is similar with the one presented in Fig. 7. Comparison of change in axial offset and the position of the control rod group in Program 2 is presented in Fig. 9, 10. Comparison of changes in boron acid concentration during the daily maneuver using the proposed ACS for two control programs is presented in Fig. 11.

For the implementation of the improved ASCs by the unit power, the parameters of the PI controllers of axial offset, average temperature of the primary coolant, vapor pressure in the secondary system and unit power were calculated by Kopelovich method.

A universal method for adjusting power change at VVER-1000 nuclear power unit was developed, which allows preventing the occurrence of fluctuations in process parameters and improving the control system by changes in unit power. For the first time, the method of automated control of power change at VVER-1000 nuclear power unit was developed, in which three control loops were applied, one of which supports regular change in reactor power due to the regulation of boric acid concentration in the coolant, the second loop maintains the necessary value of axial offset by changing the control rod position, and the third loop maintains the temperature mode in the coolant by adjusting the position of the main valves in the turbine generator, which improved the stability of the energy output in the core with the change in its power under normal operation conditions of the reactor.

Conclusions

This work contains new scientific positions and results that consist of improved automated control system for changing the power of VVER-1000 nuclear power unit, which allowed to operate NPP in a maneuverable mode and to change the load of the reactor in an automated mode to improve the stability of energy release in the core under normal operating conditions. The following conclusions were made:

1. The advantages and disadvantages of each of the programs were identified resulting from the analysis of existing programs to control power changes at VVER-1000 nuclear power unit. It is noted that no control program considered internal-perturbations that may lead to loss of reactor constancy due to changes in the energy output field. The effect of power change on axial offset was studied to provide efficient and safe control of the unit change in maneuvered modes. It was specified that in order to ensure stable reactor state, the value of axial offset should be maintained within a given interval, which can be considered as a quantitative measure of stability of the processes in the core, and this value should be considered a process parameter.

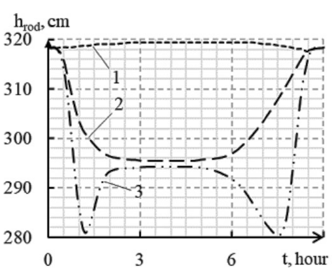

Fig. 6. Changing the position of the control rod group during the daily program maneuver 1 (h_{rod} — height of the control rods, cm):
1 — proposed ACS; *2* — known ACS, developed in [13]; *3* — initial ACS

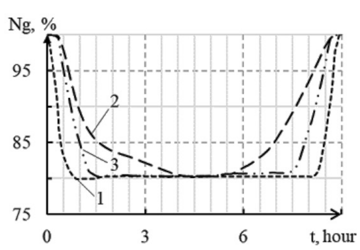

Fig. 7. Change the unit power during daily program maneuvers 1 (N_g — unit power, %):
1 — proposed ACS; *2* — known ACS, developed in [13]; *3* — initial ACS

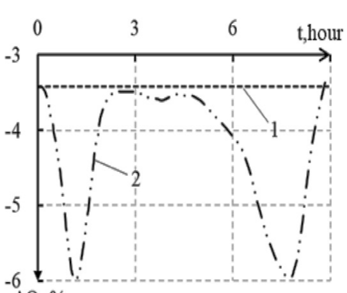

Fig. 8. Change in axial offset during daily program maneuver 1:
1, *2* — proposed and initial ACS, respectively

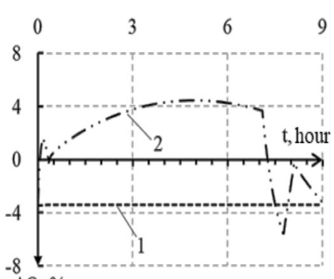

Fig. 9. Change in axial offset during day-to-day program maneuver 2:
1, *2* — pre-installed and initial ACS, respectively

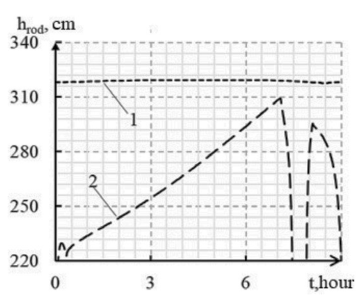

Fig. 10. Changing the regulation of control rod group during the day-to-day negligence program 2:
1, *2* — proposed and initial ACS, respectively

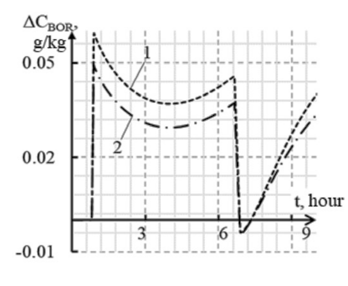

Fig. 11. Change in boric acid concentration during daily maneuver using the proposed ACS (C_{BOR} — boric acid concentration, g/kg):
1, *2* — Program 1 and 2, respectively

2. The developed multi-zone mathematical model of the core in VVER-1000, which allows detailed control of changes in the sectors of symmetry and high-level layers of the core, as well as groups of fuel assemblies of each symmetry sector are important in terms of safety of distributed reactor process parameters.

3. A mathematical model of VVER-1000 nuclear power unit was developed, based on which simulation model of the power unit as control object was developed in the software environment of Simulink software package MATLAB. This simulation model differs from the known one and includes a multi-zone imitative model of distributed-parameter of the core that allows for the inherent properties of the core (including xenon transient processes) to be taken into account and thus reduces the error in the simulation of the static and dynamic properties of the unit, namely: neutron power unit — by 2.54 (from 0.0024 to 0.000944); the coolant temperature at the reactor outlet — by 10.8% (from 0.0011 to 0.001012); the electric power of the unit — by 1.77 (from 0.0017 to 0.000956).

4. The main advantage of the first developed method of automated control of changes in VVER-1000 nuclear power unit is possible simultaneous use of three control loops, one of which supports regular change in reactor power due to the regulation of boric acid concentration in the coolant, the second loop maintains the required value of axial offset by changing the control rod position, and the third one maintains coolant temperature mode by adjusting the position of the main valves in the turbine generator. The use the proposed method of automated control of power change at VVER-1000 nuclear power unit significantly improves the stability of energy output in the core when maneuvering its power under normal operating conditions in the reactor, namely:

for coolant temperature mode with $\langle t \rangle$ = const: the value of axial offset modulus decreases by 1.8 (from 6 to 3.41 %);

for coolant temperature mode with $\langle p_{II} \rangle$ = const: the value of axial offset modulus decreases by 1.3 (from 4.3 to 3.41 %).

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