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ANALYSIS OF MODELS OF AN AUTOMATIC POWER CONTROL SYSTEM FOR A PRESSURIZED WATER REACTOR IN DYNAMIC MODE WITH A CHANGE IN THE STATIC CONTROL PROGRAM

В. Ватаман, Т. Петік, К. Беглов. **Аналіз моделей АСР потужності ВВЕР-1000 в динамічному режимі при зміні статичної програми регулювання.** Процес декарбонізації промисловості посилився рішенням про внесення атомної енергетики до "Зеленої таксономії ЄС". Збільшення потужності АЕС та привабливості цієї галузі на тривалий термін викликає інтерес до використання АЕС для диспетчеризації електроенергії. У такий спосіб стає актуальною розробка АСР зміни потужності АЕС у динамічних режимах. Енерговиділення по висоті активної зони реактора під дією коливань ксенону є однією з проблем при забезпеченні безпечної експлуатації та стійкості енергоблоку АЕС при зміні рівня потужності. Порушення критеріїв безпеки енерговиділення відбувається, коли в активній зоні виникають нестаціонарні перехідні ксенонові процеси з позитивним зворотним зв'язком. У межах теорії автоматичного управління моделювання системи та об'єкта управління складається на основі використання фізико-математичних моделей. Визначаються передавальні функції, структура і параметри системи управління. Як альтернатива пропонується використовувати апроксимуючу модель процесу в активній зоні. Модель представляється на основі перетворень таким чином, щоб результати розрахунку на деякому проміжку збігалися або були близькі до результатів розрахунку при використанні аналітичної моделі. Така модель може апроксимувати процес, який залежить від двох параметрів: часу та ступеня зміни навантаження. Було проаналізовано результати зміни регулювання технологічних параметрів системою автоматизованого керування потужністю ЯЕУ в динамічному режимі в модель якої закладено апроксимаційну або фізико-математичну модель. Результати зміни параметрів отримано на прикладі співставлення результатів при переході керування від одної статичної програми до іншої. Було досліджено перемикання між статичними програмами регулювання за такими технологічними параметри, як температура теплоносія при вході в активну зону реактора, середня температура теплоносія та тиск пари в II-му контурі. При знижені потужності реактору до 80 % при роботі однієї програми регулювання та збільшувалася до 100% потужності при зворотному використанні іншої статичної програми.
Ключові слова: АСР, статична програма регулювання, активна зона ЯЕУ, апроксимаційна модель, фізико-математична модель

V. Vataman, T. Petik, K. Beglov. **Analysis of models of an automatic power control system for a pressurized water reactor in dynamic mode with a change in the static control program.** The process of decarbonization of the industry was intensified by the decision to include nuclear energy in the EU Green Taxonomy. The increase in the capacity of nuclear power plants and the attractiveness of this industry for a long time causes interest in the use of nuclear power plants for electricity dispatching. Thus, the development of an automatic control system for changing the power of nuclear power plants in dynamic modes becomes relevant. The energy release along the height of the reactor core under the action of xenon fluctuations is one of the problems in ensuring the safe operation and stability of the NPP power unit when the power level changes. Violation of the energy release safety criteria occurs when non-stationary transient xenon processes with positive feedback occur in the core. Within the framework of the automatic control theory, the modeling of the system and the control object is compiled on the basis of the use of physical and mathematical models. Transfer functions, structure and parameters of the control system are determined. As an alternative, it is proposed to use an approximation model of the process in the core. The model is presented on the basis of transformations in such a way that the results of the calculation on a certain interval coincide or are close to the results of the calculation when using the analytical model. Such a model can approximate a process that depends on two parameters: time and degree of load change. The results of changing the regulation of technological parameters by the automated power control system of nuclear power unit in dynamic mode, the model of which is based on an approximation or physical and mathematical model, were analyzed. The results of changing the parameters are obtained by comparing the results during the transition of control from one static program to another. Switching between static control programs was studied for such process parameters as coolant temperature at the entrance to the reactor core, average coolant temperature and steam pressure in the secondary circuit. The studies were carried out under the condition that the reactor power was reduced to 80 % during the operation of one control program and increased to 100% of the power when another static program was used in reverse.
Keywords: ACS, static control program, nuclear power unit core, approximation model, physical and mathematical model

1. Introduction

The philosophy of decarbonization of industry in the industrialized countries, and in particular of modern energy, has led to the ever-increasing use of renewable energy sources to meet the demand for energy products of energy systems. Currently, wind and solar generation devices are mainly used as

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them. With the growth of their share, the problem of dispatching energy flows becomes more acute. One of the ways to solve it in the future is the use of hydrogen energy.

One of the locomotives of the process of decarbonization of industry are the countries of the European Union. They have a high percentage of renewable energy sources. Nevertheless, recently there has been an increase (several times) in their imports of electricity from third countries. Against this backdrop, the European Commission has decided to include nuclear power and natural gas in the EU Green Taxonomy. It is a classification of environmentally sound activities for investors. With all the restrictions imposed by this decision, the possibilities of stabilizing the operation of the power system are expanding.

A distinctive feature of "atomic" generation is the duration of the process of controlled change in the reactor power. At the moment, this makes it difficult or even excludes the possibility of using the electricity of nuclear power plants (NPPs) to solve dispatching problems. Significant installed capacity of nuclear power plants, increasing long-term attractiveness in this industry for investors give rise to interest in using the capacity of nuclear reactors for dispatching electricity flows. Thus, the development of automatic control system (ACS) for changing the power of reactors is relevant.

2. Literature analysis and problem statement

The state of energy release of the core as a control object is well represented by the reactivity value and, as a result, its change. Usually, rapid changes in reactivity, which include processes that determine changes in reactor power, temperature of the first circuit coolant, and non-stationary changes in the concentration of toxic substances in the fuel, are taken into account in order to take into account the change in the power of the reactor installation. In case of external influence on the active zone of the reactor in order to change its energy release, an unsteady concentration of ^{135}Xe begins to form. This element causes changes in the distribution of energy release along the height of the core. The reactivity value takes a negative or positive value. This phenomenon is called "xenon oscillations".

Periodic redistribution of energy release along the height of the reactor core under the action of xenon oscillations is one of the key problems in ensuring the safe operation and stability of the power unit with pressurized water reactor (PWR) when the power level changes. Violation of safety criteria during energy release usually occurs when non-stationary xenon transients with positive feedback appear in the active zone [1]. After the reactor start-up, it takes 12 hours to reach the equilibrium concentration of ^{135}Xe at 0.99, and about 18 days are required to reach the equilibrium concentration of samarium at 0.99. The oscillation amplitude of the xenon process is almost an order of magnitude higher than that of the samarium process. Changes in the ^{149}Sm concentration are comparable in time scale, rate of influence on reactivity with the process of nuclear fuel burnup, and do not affect the stability of energy distribution in the core. It is shown in the literature [2] that the change in reactivity due to xenon transient processes is decisive in terms of the degree of influence on safety and is considered separately from samarium and regardless of the process of fuel burnup.

The current regulation of the power plant with pressurized water reactor allows its operation in the mode of power stabilization from current disturbances at a given level with a deviation of 2.5% in any direction, but not more than the design power. Therefore, the physical and technical basis of the existing methods for controlling the core power is to change the power level of the core with the unconditional fulfillment of safety criteria.

Changing the power level in excess of the specified range is carried out by the plant operators directly in the remote mode and only at the command of the power system dispatchers. The execution of such a command, first of all, entails the occurrence of nonlinear dynamic processes, including oscillatory ones in process equipment. The operator must simultaneously ensure the control of changing neutron-physical properties in the core space and the compliance of the process parameters of the power plant, with the current power generation.

Regardless of the method of core power control, oscillations always occur due to the spatial redistribution of the neutron flux. Within the framework of the automatic control theory, the modeling of the system and the control object is built on the base of differential models. The transfer functions (or their analogues) for the models used, the structure and parameters of control systems are determined. There are various models describing the processes inside the reactor. They can be described, for example, using ordinary differential equations [3], or using models based on fractional order differential equations [4]. The level of development of computer technology, which is approved for the control of a nuclear reactor, does not allow real-time calculations based on such models. As a result, there is no

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possibility of organizing automatic control of the change in the power of the core during the transition from one state to another.

It should be noted the proposed in some cases options for possible automatic control systems based on simplified models [5]. However, traditionally in operational practice, for each campaign, operators are provided with nomograms, which present the values of the amplitudes of oscillations in changing reactivity. They display the dependence of the change in reactivity on the effective burnup time of nuclear fuel, the current power of the core and the increase or decrease in the level of power change. Nomograms about the properties of the core are built on the basis of the results of a preliminary solution by numerical methods of the complete system of nonlinear differential equations of the model of in-reactor processes. A contradictory situation arises. On the one hand, there is a complete system of differential equations and the results of its solution (in particular, displayed in the form of nomograms). On the other hand, automatic control systems are built on the basis of a simplified model, or control is carried out manually based on pre-calculated nomograms.

In a number of studies, it was proposed to use an approximation model of the process. The approximation and simplified models have a difference. A simplified model is based on ranking the influencing factors and discarding, according to the researcher, the least significant ones. The subjective factor has a significant influence. Important information may be missing. The approximation model is built on the basis of the results of calculations for the complete model [6]. In this case, no information is discarded. The model is built on the basis of formal transformations in such a way that the calculation results at a certain time interval coincide or are close to the calculation results when using the complete model. In this case, the approximation model consists of a set of linear ordinary differential equations of the first order or, respectively, of an ordinary linear differential equation of the corresponding order. The order of the equation can be selected. The limiting property of this method in the form [7] is the use of a single response of the system to an external impact. In other words, a model constructed in this way can approximate a process that depends on changing only one argument, such as time. At the same time, nomograms of changes in the amplitude of reactivity will be built depending on two parameters: time and degree of change (leap) in the load. Thus, it is of interest to compare models of an ACS with a complete physical and mathematical model and an approximation one.

3. Research aim and objectives

The aim of the study is to analyze the results of changing the controlled parameters of the automated power control system of a nuclear power unit in a dynamic mode, the model of which contains an approximation or physical and mathematical model of changing parameters by comparing the control results when changing static control programs.

To achieve this goal, the following tasks were set:

– to analyze the current state of operation of the automated power control systems of a nuclear power unit in various static control programs;

– to study the models of the ACS at the change of control parameters, when the properties are calculated on the basis of: a 3-dimensional mathematical model of a nuclear reactor of the PWR type, that includes the distribution of the active zone on an elementary cell and an approximation model that will correspond to the results of solving a system of nonlinear differential equations, which calculates the properties of the core.

4. The main part

4.1 The current state of operation of the ACS for the power of a nuclear power unit in various static control programs

Power units with PWR reactors are double-circuit, so the energy from the reactor to the turbogenerator is transferred using two working fluids: the primary coolant and steam. The interface between the circuits is the steam generator in which saturated steam is generated. The imbalance between the generated and consumed energy is manifested in a change in the parameters of the working fluids, namely, in oscillations in the steam pressure before the turbine and in the temperature of the primary coolant. For such power units, the nature of the change in the parameters of the working media in the primary and secondary circuits (pressure, temperature, flow rate) with a change in power in static operating modes is important. This dependency is called a control program. The choice of a static control program depends on many factors, including the mode of use of the power unit in the power system.

A feature of the operation of a nuclear power unit with a PWR is the search for a compromise when choosing the parameters of process variables. On the one hand, in order to be able to change the power of the power unit, it is necessary to have a margin of steam pressure at low power unit operation. As the power increases, the steam pressure before the turbine decreases and reaches the nominal value at maximum power [8]. This mode is favorable for the primary circuit, as it provides stable temperature conditions for its equipment. On the other hand, such a mode is not economical, since the thermal efficiency of the cycle remains low at all powers. This is explained by the fact that the steam pressure before the turbine is low at rated power, and at reduced powers the steam pressure, although high, is throttled in the turbine control valves. In addition, the dependence of the steam saturation temperature on pressure leads to the temperature feedback that affects the coolant temperature at the reactor inlet. It is a disturbing action, because there is a so-called temperature coefficient of reactivity [9].

Another control method is to stabilize the steam pressure in the secondary circuit [10]. Obviously, this is favorable for the equipment of the secondary circuit. However, the desire to stabilize the steam pressure and, consequently, the saturation temperature of the steam in the secondary circuit leads to the fact that the coolant temperature at the reactor inlet changes to a lesser extent, but with an increase in power, the average temperature of the primary coolant increases significantly. A significant change in the average temperature limits the maneuverability of the power unit and complicates the operation of the automatic pressure control system in the volume compensator.

Currently, power units are operated according to a combined-compromise program, according to which, depending on the power, one or another mode of maintaining process parameters is selected [11]. These programs are shown on Fig. 1. Vertical lines located at the power level of 80 % indicate transitions from one control program to another when they are switched.

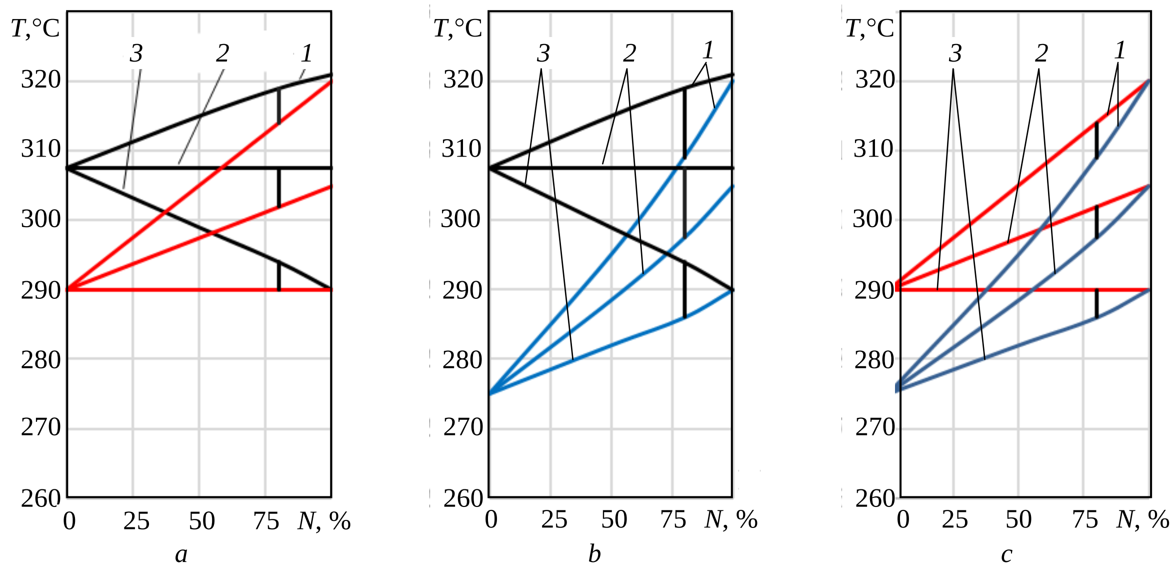

Fig. 1. Characteristics of a nuclear power unit with a pressurized water reactor: *a* – when switching $T_{av} = \text{const}$ and $T_{in} = \text{const}$; *b* – when switching $T_{av} = \text{const}$ and $P_2 = \text{const}$; *c* – when switching $T_{in} = \text{const}$ and $P_2 = \text{const}$; *1* – temperature of coolant at the core outlet, t_{out}; *2* – average temperature of coolant in the core, t_{av}; *3* – temperature of coolant at the core inlet, t_{in}

Numerous studies show that the operation of nuclear power units with pressurized water reactors is possible if certain measures are taken to improve the reliability of fuel claddings [12] and reduce the operator’s involvement in operational actions during power maneuvering [13].

One of these measures is the improvement of the automatic power controller (APC) of the power unit. One of the tasks of a regular APC is to maintain the set process parameters in accordance with the selected control program. A number of works [14] are also devoted to the improvement of the controller. It has been shown [15] that for reliable operation of a nuclear power plant, the APC must take into account the nonlinear properties of the core. Various controller schemes have been investigated, including cascaded, fuzzy, and adaptive ones. The PI control law unites all these schemes. This is explained by the requirement of the NPP operation regulations for the reliability of control algorithms. The PI control law cannot take into account changes in the properties of the control object, therefore, when the operating modes of the power unit change, it is necessary to adjust the settings of the power controller.

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The problem of adjusting the controller settings is the complexity of calculating the mathematical model of the control object for predicting its behavior. It is impossible to apply such an approach to correct the settings of the regulators, since the calculation of the complete mathematical model on standard NPP computers takes about 20 hours. Therefore, during the operation of the power unit, nomograms are calculated in advance to calculate both the settings of the regulators and the values of the process parameters that must be maintained in a given mode of operation of the power unit. This approach is fraught with errors when performing functional switching.

The way out of this situation was the development of such a mathematical model that, on the one hand, adequately describes the nonlinear properties of the core. On the other hand, the calculation according to this model is performed fast enough to be used for real-time recalculation of the controller parameters. Such a model was developed and studied in [16].

Simulation modeling showed an increase in the quality of transient control processes during daily and weekly power unit maneuvering. However, the simulations were carried out for one control program.

As shown in a number of works [17], for reliable operation of fuel elements during the operation of a nuclear power plant, it is necessary to switch from one static control program to another. This improves the operating conditions of fuel elements, but worsens the quality of transient processes for regulating the main process parameters of the power unit.

This work is devoted to the verification of the approximation model and the study of the modernized control system when switching control programs.

4.2 Study of ACS models with a change in control parameters

The simulation model performed in this work is based on the mathematical model of NPP with pressurized water reactor, presented in [18], and its adequacy was confirmed and compared with the experimental data of the South Ukraine NPP in [19, 20]. The difference between calculations and experimental data is 1 %. Where the total error was the error of the original data and the calculated error of the numerical methods.

Mathematical models of control sections of a nuclear power unit are presented in the form of specialized software and are regulated by norms and standards [21]. The simplest linearized mathematical models are used to regulate local controllers. The most sophisticated software is used to calculate the neutronic properties of the reactor core. Even such a model is a system of non-linear differential equations. Its calculations are used to predict the state of the reactor and at the same time take into account the entire history of the operation of the current nuclear fuel campaign [22]. To control a nuclear reactor, the operator uses special nomograms, which are designed according to preliminary calculations using a mathematical model and regulations that indicate the procedure depending on the values of process parameters. However, even with the use of nomograms and regulations, the operator can still make a mistake. In this study, a model [23] was assembled, which included the developed approximation model that describes the change in reactivity under the influence of xenon poisoning depending on the magnitude of the disturbance.

The approximation model of a nuclear power unit was based on mathematical models of a nuclear reactor [24] and technical equipment of the secondary circuit [25]. As a control object for a nuclear power unit, a simulation model of the following type is presented:

$$\text{control object physical}\,(\delta h_{\mathrm{SCV}};\, \delta h_{y,x,z,r};\, \delta C_{\mathrm{Bor}}^{y,x,z};\, N_{\mathrm{e}}) = \mathrm{AO};\, Q_{y,x,z};\, \delta t_{r,\mathrm{hc,out}};\, \delta t_{r,w,\mathrm{av}};\, P_{\mathrm{st}}\,.$$

This simulation model is shown in Fig. 2 and additionally includes:

– Steam generator model:

$$SG(t_{\mathrm{sg,hc,in}}(\tau);\, G_{\mathrm{sg,hc,out}}(\tau)) = t_{\mathrm{sg,hc,out}}(\tau);\, P_{\mathrm{s}}(\tau)\,;$$

– Delay model for the movement of coolant circulating in pipelines from the reactor to the steam generator and then moving in the opposite direction:

$$PL(t_{\mathrm{sg,hc,in}}(\tau);\, t_{r,\mathrm{hc,out}}(\tau)) = t_{r,\mathrm{hc,in}}(\tau);\, t_{\mathrm{sg,hc,in}}(\tau)\,;$$

– Turbine generator model:

$$TG(P_{\mathrm{st}}(\tau);\, N_{\mathrm{e}}) = G_{\mathrm{s}}(\tau);\, N_{t}(\tau)\,.$$

As a representation of intrinsic feedbacks in the control object, a part of the object is represented as interrelated functions. Where $\Phi_{y,x,z} = f(\delta\rho_{y,x,z})$ is a function of the dependence of the change in the density of the neutron flux on the change in the total reactivity. $\delta Q_{y,x,z} = f(\delta\Phi_{y,x,z}, t_f^{y,x,z})$ is a function of the dependence of the change in energy release on the density of the neutron flux and the temperature of the fuel. A function indicating the dependence of the change in the average temperature of the coolant of the primary circuit on the change in energy release and the temperature of the coolant of the primary circuit at the inlet to the nuclear reactor: $\delta t_{r,\text{hc,avr}} = f(\delta Q_{y,x,z}, t_{r,\text{hc,in}})$. The function of the dependence of the change in the temperature of the coolant of the primary circuit at the reactor outlet on the change in energy release and the temperature of the coolant of the primary circuit at the reactor inlet: $\delta t_{r,\text{hc,out}} = f(\delta Q_{y,x,z}, t_{r,\text{hc,in}})$. The function of the dependence of the fuel temperature change on the change in the energy release and the temperature of the coolant of the primary circuit at the reactor inlet: $\delta t_{y,x,z} = f(\delta Q_{y,x,z}, t_{r,\text{hc,in}})$. The function of the dependence of the change in reactivity in case of poisoning of the core with xenon Xe on the change in the neutron flux density: $\delta\rho_{y,x,z,\text{Xe}} = f(\delta\Phi_{y,x,z})$. The function of dependence of reactivity change on energy release change: $\delta\rho_{y,x,z,\text{N}} = f(\delta Q_{y,x,z})$. The function of the dependence of the reactivity change on the change in the average coolant temperature: $\delta\rho_{y,x,z,t} = f(\delta t_{r,\text{hc,avr}})$. The function of the dependence of the reactivity change on the change in the concentration of boric acid in the coolant of the primary circuit: $\delta\rho_{\text{Bor}}^{y,x,z} = f(\delta C_{\text{Bor}}^{y,x,z})$. The dependence function of the reactivity change introduced by the regulatory group of regulatory authorities: $\delta\rho_{y,x,z,r} = f(\delta t_{y,x,z,r})$.

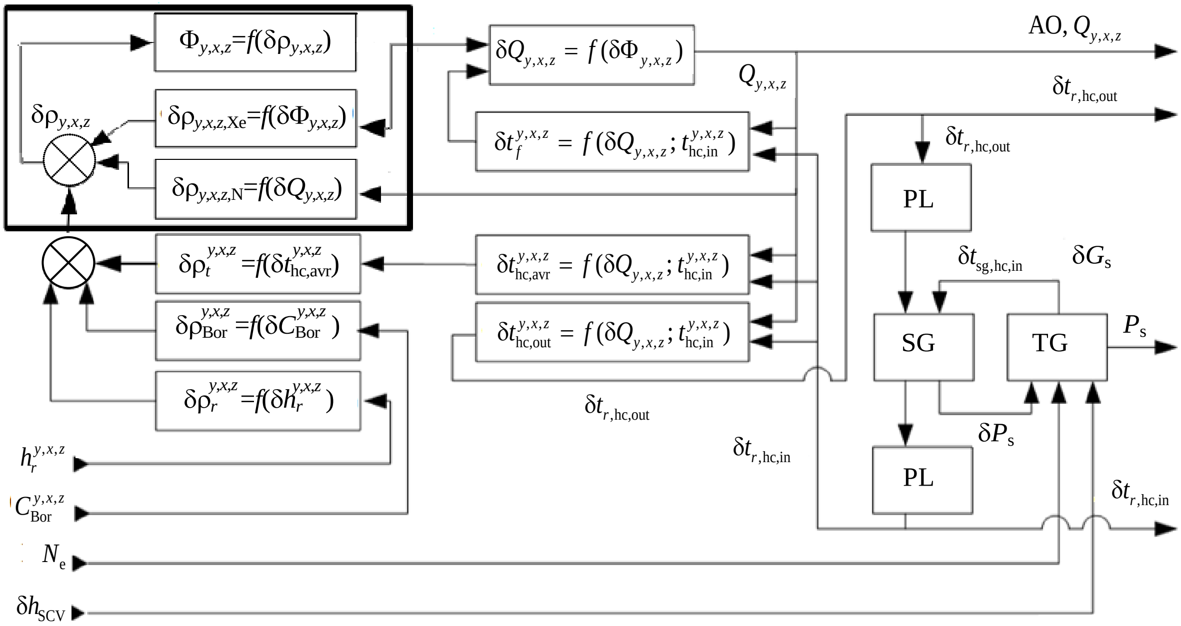

Fig. 2. Simulation model of a nuclear power unit [22]

The simulation model of a nuclear power unit (Fig. 3) has the following notation: $\delta h_r^{y,x,z}$ – deviation of the position of the control rods; $\delta C_{\text{Bor}}^{y,x,z}$ – deviation of the concentration of boric acid in the coolant; N_e – electrical power of the equipment; AO – axial offset; $t_{r,\text{hc,out}}$ – temperature of the coolant of the primary circuit at the nuclear reactor outlet; $t_{r,\text{hc,in}}$ – temperature of the coolant of the primary circuit at the nuclear reactor inlet; $t_{r,\text{hc,avr}}$ – average temperature of the coolant of the primary circuit; P_s – saturated steam pressure at the steam turbine inlet; $t_{\text{sg,hc,in}}$ – temperature of the coolant of the primary circuit at the steam generator inlet; G_s – steam consumption; $t_{\text{sg,hc,out}}$ – temperature of the coolant of the primary circuit at the steam generator outlet; i – value of the required parameter for the design group; $Q_{y,x,z}$ – energy release in the core fuel cell; $t_{\text{hc,out}}^{y,x,z}$ – temperature of the coolant of the primary circuit at the core fuel cell outlet; δh_{SCV} – deviation of the position of the stop and control valve.

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By calculating the complete mathematical model:

$$\text{control object physical}\,(\delta h_{\text{SCV}};\, \delta h_{y,x,z,r};\, \delta C^{y,x,z}_{\text{Bor}};\, N_{\text{e}}) = \text{AO};\, Q_{y,x,z};\, \delta t_{r,\text{hc,out}};\, \delta t_{r,w,\text{av}};\, P_{\text{s}}\,.$$

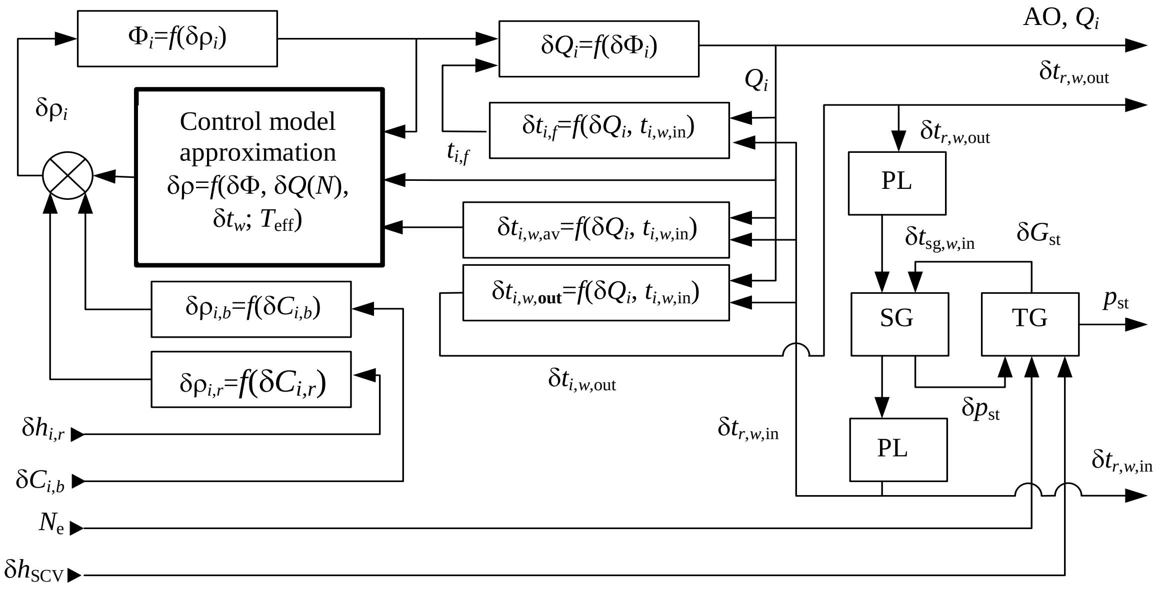

Fig. 3. Approximation model of a nuclear power unit

We get the values $\delta\rho_{y,x,z,\text{N}} = f(\delta Q_{y,x,z})$, $\delta\rho_{y,x,z,\text{Xe}} = f(\delta\Phi_{y,x,z})$ and $\Phi_{y,x,z} = f(\delta\rho_{y,x,z})$, which describe the state of the reactor core at a stationary power level and minimum error [26, 27]. In turn, these values cannot be used in operationalcontrol during power maneuvering, since a lot of time is required for calculations. The approximation model was used to solve this problem:

$$\text{control object approximation}\begin{pmatrix}\delta h_{\text{SCV}}; & \delta h_{y,x,z,r};\\ \delta C^{y,x,z}_{\text{Bor}}; & N_{\text{e}}\end{pmatrix} = \text{AO};\, Q_{y,x,z};\, \delta t_{r,\text{hc,out}};\, \delta t_{r,w,\text{av}};\, P_{\text{s}}\,.$$

Since it is impossible to measure the change in reactivity caused by a change in the xenon concentration in a working reactor, an automated control system for the energy release of a nuclear power unit is synthesized, taking into account internal disturbances caused by xenon oscillations.

To ensure a stable state of operation of the reactor of a nuclear power unit, it is necessary to maintain a constant value of AO and simultaneously control the change in the power density field. To do this, the ACS has three control loops [28]: the 1st one maintains the set value or changes (at the operator's instruction) the reactor power by controlling boric acid in the coolant of the primary circuit; the 2nd one maintains the value of the AO due to the position of the control rods of the control and protection system; the 3rd one maintains the value of the set process parameter (temperature or pressure) of the coolant of the secondary circuit by controlling the position of the control valves of the turbogenerator.

It is not possible to directly measure the change in reactivity caused by a change in the xenon concentration in the operating reactor. Therefore, an automated control system for the energy release of a nuclear power unit is being synthesized, which takes into account internal disturbances due to the occurrence of "xenon oscillations".

There are a number of limitations in the power regulator of operation algorithm:

– to obtain the "iodine pit" effect, the concentration of boric acid must remain unchanged when the power is reduced, that is, the regulator must not respond to power changes;

– when the power unit returns to maximum power, the concentration of boric acid must differ from the initial value that was before the maneuver. This is necessary to compensate for the change in ^{135}Xe and ^{135}I concentrations caused by the maneuver;

– during the operation of controllers, operator participation is not required.

Therefore, a cascade control scheme was adopted. The internal controller maintains the set value of the boric acid concentration, the external (corrective) sets the concentration value depending on the power of the nuclear power unit. In this case, the corrective controller takes into account the change in the properties of the reactor core caused by the load value.

Figure 4 shows a block diagram of an automatic power change control system for a nuclear power unit.

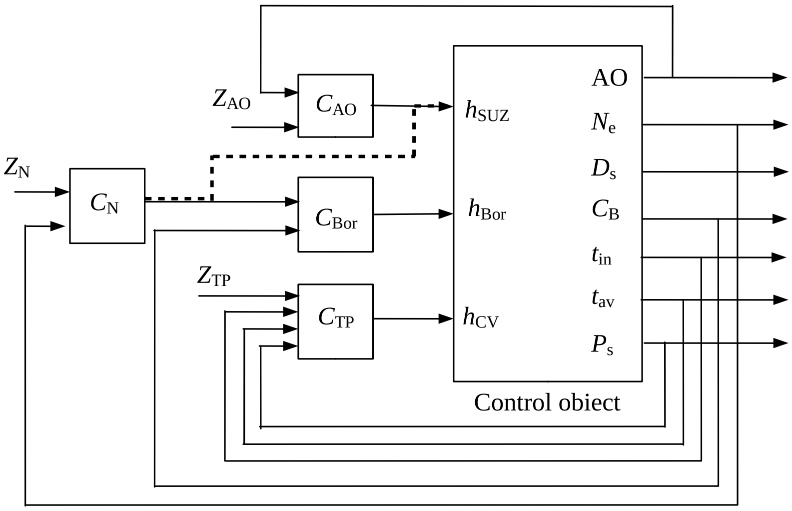

Fig. 4. Structural diagram of the automatic control system for power change of a nuclear power plant

The following designations are used in the figure:

AO – axial offset;

N_e – electric power;

D_s – steam flow rate;

C_B – concentration of boric acid;

t_{in} – temperature of coolant at the reactor inlet;

t_{av} – average temperature of the primary coolant;

P_s – steam pressure.

C_{AO}, C_N, C_{Bor}, C_{TP} – axial offset, power, boric acid concentration and process parameter controllers, respectively;

Z_{AO}, Z_N, Z_{TP} – set values of axial offset, power and process parameter, respectively;

h_{SUZ}, h_{Bor}, h_{CV} – positions of the control and protection system, boric acid inlet valve and turbine control valves, respectively.

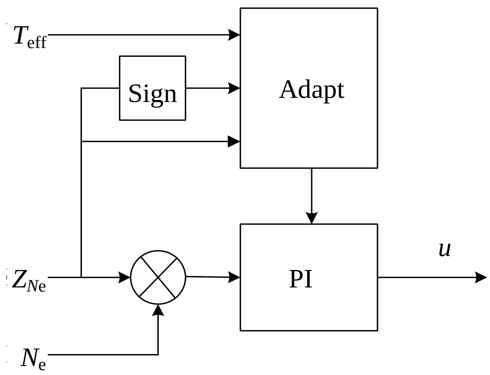

Fig. 5. Structure of the corrective controller C_N

The corrective power controller C_N is synthesized on the basis of a standard PI control law (Fig. 5). The specified and current values of electric power are fed to the controller input, on the basis of which the value of the control action is calculated. This control action is applied as input to the boric acid concentration controller C_{Bor}. The main difference of the C_N power controller is the adjustment of the k_P and T_1 settings. To do this, based on the signals about the effective time of fuel operation (Teff), the value and sign of the change in electric power in the Adapt block, the k_P and T_1 settings are calculated. The calculation is based on the method of obtaining transfer functions, described in [29].

After determining the settings of the C_N and C_{Bor} controllers, simulation modeling of the automatic power control system of the NPP power unit was carried out.

5. Research results

Figures 6 – 11 show a comparison of changes in process parameters (coolant temperature at the reactor core inlet – t_{in}, average coolant temperature – t_{av}, steam pressure in the secondary circuit – p_2 and axial offset – AO) during bumpless switching [30] of control programs (blue), when using an approximation model (orange), and during direct and reverse switching.

The results showed that with shockless switching of control programs, the coolant temperature at the core inlet changed by 5 relative units per 100 s with direct switching, and by 4 relative units with reverse switching, the control time of which was 115 s. When using the approximation model, the

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temperature of the coolant at the inlet to the reactor core changed by 6 relative units in 130 s with direct switching and by 5 relative units with reverse switching, and the control time was 120 s.

The results showed that with shockless switching of programs, the maximum deviation of steam pressure with direct switching of programs was 0.45 relative units and 0.3 relative units – with the reverse. The transition process time in both cases was 100 s. When using the approximation model, the deviation of the steam pressure in the secondary circuit is 0.45 relative units for 70 seconds with direct switching and 0.4 relative units for 97 seconds with reverse switching of control programs.

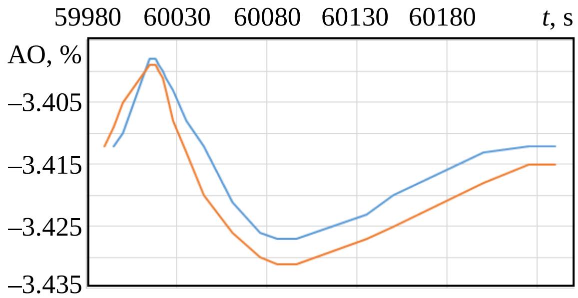

Fig. 6. Response of axial offset (AO) to direct switching of power control programs in the case of $t_{\text{hc,avr}} - p_2$

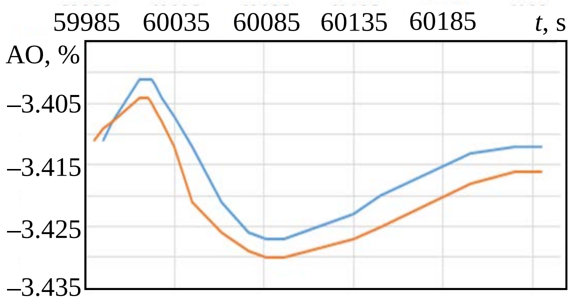

Fig. 7. Response of axial offset (AO) to direct switching of power control programs in the case of $t_{\text{hc,in}} - p_2$

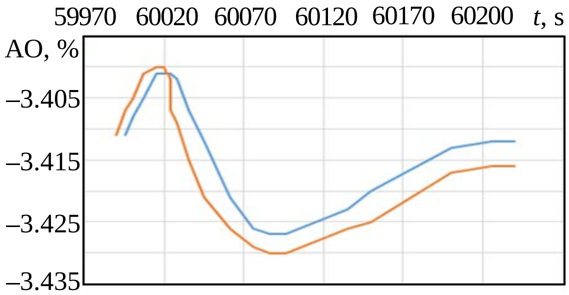

Fig. 8. Response of axial offset (AO) to direct switching of power control programs in the case of $t_{\text{hc,avr}} - t_{\text{hc,in}}$

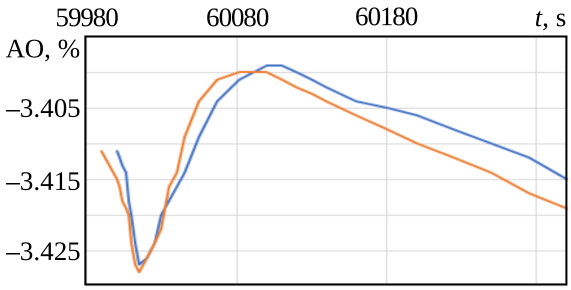

Fig. 9. Response of axial offset (AO) to reverse switching of power control programs in the case of $p_2 - t_{\text{hc,avr}}$

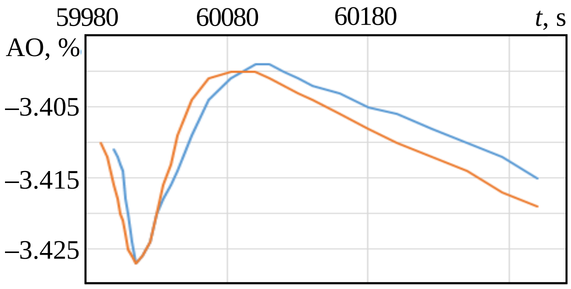

Fig. 10. Response of axial offset (AO) to reverse switching of power control programs in the case of $t_{\text{hc,in}} - t_{\text{hc,avr}}$

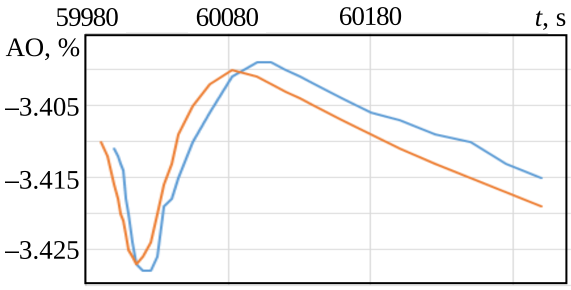

Fig. 11. Response of axial offset (AO) to reverse switching of power control programs in the case of $p_2 - t_{\text{hc,in}}$

The value of the axial offset in the shockless switching mode shows the maximum deviation of the axial offset (AO) of 0.015 % for direct switching and 0.027 % for reverse switching, and the control time is 245 seconds for direct switching and 295 seconds for reverse switching. When using an approximation model, the maximum deviation of the axial offset is 0.014 % for 240 s for direct switching and 0.024 % for 300 s for reverse switching.

6. Conclusions

1. A study was considered of switching static control programs according to such process parameters as the temperature of the coolant at the inlet to the reactor core, the average temperature of the coolant and the steam pressure in the secondary circuit. The following combinations were applied: from $t_{\text{hc,avr}} = \text{const}$ to $t_{\text{hc,in}} = \text{const}$; from $t_{\text{hc,in}} = \text{const}$ to $t_{\text{hc,avr}} = \text{const}$; from $t_{\text{hc,avr}} = \text{const}$ to $p_2 = \text{const}$; from $p_2 = \text{const}$ to $t_{\text{hc,avr}} = \text{const}$; from $t_{\text{hc,in}} = \text{const}$ to $p_2 = \text{const}$; from $p_2 = \text{const}$ to $t_{\text{hc,in}} = \text{const}$ when the reactor power is reduced to 80 % during the operation of one control program and the power is increased to 100 % during the operation of another static program.

2. Controllers with smooth switching of control modes are used, with the help of which a sharp jump is eliminated in the simulation by adding additional feedback. The feedback of each of the controllers eliminates the regulation error, which makes it possible to obtain a response to switching programs in the range of permissible deviations.

Література

1. Jiang Q., Liu Y., Zeng W., Yu T. Study on switching control of PWR core power with a fuzzy multimodel. *Annals of Nuclear Energy*. 2020. 145. 107611. DOI: https://doi.org/10.1016/j.anucene.2020.107611.
2. State-space model predictive control method for core power control in pressurized water reactor nuclear power stations / Wang G., Wu J., Zeng B., Xu Z., Wu W., Ma X. *Nuclear Engineering and Technology*. 2016. 49. 134–140. DOI: http://dx.doi.org/10.1016/j.net.2016.07.008.
3. Offset-cardinality phase diagram method of controlling reactor power / Aver'yanova S.P., Vokhmyanina N.S., Zlobin D.A., Filimonov P.E., Kuznetsov V.I., Lagovskii V.B. *Atomic Energy*. 2017. 121. 155–160. DOI: https://doi.org/10.1007/s10512-017-0176-5.
4. Comparison of Two Control Programs of the VVER-1000 Nuclear Power Unit Using Regression Data Mining Models / Foshch T., Machado J., Portela F., Maksimov M., Maksimova O. *Nuclear and Radiation Safety*. 2017. 3 (75). 11–17. DOI: https://doi.org/10.32918/nrs.2017.3(75).02.
5. Foshch T., Maksimov M., Pelykh S., Maksimova O. Models and methods for automated control of power change at VVER-1000 nuclear power unit. *Nuclear and Radiation Safety*. 2018. 1 (77). 24–30. URL: http://dspace.opu.ua/jspui/bitstream/123456789/10580/1/ydpb_2018_1_6.pdf.
6. Designing an automated control system for changing NPU energy release compensating for arising internal disturbing factors based on their approximation model / Brunetkin O., Beglov K., Maksymov M., Baskakov V., Vataman V., Kryvda V. *Eastern-European Journal of Enterprise Technologies*. 2022. 3 (2 (117)). 63–75. DOI: https://doi.org/10.15587/1729-4061.2022.258394.
7. Kokol E. A. Structural optimization of static power control programs for nuclear power plants with WWER-1000. *Proceedings of Odessa Polytechnic University*. 2015. 3 (47). 41–46. DOI: https://doi.org/10.15276/opu.3.47.2015.07.
8. A graph-theory-based dynamic programming planning method for distributed energy system planning: Campus area as a case study / Ding Y., Wang Q., Tian Z., Lyu Y., Li F., Yan Z., Xia X. *Applied Energy*. 2023. 329. 120258. DOI: https://doi.org/10.1016/j.apenergy.2022.120258.
9. Fathy E., Hassanien A. E. Fuzzy harmonic mean technique for solving fully fuzzy multilevel multiobjective linear programming problems. *Alexandria Engineering Journal*. 2022. 61 (10). 8189–8205. DOI: https://doi.org/10.1016/j.aej.2022.01.021.
10. Plakhotnuk A.A., Kokol E. A., Maksimov M. V. Improved PCS software switch control power generating. *Automation Technological and Business-Processes*. 2015. 7 (3). 26–33. DOI: https://doi.org/10.15673/2312-3125.4/2015.50429.
11. Pelykh S. N., Maksimov M. V., Parks G. T. A method for VVER-1000 fuel rearrangement optimization taking into account both fuel cladding durability and burnup. *Nuclear Engineering and Design*. 2013. 257. 53–60. DOI: https://doi.org/10.1016/j.nucengdes.2012.12.022.
12. Maksymov M., Alyokhina S., Brunetkin O. Thermal and reliability criteria for nuclear fuel safety. River Publishers (Verlag). 2021. 260 p. e-ISBN: 9788770224000.
13. Maksimov M.V., Tsiselskaya T.A., Kokol E.A. The Method of Control of Nuclear Power Plant with VVER-1000 Reactor in Maneuverable Mode. *Journal of Automation and Information Sciences*. 2015. 47 (6). 17–32. DOI: https://doi.org/10.1615/JAutomatInfScien.v47.i6.20.
14. Modeling and control of nuclear reactor cores for electricity generation. A review of advanced technologies / Li G., Wang X., Liang B., Li X., Zhang B., Zou Y. *Renewable and Sustainable Energy Reviews*. 2016. 60. 116–128. DOI: https://doi.org/10.1016/j.rser.2016.01.116.
15. Maksimov M. V., Pelykh S. N., Gontar R. L. Principles of controlling fuel-element cladding lifetime in variable VVER-1000 loading regimes. *Atomic Energy*. 2012. 112 (4). 241–249. DOI: https://doi.org/10.1007/s10512-012-9552-3.

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

16. Foshch T., Portela F., Machado J., Maksimov M. Regression Models of the Nuclear Power Unit VVER-1000 Using Data Mining Techniques. *Procedia Computer Science*. 2016. 100. 253–262. DOI: https://doi.org/10.1016/j.procs.2016.09.151.
17. Robust power control design for a small pressurized water reactor using an H infinity mixed sensitivity method / Yan X., Wang P., Qing J., Wu S., Zhao F. *Nuclear Engineering and Technology*. 2020.52 (7). 1443–1451. DOI: http://doi.org/10.1016/j.net.2019.12.031.
18. Filimonov P. E., Aver'yanova S. P., Filimonova M. P. Control of control-rod groups in the maneuvering regime of VVÉR-1000 operation. *Atomic Energy*. 1998. 84 (5). 383–387. DOI: https://doi.org/10.1007/BF02414876.
19. Kokol. E.A. Optimal power control of VVER-1000 due to targeted selection of the control program. *Automation 2015: XXII International Conference on Automatic Control*. Odessa: Ukraine. 2015. p. 119–120.
20. Plakhotnuk A. A., Kokol E. A., Maksimov M. V. Simulation of changes in the structure of technical means of automation at the VVER-1000 while maneuvering. *Automation Technological and Business-Processes*. 2015. 7 (4). 64–71. DOI: https://doi.org/10.15673/2312-3125.24/2015.56334.
21. Ningbo L., Xueyao S. Research on the control of containment pressure of Gen II+ NPP under severe accident condition. *Energy Procedia*. 2017. 117. 220–224. DOI: https://doi.org/10.1016/j.egypro.2017.08.098.
22. Zhou H., Pelykh S. N., Odrekhovska I. O., Maksymova O. B. Optimization of power control program switching for a WWER-1000 under transient operating conditions. *Problems of Atomic Science and Technology*. 2018. 1 (113). 218–221. URL: http://dspace.nbuv.gov.ua/bitstream/handle/123456789/137360/36-Zhou.pdf?sequence=1.
23. Pelykh S. N., Maksimov M. V., Ryabchikov S. D. The prediction problems of VVER fuel element cladding failure theory. *Nuclear Engineering and Design*. 2016. 302. 46–55. DOI: https://doi.org/10.1016/j.nucengdes.2016.04.005.
24. Pelykh S. N., Maksimov M. V., Nikolsky M. V. A method for minimization of cladding failure parameter accumulation probability in VVER fuel elements. *Problems of Atomic Science and Technology*. 2014. 92 (4). 108–116. URL: https://vant.kipt.kharkov.ua/ARTICLE/VANT_2014_4/article_2014_4_108.pdf.
25. Beglov K. V., Odrekhovska Y. O., Petik T. V., Vataman V. V. A method for searching the best static program for nuclear power unit control in the event of perturbations of different nature. *Herald of Advanced Information Technology*. 2023. 6 (2). 139–151. DOI: https://doi.org/10.15276/hait.06.2023.9.
26. Pelykh S. N., Maksimov M. V. Theory of VVER-1000 fuel rearrangement optimization taking into account both fuel cladding durability and burnup. *Problems of Atomic Science and Technology*. 2013. 44 (2) 84. 50–54. URL: http://dspace.nbuv.gov.ua/bitstream/handle/123456789/111678/08-Pelykh.pdf?sequence=1.
27. Pelykh S. N., Maksimov M. V. The method of fuel rearrangement control considering fuel element cladding damage and burnup. *Problems of Atomic Science and Technology*. 2013. 87 (5). 84–90. URL: https://vant.kipt.kharkov.ua/ARTICLE/VANT_2013_5/article_2013_5_84a.pdf.
28. Maksimov M. V., Kanazirskyi N. F., Kokol E. A. Control of the axial offset in a nuclear reactor at power maneuvering. *Proceedings of Odessa Polytechnic University*. 2014. 2 (4). 75–81. DOI: https://doi.org/10.15276/opu.2.44.2014.15.
29. Foshch T., Pelykh S. Improved models and method of power change of NPP unit with VVER-1000. *Automation of Technological and Business–Processes*. 2017. 9 (1). 56–66. DOI: http://dx.doi.org/10.15673/atbp.v9i1.505.
30. Odrekhovska Y. O., Zhou H., Pelykh S. N. Optimization of switching of static power control programs for NPPs with VVER-1000 in transient operating modes. *XXIV Annual Scientific Conference of the Institute for Nuclear Research of the National Academy of Sciences of Ukraine: abstracts*. Kiev: Ukraine. 2017. p. 123.

References

1. Jiang, Q., Liu, Y., Zeng, W., & Yu, T. (2020). Study on switching control of PWR core power with a fuzzy multimodel. *Annals of Nuclear Energy*, *145*, 107611. DOI: https://doi.org/10.1016/j.anucene.2020.107611.
2. Wang, G., Wu, J., Zeng, B., Xu, Z., Wu, W., & Ma, X. (2016). State-space model predictive control method for core power control in pressurized water reactor nuclear power stations”. *Nuclear Engineering and Technology*, 49, 134–140. DOI: http://dx.doi.org/10.1016/j.net.2016.07.008.
3. Aver’yanova, S. P., Vokhmyanina, N. S., Zlobin, D. A., Filimonov, P. E., Kuznetsov, V. I., & Lagovskii, V. B. (2017). Offset-cardinality phase diagram method of controlling reactor power. *Atomic Energy*, 121, 155–160. DOI: https://doi.org/10.1007/s10512-017-0176-5.

4. Foshch, T., Machado, J., Portela, F., Maksimov, M., & Maksimova, O. (2017). Comparison of Two Control Programs of the VVER-1000 Nuclear Power Unit Using Regression Data Mining Models. *Nuclear and Radiation Safety*, 3 (75), 11–17. DOI: https://doi.org/10.32918/nrs.2017.3(75).02.
5. Foshch, T., Maksimov, M., Pelykh, S., & Maksimova, O. (2018). Models and methods for automated control of power change at VVER-1000 nuclear power unit". *Nuclear and Radiation Safety*, 1 (77), 24–30. URL: http://dspace.opu.ua/jspui/bitstream/123456789/10580/1/ydpb_2018_1_6.pdf.
6. Brunetkin, O., Beglov, K., Maksymov, M., Baskakov, V., Vataman, V. & Kryvda, V. (2022). Designing an automated control system for changing NPU energy release compensating for arising internal disturbing factors based on their approximation model. *Eastern-European Journal of Enterprise Technologies*, 3 (2 (117)), 63–75. DOI: https://doi.org/10.15587/1729-4061.2022.258394.
7. Kokol, E. A. (2015). Structural optimization of static power control programs for nuclear power plants with WWER-1000. *Proceedings of Odessa Polytechnic University*, 3 (47), 41–46. DOI: https://doi.org/10.15276/opu.3.47.2015.07.
8. Ding, Y., Wang, Q., Tian, Z., Lyu, Y., Li, F., Yan, Z., & Xia, X. (2023). A graph-theory-based dynamic programming planning method for distributed energy system planning: Campus area as a case study. *Applied Energy*, 329, 120258. DOI: https://doi.org/10.1016/j.apenergy.2022.120258.
9. Fathy, E., & Hassanien, A. E. (2022). Fuzzy harmonic mean technique for solving fully fuzzy multilevel multiobjective linear programming problems. *Alexandria Engineering Journal*, 61 (10), 8189–8205. DOI: https://doi.org/10.1016/j.aej.2022.01.021.
10. Plakhotnuk, A.A., Kokol, E. A., & Maksimov, M. V. (2015). Improved PCS software switch control power generating. *Automation Technological and Business-Processes*, *7* (3), 26–33. DOI: https://doi.org/10.15673/2312-3125.4/2015.50429.
11. Pelykh, S. N., Maksimov, M. V., & Parks, G. T. (2013). A method for VVER-1000 fuel rearrangement optimization taking into account both fuel cladding durability and burnup. *Nuclear Engineering and Design*, *257*, 53–60. DOI: https://doi.org/10.1016/j.nucengdes.2012.12.022.
12. Maksymov, M., Alyokhina, S., & Brunetkin, O. (2021). *Thermal and reliability criteria for nuclear fuel safety*. River Publishers (Verlag). 260 p. e-ISBN: 9788770224000.
13. Maksimov, M.V., Tsiselskaya, T.A., & Kokol, E.A. (2015). The Method of Control of Nuclear Power Plant with VVER-1000 Reactor in Maneuverable Mode. *Journal of Automation and Information Sciences*, *47* (6), 17–32. DOI: https://doi.org/10.1615/JAutomatInfScien.v47.i6.20.
14. Li, G., Wang, X., Liang, B., Li, X., Zhang, B., & Zou, Y. (2016). Modeling and control of nuclear reactor cores for electricity generation. A review of advanced technologies. *Renewable and Sustainable Energy Reviews*, *60*, 116–128. DOI: https://doi.org/10.1016/j.rser.2016.01.116.
15. Maksimov, M. V., Pelykh, S. N., & Gontar, R. L. (2012). Principles of controlling fuel-element cladding lifetime in variable VVER-1000 loading regimes. *Atomic Energy*, *112* (4), 241–249. DOI: https://doi.org/10.1007/s10512-012-9552-3.
16. Foshch, T., Portela, F., Machado, J., & Maksimov, M. (2016). Regression Models of the Nuclear Power Unit VVER-1000 Using Data Mining Techniques. *Procedia Computer Science*, *100*, 253–262. DOI: https://doi.org/10.1016/j.procs.2016.09.151.
17. Yan, X., Wang, P., Qing, J., Wu, S., & Zhao, F. (2020). Robust power control design for a small pressurized water reactor using an H infinity mixed sensitivity method. *Nuclear Engineering and Technology*, 52 (7), 1443–1451. DOI: http://doi.org/10.1016/j.net.2019.12.031.
18. Filimonov, P. E., Aver'yanova, S. P., & Filimonova, M. P. (1998). Control of control-rod groups in the maneuvering regime of VVÉR-1000 operation. *Atomic Energy*, *84* (5), 383–387. DOI: https://doi.org/10.1007/BF02414876.
19. Kokol, E.A. (2015). Optimal power control of VVER-1000 due to targeted selection of the control program. *Automation 2015: XXII International Conference on Automatic Control.* Odessa: Ukraine. 2015. p. 119–120.
20. Plakhotnuk, A. A., Kokol, E. A., & Maksimov, M. V. (2015). Simulation of changes in the structure of technical means of automation at the VVER-1000 while maneuvering. *Automation Technological and Business-Processes*, *7* (4), 64–71. DOI: https://doi.org/10.15673/2312-3125.24/2015.56334.
21. Ningbo, L., & Xueyao, S. (2017). Research on the control of containment pressure of Gen II+ NPP under severe accident condition. *Energy Procedia*, *117*, 220–224. DOI: https://doi.org/10.1016/j.egypro.2017.08.098.
22. Zhou, H., Pelykh, S. N., Odrekhovska, I. O., & Maksymova, O. B. (2018). Optimization of power control program switching for a WWER-1000 under transient operating conditions. *Problems of Atomic Science and Technology, 1* (113), 218–221. URL: http://dspace.nbuv.gov.ua/bitstream/handle/123456789/137360/36-Zhou.pdf?sequence=1.
23. Pelykh, S. N., Maksimov, M. V., & Ryabchikov, S. D. (2016). The prediction problems of VVER fuel element cladding failure theory. *Nuclear Engineering and Design*, *302*, 46–55. DOI: https://doi.org/10.1016/j.nucengdes.2016.04.005.
24. Pelykh, S. N., Maksimov, M. V., & Nikolsky, M. V. (2014). A method for minimization of cladding failure parameter accumulation probability in VVER fuel elements. *Problems of Atomic Science and*

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Technology, *92* (4), 108–116. URL: https://vant.kipt.kharkov.ua/ARTICLE/VANT_2014_4/article_2014_4_108.pdf.

25. Beglov, K. V., Odrekhovska, Y. O., Petik, T. V., & Vataman, V. V. (2023). A method for searching the best static program for nuclear power unit control in the event of perturbations of different nature". *Herald of Advanced Information Technology*, *6* (2), 139–151. DOI: https://doi.org/10.15276/hait.06.2023.9.
26. Pelykh, S. N., & Maksimov, M. V. (2013). Theory of VVER-1000 fuel rearrangement optimization taking into account both fuel cladding durability and burnup. *Problems of Atomic Science and Technology*, *44* (2) 84, 50–54. URL: http://dspace.nbuv.gov.ua/bitstream/handle/123456789/111678/08-Pelykh.pdf?sequence=1.
27. Pelykh, S. N., & Maksimov, M. V. (2013). The method of fuel rearrangement control considering fuel element cladding damage and burnup. *Problems of Atomic Science and Technology*, *87* (5), 84–90. URL: https://vant.kipt.kharkov.ua/ARTICLE/VANT_2013_5/article_2013_5_84a.pdf.
28. Maksimov, M. V., Kanazirskyi, N. F., & Kokol, E. A. Control of the axial offset in a nuclear reactor at power maneuvering. *Proceedings of Odessa Polytechnic University, 2* (4), 75–81. DOI: https://doi.org/10.15276/opu.2.44.2014.15.
29. Foshch, T., & Pelykh, S. (2017). Improved models and method of power change of NPP unit with VVER-1000. *Automation of Technological and Business–Processes*, *9* (1): 56–66. DOI: http://dx.doi.org/10.15673/atbp.v9i1.505.
30. Odrekhovska, Y. O., Zhou, H., & Pelykh, S. N. (2017). Optimization of switching of static power control programs for NPPs with VVER-1000 in transient operating modes. *XXIV Annual Scientific Conference of the Institute for Nuclear Research of the National Academy of Sciences of Ukraine: abstracts*. Kiev: Ukraine. p. 123.

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