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DESIGNING AN AUTOMATED CONTROL SYSTEM FOR CHANGING NPU ENERGY RELEASE COMPENSATING FOR ARISING INTERNAL DISTURBING FACTORS BASED ON THEIR APPROXIMATION MODEL

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The necessity of keeping in working condition the sources of traditional electricity generation for dispatching the supply of electricity from the capacities of "green" electricity generation has been realized. Potentially, one of these sources may be nuclear power plants. An obstacle to their use as a participant in dispatching is the impossibility of maneuvering with power in a wide range. One of the reasons for limiting the range is the complexity of automatic compensation for the reactivity of the xenon oscillation reactor.

Existing physical and mathematical models for calculating the parameters of processes in the reactor due to changes in its power because of its complexity cannot be used in operational automatic control systems. The task of constructing an approximation linear model of processes in the reactor in the form of a transfer function is set.

To build an approximation model, the inverse problem is solved. The desired model is based on the condition of coincidence at some time interval of the results of solving it with the results of a detailed physical-mathematical model. To this end, a number of sequential actions are performed, including approximation of the results of the expanded physical-mathematical model using a series, the application of the Laplace transformation to this series, as well as Pade approximation obtained in the space of the images of the series.

The method of control was proposed and an automatic control system (ACS) for energy production of nuclear power plant has been synthesized. To this end, the management system was integrated with the approximation model of the active zone, which provided the possibility for adjusting the quantitative degree of stability of the active zone.

ACS consists of three control circuits. Such a structure has made it possible to compensate for the xenon oscillations that occur.

Additionally, ACS reduces the movement of adjusting rods in the active zone, which reduces local power surges in nuclear fuel and leads to an increase in its durability

Keywords: power maneuvering, Pade approximation, approximation model, xenon oscillations, reactivity

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1. Introduction

The philosophy of industry decarbonization in industrialized countries and, in particular, modern energy engineering has led to the ever-increasing use of renewable energy sources to cover the demand for energy products of energy systems. In their capacity, mainly wind and solar generation devices are used. With the growth of their share, the problem of dispatching energy flows is exacerbated. One of the ways to solve it in the future is planned to use hydrogen energy. This problem can be solved through the use of various kinds of storage devices and the efficient use of other available sources related to "green energy" (for example, hydroelectric power plants (HPPs)).

One of the locomotives of the process of decarbonization of industry is the countries of the European Union. They have a high percentage of renewable electricity. Nevertheless, recently there has been an increase (by several times) in their imports of electricity from third countries. This may indicate a shortage of capacity to dispatch electricity flows within the framework of "green" energy with a large total installed capacity of electricity generating devices. Against this background, the European Commission has decided to include nuclear energy and natural gas in the "EU Green Taxonomy". This is a classification of environmentally sound activities for investors. With all the limitations imposed by this solution, the possibilities of stabilizing the operation of the power system are expanding.

Of particular interest is the solution on nuclear energy. Environmentally sustainable are stations whose projects will be approved before 2045, if there are sufficient volumes of radioactive waste storage facilities. The timing of construction and operation indicates the possibility of their work even after the end of the XXI century. Throughout this period, they will have an impact on the generation and stabilization of electricity supplies.

A distinctive feature of "nuclear" generation is the duration of the process of controlled change in reactor power. This makes it difficult or even impossible to use the electricity of nuclear power plants to solve dispatching problems. Significant installed capacities of nuclear power plants, an increase in the long-term attractiveness in this industry for investors give rise to interest in using the capacities of nuclear reactors to dispatch electricity flows. Thus, the development of methods for operational control of reactor power changes is relevant.

2. Literature review and problem statement

Modern automatic control systems (ACS) of the power unit in a nuclear power plant (NPP) are built on the classical principle of "deviation" regulation. That is, they are designed to compensate for the imbalance of electricity arising from the consumer. The consequence of this and the technological features of the operation of a nuclear power unit (NPU) is the operation of the power unit only in stationary mode at a power level close to the maximum. The ACSs used make it possible to compensate for the change in power plus minus 1.5 % of the current power value [1, 2]. However, they do not have the ability to control the transfer of NPUs from one stationary state to another stationary state corresponding to different power values. Such control is carried out by the operator of the installation in remote mode by influencing the relevant regulatory authorities when receiving current information from the sensors [3].

At the same time, the speed of the control computing complex is sufficient to perform the tasks of measuring technological parameters and developing control actions. The sensors of the intra-reactor control system are surveyed at least once in 0.2 seconds, the delay in control action does not exceed 2 seconds [3].

In [4], it is noted that all the effects carried out by the operator on the control object are their duration and depth are pre-calculated values. However, the calculation in real time of the model of the transition process from one level of the state of power to another is impossible to carry out because of the computational performance of computer systems, which for existing systems lags behind by more than three orders of magnitude.

Thus, the main limitation is the time to calculate the neutron-physical parameters of the reactor core.

Regardless of the method of power regulation in the core, there are always fluctuations caused by the spatial redistribution of the neutron flux. In [2], it is shown that changes in reactivity due to xenon transients are determinative in the degree of impact on safety and are considered separately from samarium and independently from the process of fuel burnout. Within the framework of the theory of automatic control, modeling of the system and the control object is based on differential models. There are various models that describe intra-reactor processes. They can be described, for example, using ordinary differential equations [6], or using models based on differential equations of fractional order [7]. The level of development of computer technology, which is allowed to control a nuclear reactor, does not allow for real-time calculations based on such models. As a result, it is not possible to organize automatic control of the change in the capacity of the core during the transition from one state to another.

It should be noted that in some cases there are options for possible automatic control systems based on simplified models [8]. However, traditionally, in operational practice for each campaign, operators are provided with nomograms, which represent the magnitudes of the amplitudes of oscillations of changing reactivity. Nomograms about the properties of the core are generated on the basis of the results of a preliminary solution by numerical methods of a complete system of nonlinear differential equations of the model of intra-reactor processes. A contradictory situation arises. There is a complete system of differential equations and the results of its solution (in particular, displayed in the form of nomograms). At the same time, an attempt is made to create an automatic control system based on a simplified model or control takes place manually based on pre-calculated nomograms.

One possible way to overcome this contradiction may be to develop an approximating model of the process. The approximation and simplified models have a difference. The approximation model is built on the basis of the results of calculations on the full model [9]. In this case, the information is not discarded. The model is built on the basis of formal transformations in such a way that the results of the calculation over a certain period of time coincide or are close to the results of the calculation when using the full model. The order of the equation can be chosen. The limiting property of this method in the form of [9] is the use of a single reaction of the system to external influences. In other words, a model constructed in this way can approximate a process that depends on a change in only one argument, such as time. At the same time, nomograms of changes in the amplitude of reactivity are built depending on two parameters: time and the degree of change (jump) of the load. Thus, in addition to the problem of constructing an approximation differential model, the problem of two-dimensional approximation arises – a special case of multivariate approximation.

The approximation of multidimensional data is a complex task. From the point of view of fundamental mathematics, it is correlated with the 13th problem of D. Hilbert [10] of the 23 problems of mathematics published by him in the early twentieth century. In the middle of the XX century, this problem was solved in works [11, 12]. However, the problem itself and, accordingly, its solution is built from the point of view of proving the possibility of realizing the proposition: "Is it possible to solve the general equation of the seventh power with the help of functions that depend only on two variables?". The mathematicians' answer is, "Yeah". But the implementation path for solving diverse problems is not indicated. The practical implementation of such a possibility in solving approximation problems in each case is built individually, depends on the available data and the goals set. The main task is to find a function or functions of two or three variables, with the help of which the function of more variables is approximated [13].

After work [11], a "stronger" position was proved. The application of its results to the problems of approximating the functions of several variables makes it possible to talk about the possibility of their approximation using the superposition of the functions of one variable and the addition operation [14]. Ha, the proof is given from the standpoint of the possibility of such an operation but the path is not presented.

Considering the possibility of approximating a multidimensional function using the superposition of functions of one variable, we can also talk about representing the solution to systems of differential equations of several variables using such superpositions. The practical implementation of both the possibility of such approximation and the possibility of numerical solution of differential equations is based on the use of neural networks [15]. The possibility of solving a set of two problems within the framework of one tool smells attractive. However, while confirming in [15] such a potential possibility, difficulties are also noted that exclude the possibility of practical implementation in technical areas requiring predictable quality of the solution.

Our review [1–15] reveals the need to find a method for displaying the results of solving systems of nonlinear differential equations of several arguments in the form of an approximation model. It can be used to modernize the computational and control subsystem, which is part of the intra-reactor control system. The task of this subsystem is to compensate for changes in reactivity caused by xenon transient processes during the transition of the reactor core from one level of stationary power to another.

3. The aim and objectives of the study

The purpose of this study is to devise a method for two-dimensional identification of a nonlinear control object, the properties of which change over time and depend on the operating conditions of technological equipment. This will make it possible to set up an automated control system for the operation of NPU (nuclear power unit) with VVER-1000 in the variable part of the electrical load schedule and provide partial regulation of the power system. Such an automated control capability will ensure the maneuverable properties of the installation while maintaining a balance between the economic feasibility of adjusting the load of the power system and its safe operation under a cyclic mode with minimal operator participation.

To accomplish the aim, the following tasks have been set:

– to implement an approximation model and a transfer function that correspond to the results of solving a system of nonlinear differential equations;

– to devise a control method and, on its basis, synthesize ACS for changing the energy release by NPU to ensure a quantitative measure of the stability of the core.

4. The study materials and methods

The most complex nature of the fluctuation of reactivity is at the end of the company with an increase or decrease in the load. There are developed physical-mathematical simulation models of nuclear power units (NPUs) as an object of control. Let's call them control objectphysical. These models cannot be integrated into existing ACS (automated control system) of existing NPUs for real-time operation. The reason is their nonlinearity, the lack of computing tools of the required performance, allowed to work in NPU ACS. Control is carried out by the operator in manual mode based on nomograms. They are built on the basis of the results of numerical solutions made in advance based on the corresponding control objectphysical models.

Fig. 1, 2 show the results obtained for the transient processes of the VVER-1000 reactor at a rated thermal power of 3000 MW. The power varies in stages with a stage value of 25 %, 50 %, 75 %, 100 % (750 MW, 1500 MW, 2250 MW, 3000 MW). The results of a numerical calculation of the change in reactivity are given both when increasing the power from a certain value to the rated power (Fig. 1), and when decreasing the power from the rated value to the specified one (Fig. 2).

Linear differential models are used to configure the regulators that are part of the ACS. A method for constructing such a model is proposed, but not based on a developed control objectphysical model (for example, by simplifying it) but based on the results of numerical calculations based on it (Fig. 1, 2). Let's call this model control objectapproximation. It is built in such a way that the results of calculations based on both models coincide best on a certain range of changes in the original data. It is assumed that, in this case, the control objectapproximation model, being linear, adequately displays the dependences necessary to configure the ACS regulators. In our case, the results of calculations based on control objectphysical are two-dimensional: they change depending on the time and on the magnitude of the stage of change in reactor power. The model being developed should take this into consideration.

The proposed model is built by performing a series of steps. All necessary computational procedures are simple and can be performed in real time on computing equipment approved for operation at NPU.

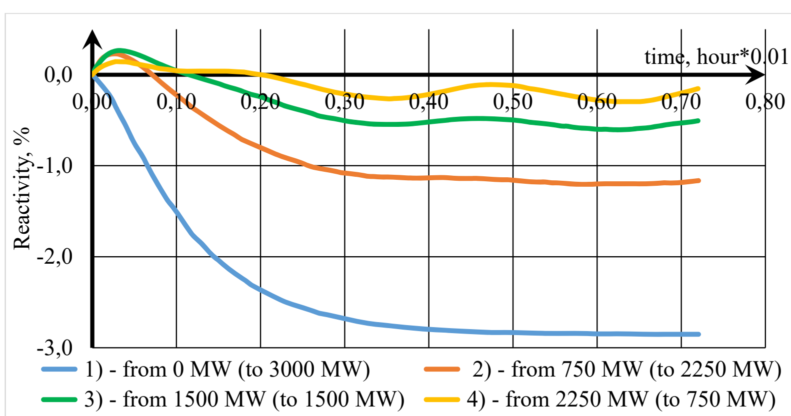

Fig. 1. Oscillation of reactivity with a stepwise increase in the load from the specified value to the rated thermal power of the reactor (up to 3000 MW)

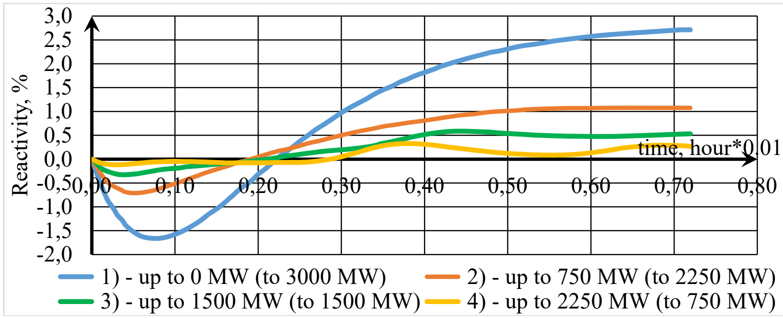

Fig. 2. Oscillation of reactivity with stepwise load reduction from the rated thermal power of the reactor (3000 MW) to the specified value

In the first step, each curve (Fig. 1, 2) is fitted to a single argument function. Potentially, the type of function may be different. However, due to the peculiarities of the operations of the third stage, a polynomial was chosen as an approximating function. All plots at the initial time $t=0$ have a reactivity value of $\rho(0)=0$. Thus, approximation polynomials do not have a free term:

$$f_j(t)=\sum_{i=1}^{n} c_{ij}t^i, \quad (1)$$

where n is the number of terms of the polynomial.

In the second step, the transfer functions of the approximation models for each plot (process) are determined (Fig. 1, 2). Apply to the approximating polynomial (1) the Laplace transform:

$$L\{f_j(t)\}=L\left\{\sum_{i=1}^{n} c_{ij}t^i\right\}=$$
$$=\sum_{i=1}^{n} c_{ij}i!\left(\frac{1}{p}\right)^{i+1}=$$
$$=\left(\frac{1}{p}\right)^2\cdot\left[\sum_{i=1}^{n} c_{ij}i!\left(\frac{1}{p}\right)^{i-1}\right]. \quad (2)$$

Here, i is an index that marks the terms of an approximating polynomial for one plot (process), j is an index that marks different plots. Expression (2), similar to (1), approximates the solution, but in the image space.

Consider the expression in square brackets from (2). It is truncated next to the free term. Apply to this expression the Padé approximation $[L/M]$ with respect to argument $1/p$. After a series of transformations and taking into consideration the second multiplier of (2), we obtain:

$$\left(\frac{1}{p}\right)^2\cdot\left[\sum_{i=1}^{n} c_{ij}i!\left(\frac{1}{p}\right)^{i-1}\right] \Rightarrow \left(\frac{1}{p}\right)^2\cdot[L/M]=$$
$$=\left(\frac{1}{p}\right)^2\cdot\left[\frac{g_L\left(\frac{1}{p}\right)^L+g_{L-1}\left(\frac{1}{p}\right)^{L-1}+\ldots+g_1\left(\frac{1}{p}\right)+g_0}{h_M\left(\frac{1}{p}\right)^M+h_{M-1}\left(\frac{1}{p}\right)^{M-1}+\ldots h_1\left(\frac{1}{p}\right)+1}\right] \quad (3)$$

or, after some transformations:

$$\left(\frac{1}{p}\right)^2\cdot[L/M]=\left[\frac{\left(\frac{g_L}{h_M}\right)\cdot p^{M-L}+\left(\frac{g_{L-1}}{h_M}\right)\cdot p^{M-L+1}+\ldots+\left(\frac{g_1}{h_M}\right)\cdot p^{M-1}+\left(\frac{g_0}{h_M}\right)\cdot p^{M}}{1+\left(\frac{h_{M-1}}{h_M}\right)\cdot p+\ldots\left(\frac{h_1}{h_M}\right)\cdot p^{M-1}+\left(\frac{1}{h_M}\right)\cdot p^{M}}\cdot\frac{1}{p}\cdot\frac{1}{\Delta P_j}\right]\cdot\frac{\Delta P_j}{p}. \quad (4)$$

Here, ΔP_j is the magnitude of the stepwise change in the reactor power.

Expression (4) is a representation of a numerical solution in image space, recorded using the Pade approximation. Within the framework of TAU, it can be interpreted as a reflection of the behavior of the object under a stepped action with an image in the form of $\Delta P_j/p$ (the second multiplier). In this case, the expression in square brackets can be thought of as a transfer function of the object's approximation model.

In the third step, an approximation of the two-dimensional data is performed (Fig. 1, 2). One of the arguments is time. This argument is used when constructing each curve (Fig. 1, 2) as (1) or (4). The second argument is the magnitude of the stepwise change in power ΔP. Approximation is performed on the basis of the power series according to the method of least squares (LSM). As an argument, the value of the stepwise change in power ΔP is used. The corresponding value of the function is not a fixed numeric value but the corresponding fixed form (1) or (4).

In our case, we have four values (j=1...4) of the stepwise power change (ΔP). In the absence of a change in power, the stationary mode of operation of the reactor, the reactivity r is zero. Thus, at the second coordinate (ΔP), the initial data can be approximated using a fourth-power series without a free term. The matrix equation displaying the described situation is:

$$\begin{array}{ccc} \mathbf{K} & & \mathbf{X} \\ \begin{bmatrix} \sum_{j=1}^{4}(\Delta P_j)^8 & \sum_{j=1}^{4}(\Delta P_j)^7 & \sum_{j=1}^{4}(\Delta P_j)^6 & \sum_{j=1}^{4}(\Delta P_j)^5 \\ \sum_{j=1}^{4}(\Delta P_j)^7 & \sum_{j=1}^{4}(\Delta P_j)^6 & \sum_{j=1}^{4}(\Delta P_j)^5 & \sum_{j=1}^{4}(\Delta P_j)^4 \\ \sum_{j=1}^{4}(\Delta P_j)^6 & \sum_{j=1}^{4}(\Delta P_j)^5 & \sum_{j=1}^{4}(\Delta P_j)^4 & \sum_{j=1}^{4}(\Delta P_j)^3 \\ \sum_{j=1}^{4}(\Delta P_j)^5 & \sum_{j=1}^{4}(\Delta P_j)^4 & \sum_{j=1}^{4}(\Delta P_j)^3 & \sum_{j=1}^{4}(\Delta P_j)^2 \end{bmatrix} & \times & \begin{bmatrix} a \\ b \\ c \\ d \end{bmatrix} = \end{array}$$

$$\begin{array}{ccc} & \mathbf{M} & & \mathbf{F} \\ = & \begin{bmatrix} (\Delta P_1)^4 & (\Delta P_2)^4 & (\Delta P_3)^4 & (\Delta P_4)^4 \\ (\Delta P_1)^3 & (\Delta P_2)^3 & (\Delta P_3)^3 & (\Delta P_4)^3 \\ (\Delta P_1)^2 & (\Delta P_2)^2 & (\Delta P_3)^2 & (\Delta P_4)^2 \\ (\Delta P_1)^1 & (\Delta P_2)^1 & (\Delta P_3)^1 & (\Delta P_4)^1 \end{bmatrix} & \times & \begin{bmatrix} f_1(t) \\ f_2(t) \\ f_3(t) \\ f_4(t) \end{bmatrix}, \end{array} \quad (5)$$

its solution:

$$\begin{bmatrix} a \\ b \\ c \\ d \end{bmatrix} = \left[\mathbf{K}^{-1} \cdot \mathbf{M} \right] \cdot \mathbf{F}. \qquad (6)$$

Here, $f_i(t)$ are functions of the form (1) or (4) that approximate each of the curves separately; Fig. 1 or Fig. 2; *a, b, c, d* are the coefficients of the power series describing the entire set of initial data (Fig. 1 or Fig. 2).

The resulting series (6) with $f_i(t)$ functions in the form (4) displaying two-dimensional source data can be considered their approximation model control objectapproximation.

The sequence of steps that must be followed to implement the method of constructing a linear differential approximation model control objectapproximation is schematically displayed in Fig. 3.

The approximation linear nature of the developed control objectapproximation model assumes that it cannot be used over the entire time interval necessary to compensate for changes in reactivity. At the same time, it makes it possible to solve several problems. Thus, its application makes it possible to use ACS regulators at a certain interval of time. In turn, it becomes possible during this time interval to calculate new parameters for the control objectapproximation model (without changing its structure) and reconfigure the regulators.

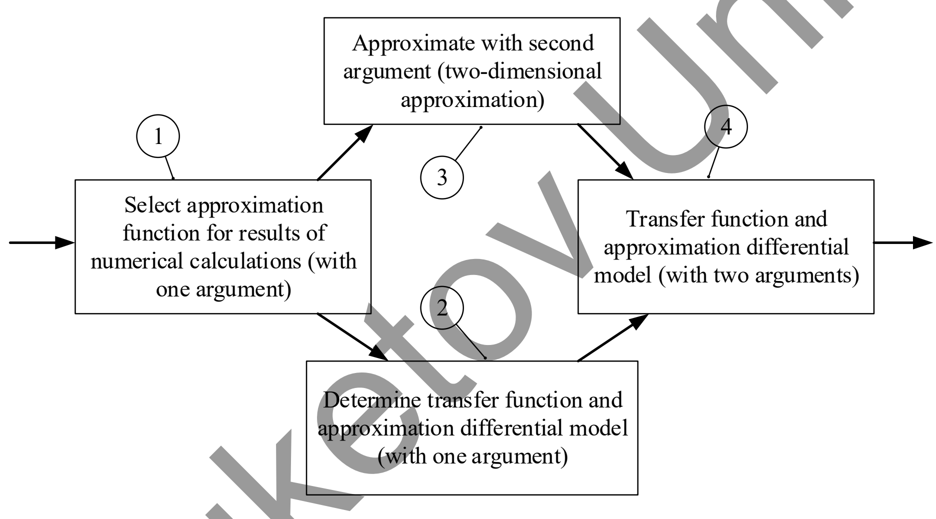

Fig. 3. Control objectapproximation build sequence

5. Results of studying an automated energy release change control system with compensation of internal disturbances

5. 1. Approximation model and transfer function of intra-reactor processes

In accordance with the first step (Fig. 3), approximation polynomials for plots in Fig. 1 were obtained in the form (1). The power of polynomial was chosen based on the need to ensure engineering accuracy (relative error <5 %). The highest, tenth power, was required for plot 4. In this case, the relative approximation error does not exceed 3.6 %. For uniformity of calculations, the remaining plots are approximated by polynomials of the same power. The coefficients of polynomials in the form (1) for the corresponding plots (Fig. 1) are given in Table 1. When approximating, the value of the argument in comparison with the real time value is reduced by two orders of magnitude and corresponds to the value of the argument on the plots (Fig. 1).

Table 1

Coefficients of approximation polynomials for changing reactivity with increasing reactor load

Coefficients of polynomials	Plot No. (Fig. 4)			
	1	2	3	4
c_{10}	–5.696E+04	–5.137E+04	–4.269E+04	7.805E+04
c_9	2.254E+05	2.254E+05	2.024E+05	–2.391E+05
c_8	–3.867E+05	–4.240E+05	–4.088E+05	2.748E+05
c_7	3.773E+05	4.473E+05	4.585E+05	–1.255E+05
c_6	–2.307E+05	–2.909E+05	–3.123E+05	–1.108E+04
c_5	9.169E+04	1.210E+05	1.330E+05	3.585E+04
c_4	–2.365E+04	–3.240E+04	–3.524E+04	–1.536E+04
c_3	3.757E+03	5.472E+03	5.643E+03	3.010E+03
c_2	–2.921E+02	–5.381E+02	–5.115E+02	–2.858E+02
c_1	–7.093E+00	1.963E+01	1.982E+01	1.095E+01

In accordance with the procedure of the third step (Fig. 3), the coefficients (6) of the approximation dependence are obtained. It makes it possible to determine the reactivity at an arbitrary point in time when the load changes from an arbitrary value to the rated value of the thermal power of the reactor. When calculating the coefficients (6), the magnitude of the stepwise change in power was reduced by three orders of magnitude. For this reason, (2) should be used in the form of:

$$\begin{aligned} \rho_j = {} & a \cdot \left(\Delta P_j \cdot 10^{-3} \right)^4 + \\ & + b \cdot \left(\Delta P_j \cdot 10^{-3} \right)^3 + \\ & + c \cdot \left(\Delta P_j \cdot 10^{-3} \right)^2 + \\ & + d \cdot \left(\Delta P_j \cdot 10^{-3} \right), \end{aligned} \qquad (7)$$

coefficients *a, b, c, d* take the form:

$$\begin{aligned} a = {} & -0.5267 \cdot f_4 + 0.7901 \cdot f_3 - \\ & -0.5267 \cdot f_2 + 0.1317 \cdot f_1, \end{aligned}$$

$$b = +3.5556 \cdot f_4 - 4.7407 \cdot f_3 + 2.7654 \cdot f_2 - 0.5930 \cdot f_1,$$

$$c = -7.7037 \cdot f_4 + 8.4444 \cdot f_3 - 4.1481 \cdot f_2 + 0.8148 \cdot f_1,$$

$$d = +5.3333 \cdot f_4 - 4.0000 \cdot f_3 + 1.7778 \cdot f_2 - 0.3330 \cdot f_1. \qquad (8)$$

Fig. 4 shows plots obtained using a single dependence (7).

Plots in Fig. 4, shown in black, correspond to the change in reactor load used in the construction of plots in Fig. 1. Because expression (7) is an interpolation dependence, the corresponding plots in Fig. 1, 4 coincide. Plots in Fig. 4, depicted by dotted lines, correspond to intermediate loads.

In accordance with the procedure of the second step (Fig. 3), images of the results of the solution of the original model (Fig. 1) were obtained from (4) in the form of Padé approximation. As the initial truncated series, (1) with coefficients from Table 1 are used. Approximation expressions in the form of a rational fraction take the form:

$$L\left\{\sum_{i=1}^{n} c_{ij} t^{i}\right\}= \\ =\left(\frac{1}{p}\right)^{2} \cdot \frac{b_4 p+b_3 p^2+b_2 p^3+b_1 p^4+b_0 p^5}{a_5+a_4 p+a_3 p^2+a_2 p^3+a_1 p^4+p^5}. \qquad (9)$$

The coefficients b_i, a_i in (9) for the corresponding plots (Table 1) are given in Table 2. Also in this table, the roots are given for the relevant solutions.

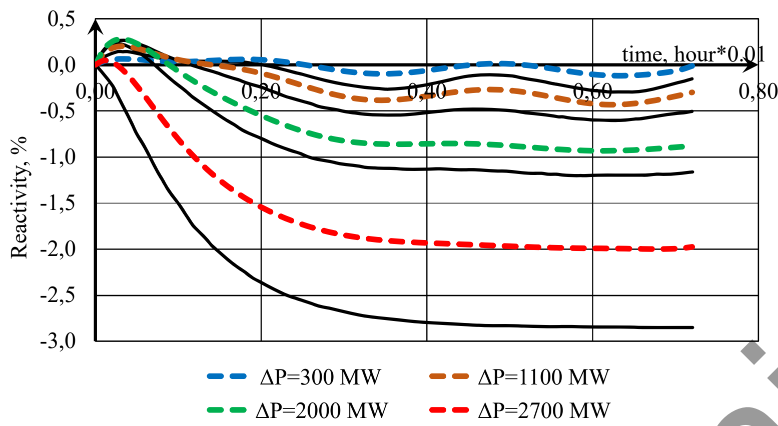

Fig. 4. Oscillation of reactivity during stepwise increase in load to the rated thermal power of the reactor (up to 3000 MW)

Table 2

The numbers of coefficients of Padé approximation and roots of its denominator for the case of a reactor load increase

No.	plot 1		plot 2		plot 3		plot 4	
	a_i	b_i	a_i	b_i	a_i	b_i	a_i	b_i
5	3.65e+6	–	1.36e+6	–	8.77e+5	–	4.12e+5	–
4	6.66e+5	–9.94e+6	2.69e+5	–1.30e+6	1.83e+5	–1.20e+5	9.86e+4	2.71e+5
3	5.70e+4	–7.63e+5	2.60e+4	–8.08e+4	1.90e+4	–5.05e+4	1.07e+4	–5.96e+4
2	2.85e+3	–4.63e+4	1.53e+3	2.70e+3	1.23e+3	7.62e+3	7.72e+2	6.47e+3
1	8.32e+1	–1.17e+3	5.60e+1	2.26e+1	4.94e+1	–4.41e+1	3.51e+1	–1.88e+2
0	–	–7.09e+0	–	1.96e+1	–	1.98e+1	–	1.09e+1
Root								
p_1	–35.444		–17.059		–15.343		–9.065	
p_2	–15.669–6.659i		–14.017–8.840i		–12.553–8.842i		–10.378–9.324i	
p_3	–15.669+6.659i		–14.017+8.840i		–12.553+8.842i		–10.378+9.324i	
p_4	–8.209–16.963i		–5.443–16.143i		–4.473–14.918		–2.618–15.052i	
p_5	–8.209+16.963i		–5.443+16.143i		–4.473+14.918		–2.618+15.052i	

To assess the adequacy of the approximation of the image of the solution in the form (9), the originals from these expressions were found and compared for some functions with the solutions in Fig. 1:

$$f_1(t)=(1.50004+3.16417i)\cdot e^{(-15.6692-6.65936\,i)t}+ \\ +(1.50004-3.16417i)\cdot e^{(-15.6692+6.65936\,i)t}- \\ -(0.136931+0.133379i)\cdot e^{(-8.2092-16.9625\,i)t}- \\ -(0.136931-0.133379i)\cdot e^{(-8.2092+16.9625\,i)t}- \\ -0.00139788\cdot e^{-35.4444t}-2.72482, \qquad (10)$$

for plot 1, Fig. 1;

$$f_4(t)=(3.77179+0.820439i)\cdot e^{(-10.3779-9.3241\,i)t}+ \\ +(3.77179-0.820439i)\cdot e^{(-10.3779+9.3241\,i)t}- \\ -(0.486884-0.194877i)\cdot e^{(-2.61771-15.0517\,i)t}- \\ -(0.486884+0.194877i)\cdot e^{(-2.61771+15.0517\,i)t}- \\ -7.22735\cdot e^{-9.06518t}+0.657543, \qquad (11)$$

for plot 4, Fig. 1. The compared plots are shown in Fig. 5.

Fig. 5 shows plots demonstrating the results of the approximation for two boundary data series available: an increase in load by ΔP=3000 MW (from about 0 to 3000 MW) and by ΔP=750 MW (from 2250 to 3000 MW). They cover the rest of the data. The comparison results for other data are similar. It follows from Fig. 5 that up to 30 hours (0.30 on the plot), the representation of the results of the solution to the original model using polynomial approximation and the Pade approximation based on it are practically indistinguishable and, therefore, interchangeable.

In accordance with the first step (Fig. 3), approximation polynomials for Fig. 2 plots were obtained in the form (1). The coefficients of polynomials in the form (1) for the corresponding plots (Fig. 2) are given in Table 3. During approximation, as in the case of increasing the reactor load, the value of the argument compared to the real time value is reduced by two orders of magnitude and corresponds to the value of the argument on the plots (Fig. 2).

In accordance with the procedure of the second stage (Fig. 3), the coefficients of the approximation dependence are obtained from ratio (6). When calculating the coefficients (6), the magnitude of the stepwise change in power, as in the case of increasing the reactor load, was reduced by three orders of magnitude.

The coefficients *a, b, c, d* take a form similar to (8):

$$a=-0.5267\cdot f_4+0.7901\cdot f_3- \\ -0.5267\cdot f_2+0.1317\cdot f_1,$$

$$b=+3.5556\cdot f_4-4.7407\cdot f_3+ \\ +2.7654\cdot f_2-0.5930\cdot f_1,$$

$$c=-7.7037\cdot f_4+8.4444\cdot f_3- \\ -4.1481\cdot f_2+0.8148\cdot f_1,$$

$$d=+5.3333\cdot f_4-4.0000\cdot f_3+1.7778\cdot f_2-0.3330\cdot f_1. \qquad (12)$$

The difference is in the change in the values of $f_1 ... f_4$, calculated using the coefficients from Table 3.

In accordance with the procedure of the third step (Fig. 3), representations of the results of the solution of the original model (Fig. 2) were obtained from (4) in the form of Padé approximation. As the initial truncated series, (1) with coefficients from Table 3 are used. Approximation expressions in the form of a rational fraction take, as in (9), the form:

$$L\left\{\sum_{i=1}^{n} c_{ij} t^i\right\} =$$
$$=\left(\frac{1}{p}\right)^2 \cdot \frac{b_4 p + b_3 p^2 + b_2 p^3 + b_1 p^4 + b_0 p^5}{a_5 + a_4 p + a_3 p^2 + a_2 p^3 + a_1 p^4 + p^5}. \qquad (13)$$

The coefficients b_i, a_i in (13) for the corresponding plots (Table 3) are given in Table 4. Roots are also given in this table for the relevant solutions.

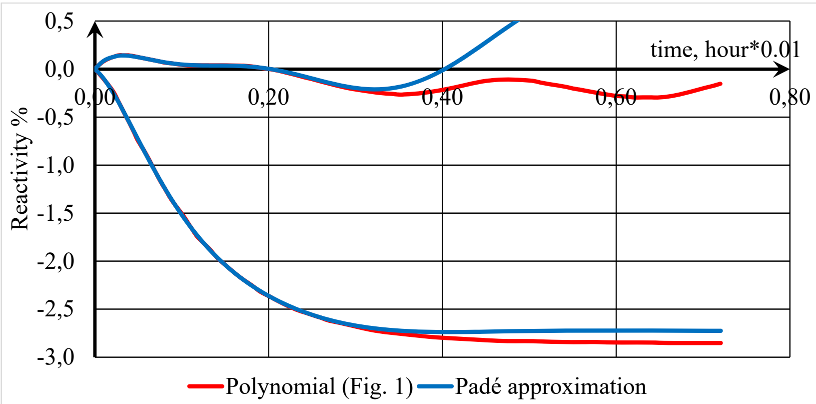

Fig. 5. Comparison of the solution representation using the approximation polynomial and the Padé approximation for the case of increase in load

Table 3

Coefficients of approximation polynomials for changing reactivity while reducing the reactor load

Coefficients of polynomials	Plot No. (Fig. 8)			
	1	2	3	4
c_{10}	7.61E+03	–9.52E+03	–4.28E+06	–1.68E+05
c_9	–2.96E+04	5.91E+03	1.07E+07	6.27E+05
c_8	5.12E+04	4.94E+04	–1.11E+07	–9.76E+05
c_7	–5.32E+04	–1.12E+05	6.23E+06	8.18E+05
c_6	3.83E+04	1.12E+05	–2.00E+06	–3.95E+05
c_5	–2.09E+04	–6.35E+04	3.56E+05	1.08E+05
c_4	8.93E+03	2.22E+04	–2.62E+04	–1.46E+04
c_3	–2.86E+03	–4.78E+03	–1.35E+03	4.68E+02
c_2	5.97E+02	6.06E+02	3.71E+02	8.13E+01
c_1	–5.41E+01	–3.49E+01	–1.99E+01	1.10E+01

To assess the adequacy of the approximation of the representation of the solution in the form (9), the originals from these expressions were found and compared for some functions with the solutions in Fig. 2:

$$f_1(t)=\begin{pmatrix}2.70904- \\ -5.63636i\end{pmatrix}\cdot e^{(-11.158-3.00707i)\cdot t}+$$
$$+\begin{pmatrix}2.70904+ \\ +5.63636i\end{pmatrix}\cdot e^{(-11.158+3.00707i)\cdot t}+$$
$$+\begin{pmatrix}0.00171661- \\ -0.0309462i\end{pmatrix}\cdot e^{(-5.11004-15.0588i)\cdot t}+$$
$$+\begin{pmatrix}0.00171661+ \\ +0.0309462i\end{pmatrix}\cdot e^{(-5.11004+15.0588i)\cdot t}-$$
$$-8.63531\cdot e^{-4.77013\cdot t}+3.21379, \qquad (14)$$

for plot 1, Fig. 2;

$$f_4(t)=\begin{pmatrix}1.00256- \\ -1.54385i\end{pmatrix}\cdot e^{(-14.1613-11.8714i)\cdot t}+$$
$$+\begin{pmatrix}1.00256+ \\ +1.54385i\end{pmatrix}\cdot e^{(-14.1613-11.8714i)\cdot t}+$$
$$+\begin{pmatrix}0.0450934+ \\ +0.371363i\end{pmatrix}\cdot e^{(-3.56517-17.5659i)\cdot t}+$$
$$+\begin{pmatrix}0.0450934- \\ -0.371363i\end{pmatrix}\cdot e^{(-3.56517+17.5659i)\cdot t}-$$
$$-2.42919\cdot e^{-18.9827\cdot t}+0.333883, \qquad (15)$$

for plot 4, Fig. 2. The compared plots are shown in Fig. 6.

Fig. 6 shows plots demonstrating the results of the approximation for two boundary data series available: load reduction by ΔP=3000 MW (from 3000 MW to 0 MW) and by ΔP=750 MW (from 3000 to 2250 MW). They cover the rest of the data. The comparison results for other data are similar. It follows from Fig. 6 that up to 30 hours (0.30 on the plot), the representation of the results of the solution of the initial model using polynomial approximation and the Padé approximation constructed on its basis are practically indistinguishable and, therefore, interchangeable. This corresponds to the results shown in Fig. 2 obtained for the case of increased load.

Table 4

Padé approximation coefficients and the roots of its denominator for the case of reactor load reduction

Coefficient No.	plot 1		plot 2		plot 3		plot 4	
	a_i	b_i	a_i	b_i	a_i	b_i	a_i	b_i
5	1.61E+5	–	9.91E+5	–	1.41E+7	–	2.08E+6	–
4	6.72E+4	5.18E+5	2.18E+5	7.06E+5	1.56E+6	1.21E+6	3.29E+5	6.95E+5
3	9.94E+3	–4.4E+4	2.29E+4	–1.1E+5	9.10E+4	–5.9E+5	2.79E+4	–1.2E+5
2	7.70E+0	–1.4E+4	1.45E+3	–1.1E+4	3.38E+3	–1.6E+4	1.54E+3	2.11E+3
1	3.73E+1	–8.2E+2	5.60E+1	–7.4E+2	8.04E+1	–8.6E+2	5.44E+1	–1.8E+2
0	–	–5.4E+1	–	–3.5E+1	–	–2.0E+1	–	–6.2E+0
Root								
p_1	–4.770		–18.144		–28.140		–18.983	
p_2	–11.158–3.007i		–13.306–6.878i		–20.076–17.042i		–14.161–11.871i	
p_3	–11.158+3.007i		–13.306+6.878i		–20.076+17.042i		–14.161+11.871i	
p_4	–5.110–15.059i		–5.631–14.548i		–6.068–26.205i		–3.565–17.566i	
p_5	–5.110+15.059i		–5.631+14.548i		–6.068+26.205i		–3.565+17.566i	

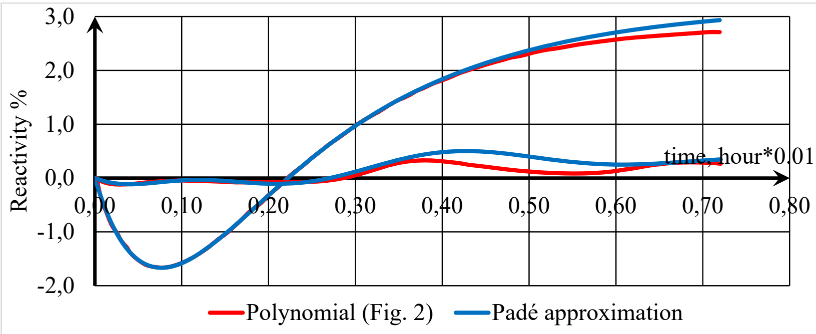

Fig. 6. Comparison of the solution representation using an approximation polynomial and the Padé approximation for the case of load reduction

5. 2. Automated control system for changing the energy release by a nuclear power plant

Mathematical models describing the properties of nuclear power units with the VVER-1000 reactor are well known and are given, in particular, in [22–24]. The simulation model used in this work is based on them. Its adequacy was confirmed in [25] by comparison with the experimental data obtained during the field experiment at the power unit No. 3 of the South Ukrainian NPP. The discrepancy between the calculated and experimental values did not exceed 1 %. The main source of error in such models is the error of the initial data and the computational error of numerical methods.

It is also necessary to take into consideration the adaptation of models in the development of information support for ACS. Mathematical models of the control sections of the nuclear power unit are presented in the form of specialized software and are regulated by norms and standards [3]. Models have varying complexity and are used to solve various control problems. For direct control of technological equipment, the simplest linearized mathematical models are used, it is with their help that local controllers are configured. The most complex mathematical software is used to calculate the neutron-physical properties of the reactor zone. Such a model is a system of nonlinear differential equations. Calculations on this model are performed to predict the state of the reactor and take into consideration the entire history of operation of the current nuclear fuel campaign [19, 20]. To control NPU, the operator has at his disposal nomograms compiled according to a previously calculated mathematical model, and regulations prescribing the order of actions depending on the state of the technological parameters. However, the presence of these documents does not guarantee control without erroneous actions on the part of the operator [21].

To study the control system, a hybrid model of NPU was synthesized. It consists of a model presented in [25] with the addition of a developed approximation model describing changes in reactivity from xenon poisoning depending on the magnitude of the perturbation.

The simulation model of NPU was based on mathematical models of a nuclear reactor [16] and power-technological equipment of the second circuit [17]. The simulation model of NPU as a control object in a general form takes the form:

$$control\ objectphysical\left(\delta h_{SCV};\delta h_{i,r};\delta C_b;N_e\right)=$$
$$=AO;Q_i;\delta t_{r,w,out};\delta t_{r,w,av};p_{st}$$

Fig. 7 shows a diagram of a simulation model of a NPU, which consists of the following models:

– a steam generator model

$$SG\left(t_{sg,w,in}(\tau);G_{sg,w,out}(\tau)\right)=t_{sg,w,out}(\tau);p_{st}(\tau);$$

– a model that takes into consideration the transport delay of the heat carrier that circulates in the pipelines from the reactor to the steam generator and in the opposite direction

$$PL\left(t_{sg,w,out}(\tau);t_{r,w,out}(\tau)\right)=t_{r,w,in}(\tau);t_{sg,w,in}(\tau);$$

– a turbo generator model

$$TG\left(p_{st}(\tau);N_e\right)=G_{st}(\tau);N_t(\tau).$$

To illustrate the intrinsic feedbacks in a control object, part of the object is represented as interrelated functions. The function of dependence of the change in neutron flux density on the change in total reactivity is $\Phi_i=f\left(\delta\rho_i\right)$. The function of dependence of the change in energy release on the neutron flux density and fuel temperature is $\delta Q_i=f\left(\delta\Phi_i;t_{i,f}\right)$. The function of dependence of the change in the average temperature of the heat carrier of the first circuit on the change in energy release and temperature of the heat carrier of the first circuit at the inlet to the reactor is $\delta t_{r,w,av}=f\left(\delta Q_i;t_{r,w,in}\right)$. The function of the dependence of the change in the temperature of the heat carrier of the first circuit at the outlet of the reactor on the change in energy release and temperature of the heat carrier of the first circuit at the inlet to the reactor is $\delta t_{r,w,out}=f\left(\delta Q_i;t_{r,w,in}\right)$. The function of dependence of changes in fuel temperature on changes in energy release and heat carrier temperature of the first circuit at the reactor inlet is $\delta t_{i,f}=f\left(\delta Q_i;t_{r,w,in}\right)$. The function of dependence of changes in reactivity as a result of poisoning of the ^{137}Xe core on changes in neutron flux density is $\delta\rho_{i,Xe}=f\left(\delta\Phi_i\right)$. The function of dependence of the change in reactivity on the change in energy release is $\delta\rho_{i,N}=f\left(\delta Q_i\right)$. The function of the dependence of the change in reactivity on the change in the average temperature of the heat carrier is $\delta\rho_{i,t}=f\left(\delta t_{r,w,av}\right)$. The function of the dependence of the change in reactivity on the change in the concentration of boric acid in the heat carrier of the first circuit is $\delta\rho_{i,b}=f\left(\delta C_{i,b}\right)$. The function of the dependence of the change in reactivity by the regulatory group of the regulatory body is $\delta\rho_{i,r}=f\left(\delta h_{i,r}\right)$.

In the above scheme of the NEU simulation model, there are the following symbols: δh is the deviation of the position of the control rods; δC_b – deviation of the concentration of boric acid in the heat carrier; N_e – electrical power of the installation; AO – axial offset; $t_{r,w,out}$ – the temperature of the heat carrier of the first circuit at the outlet of the reactor; $t_{r,w,in}$ – the temperature of the heat carrier of the first circuit at the inlet to the reactor; $t_{r,w,av}$ – the average temperature of the heat carrier of the first circuit; p_{st} is the saturated steam pressure at the inlet to the steam turbine; $t_{sg,w,in}$ – the temperature of the heat carrier of the first circuit at the entrance to the steam generator; G_{st} – steam consumption; $t_{sg,w,out}$ – the temperature of the heat carrier of the first circuit at the entrance to the steam generator; N_t is the thermal power of the turbine; i – the value of the required current parameter for the calculation group; Q_i – energy release in the fuel cell of the core; $t_{i,w,out}$ – the temperature of the heat carrier of the first circuit at the outlet; δh_{SCV} – the deflection of the position of the stop-control valve.

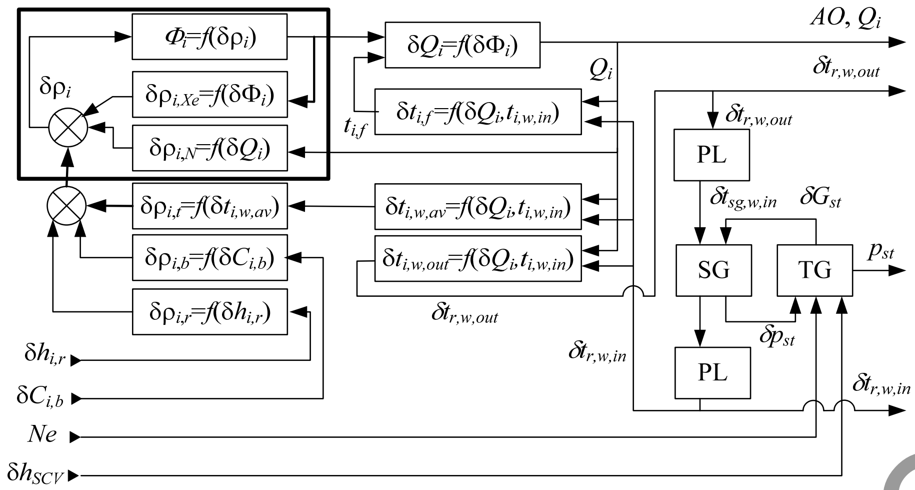

Fig. 7. Scheme of the simulation model of a nuclear power unit as a control object

The values of the functions $\delta\rho_{i,N} = f(\delta Q_i)$, $\delta\rho_{i,Xe} = f(\delta\Phi_i)$ и $\Phi_i = f(\delta\rho_i)$ are obtained as a result of calculation according to the full physical-mathematical model

$$control\ object physical\left(\delta h_{SCV};\delta h_{i,r};\delta C_b;N_e\right) = \\ = AO;Q_i;\delta t_{r,w,out};\delta t_{r,w,av};p_{st},$$

and describe the state of the core at a stationary power level with minimal error [16, 17]. However, they cannot be used in operational control during power maneuvering because of the long duration of calculations by those computational means that are allowed for operation at NPP. To solve this contradiction, the approximation model was used

$$control\ object approximation\begin{pmatrix}\delta h_{SCV};\delta h_{i,r};\\ \delta C_b;N_e\end{pmatrix} = AO;$$

$$Q_i;\delta t_{r,w,out};\delta t_{r,w,av};p_{st},$$

derived in 5. 1.

Directly for measuring changes in reactivity caused by a change in the concentration of xenon on a running reactor are not possible. Therefore, an automated system for adjusting the energy release by NPU is synthesized, taking into consideration internal disturbances due to the occurrence of "xenon oscillations".

To ensure the stable state of the NPU reactor with VVER-1000, it is necessary to maintain a constant AO value and, at the same time, control the change in the energy release field, which can adversely affect the reactor plant as a whole due to internal disturbances. Therefore, the automated control system has three control circuits in its composition:

– the first maintains the set value or changes the reactor power in accordance with the instructions of the dispatcher by adjusting the concentration of boric acid in the heat carrier of the first circuit;

– the second maintains the necessary value of the axial offset by changing the position of SPS rods;

– the third maintains the specified value of the technological parameter (temperature or pressure) of the heat carrier of the second circuit by controlling the position of the control valves of the turbine unit.

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

– to obtain the effect of the "iodine pit", the concentration of BC should remain unchanged with a decrease in power, that is, the regulator should not respond to changes in power;

– when the power unit is returned to maximum power, the concentration of BC should differ from the original value that was before the maneuver. This is necessary to compensate for the change in the concentration of ^{135}Xe and ^{135}I caused by the maneuver;

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

Therefore, a cascading control scheme was adopted. The internal regulator maintains the set value of the concentration of boric acid, the external (corrective) sets the concentration value depending on the power of the NPU. At the same time, the corrective regulator takes into consideration the change in the properties of the reactor core caused by the magnitude of the load.

Fig. 8 shows the structural diagram of ACS for changing the power of NPU.

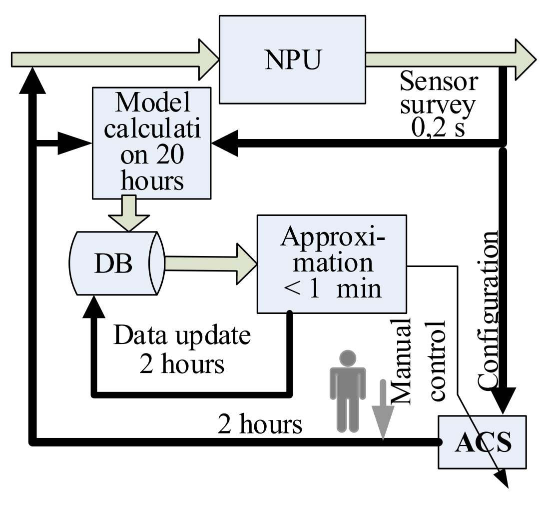

Fig. 8. Structural diagram of the automated control system for changing the power of a nuclear power unit

Calculation according to the approximation model makes it possible to predict the processes in the core in advance for an interval of about 24 hours or for 3 shifts of operational personnel. Such a forecast is enough to perform a power maneuver, which is performed by the operator within two hours. The consequences of the power maneuver are manifested for about 30 hours. Therefore, after 18–20 hours of operation of the nuclear power unit, the control subsystem receives updated calculation data for the model

$$\textit{control objectphysical}\left(\delta h_{SCV};\delta h_{i,r};\delta C_b;N_e\right)=$$
$$=AO;Q_i;\delta t_{r,w,out};\delta t_{r,w,av};p_{st},$$

and recalculates the coefficients of the approximation model. According to the updated coefficients, the coefficients of the power regulator are adjusted. This method of operation underlies the work of the power maneuvering subsystem.

The power correction regulator C_N is synthesized on the basis of the standard PI control law. An adaptation unit has been added to the standard regulator. The diagram of the power regulator is shown in Fig. 9.

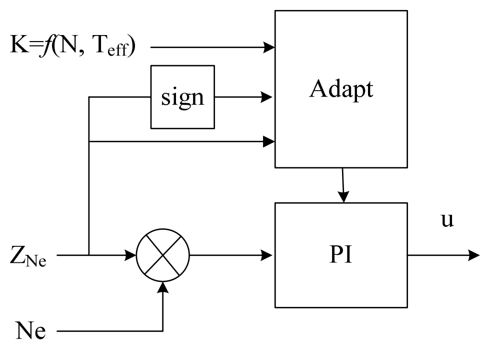

Fig. 9. C_N corrective regulator structure

At the input of the adaptation unit, the specified values of electric power, the sign of its change, and the vector of coefficients of the application model are sent. The main difference between the C_N power regulator is to adjust the k_P and $T_И$ settings. To this end, based on the coefficient vector of the approximation model $K=f(N, T_{eff})$, the value and sign of the change in electrical power in the Adapt block, the k_P and $T_И$ settings are calculated. The calculation of the coefficient vector is based on the method of obtaining transfer functions set forth in 5. 1. The control effect is supplied as an input regulator of the concentration of boric acid C_{Bor}.

After determining the settings of the C_N and C_{Bor} regulators, a calculation experiment was conducted. The purpose of the experiment was to compare the quality of power control transient processes in modeling based on the linearized control objectphysical model and the control objectapproximation approximation model.

Fig. 10–13 show the transient processes for adjusting the power of the power unit corresponding to the weekly maneuver. The maneuver consists in the following schedule of changes in electrical power: on working days during the daytime, the power is maintained at 100 %, at night the power is reduced to 80 %. On weekends (Saturday and Sunday) the capacity is reduced to 60 %.

The dashed line shows the plots for a weekly power maneuver under the control of a traditional ACS. A solid line shows a maneuver of power under the control of the modernized ACS.

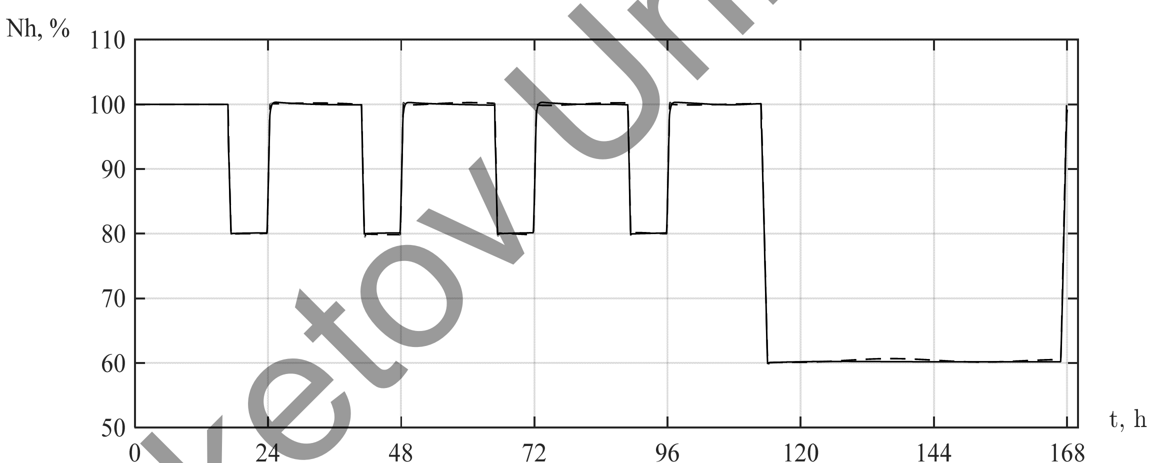

Fig. 10. Dependence of changes in the thermal power of a nuclear reactor for weekly load regulation

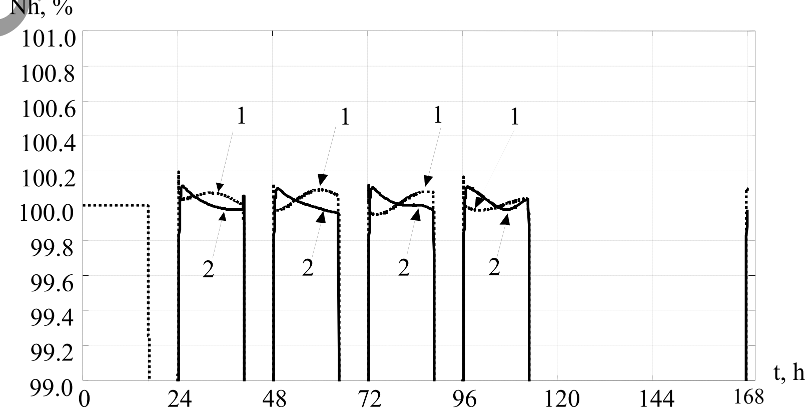

Fig. 11. Dependence of the change in the thermal power of a nuclear reactor for weekly load regulation (increased scale)

The plot of changes in the capacity of NPU is shown in Fig. 10, 11. Fig. 11 demonstrates small (up to 0.2 %) fluctuations in the power of NPU both in the traditional method of regulation and in the boron method.

With the traditional method of control, the power is reduced by changing the position of the control bodies of ACS (ACS CB). At the same time, it is not possible to control the axial offset, and it changes due to internal processes of changing the concentration of xenon Xe and Samarium Sm atoms. In the synthesized ACS, the power change is made by changing the concentration of boric acid in the heat carrier of the first NPU circuit. The resulting oscillations of the axial offset are compensated by the movement of ACS CB.

To maintain the constancy of the thermal capacity of the core, the traditional control system moves ACS CB. A plot of change in the position of the ACS CB is shown in Fig. 12. In this case, the value of ACS CB movement reaches 43 cm. With boron regulation, the maximum ACS CB deviation is 3 cm.

The plot of changes in AO shown in Fig. 13 demonstrates that with the traditional control method, during periods when the capacity remains constant, the AO changes. A change in AO while maintaining a constant capacity of NPU indicates that fluctuations in the energy release field occur in the core.

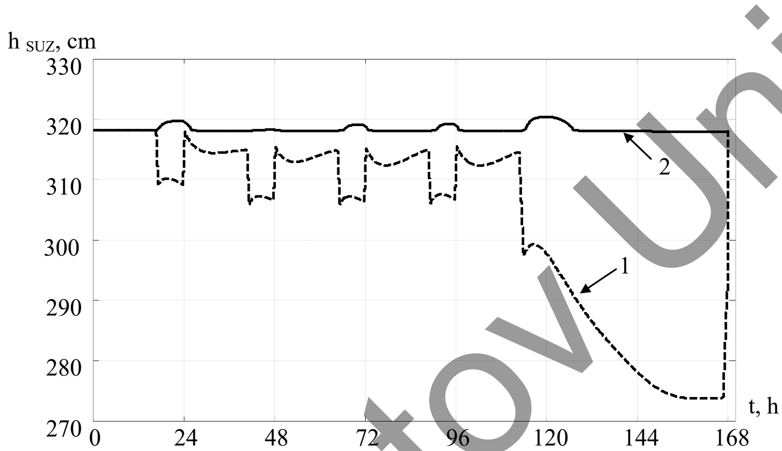

Fig. 12. Dependence of the change in the position of the control rods of a nuclear reactor with a weekly load control

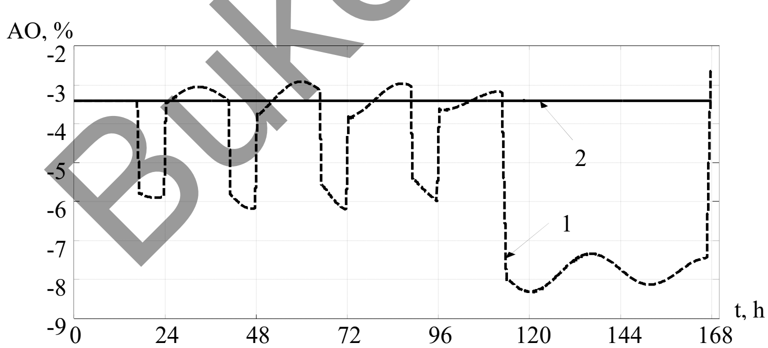

Fig. 13. Dependence of axial offset change in a nuclear reactor with weekly load control

The plot of changes in AO demonstrates that in the traditional ACS, the nature of the oscillations changes after each power maneuver. The reason for this is xenon and samarium poisoning and, as a result, a change in the power effect of reactivity, the plots of which are shown in Fig. 1, 2. Thus, the consequences of changes in the capacity of NPUs are manifested within two to three days. Moreover, the continuing changes in power change the nature and magnitude of reactivity and, as a consequence, the change in AO and thermal power, as can be seen in Fig. 11, 13. When using the new ACS, the absolute change in the value of AO is +0.015 % −0.01 % and is not shown on the scale of Fig. 13.

6. Discussion of results of studying the system of automated control of energy release of a nuclear power unit

Objectively existing intrinsic properties of the control object do not allow for automatic control due to the impossibility of creating an adequate linear model. Compensation for disturbances arising during the control of a nuclear power unit is carried out by the operator in manual mode by influencing the relevant regulatory bodies when receiving current information from the sensors.

The method described in this article provides for the ability to build a transfer function of the control objectapproximation model of some process (7). The results of its implementation and the results obtained on the basis of a detailed nonlinear physical-mathematical control objectphysical model are adequate at a certain time interval with the same initial data (Fig. 5, 6). This approach makes it possible to build an automated control system to compensate for the reactivity of a nuclear power unit.

In many cases, if it is necessary to use a simple model to implement a control system for some process and the presence of a complete and, accordingly, complex model, one is to go the way of simplifying the latter. This path is the solution to a direct problem. In this case, the ability to take into consideration some important properties of the managed process may be lost.

The proposed method proceeds from the solution to the inverse problem. As initial data, not only the parameters of the process are considered but also the results of the solution based on both models. They must match. It can be assumed that the created model, albeit simpler in structure, adequately reflects the processes occurring in the controlled object and determining nuclear safety. This approach, based on the results of its implementation, can be compared with the tuning (training) of a neural network. Although they are different in the method of implementation.

Comparisons of the resulting solution and the control objectphysical solution for cases of increasing and decreasing the load (Fig. 5, 6) showed their full compliance at a time interval of ~24 hours. This corresponds to three shifts in work of the operational personnel of the nuclear power plant and is enough from a physical point of view to compensate for the first half-period of the xenon oscillatory process. This order of things has made it possible, for cases of increasing or decreasing power within specified limits for each individual process based on the obtained transfer functions of approximation models, to calculate in advance the tuning parameters of the power regulator. And then, as the power of the NPU changes and the effective days have passed, adapt the settings of the standard power regulator to the new properties of the control object. The advantage of a given method is that the model of the object of control is not known in advance, and not only its coefficients but also its structure. The synthesis of the object model is carried out depending on the operational parameters of the nuclear power units.

The designed ACS makes it possible to control the change in the power of a nuclear power unit with the elimination of intrinsic oscillatory (xenon) processes that occur when the state of energy release changes under an automated mode. Such regulation will make it possible to ensure the maneuvering properties of the power unit to ensure a balance between the economic feasibility of adjusting the load of the power system and the safe operation of the nuclear power plant. The balance is primarily ensured by maintaining a constant main value that determines nuclear safety – the axial offset at the required regulatory value (Fig. 13). Economic feasibility increases significantly for two reasons. First, NPU acquires a new property in the ability to control the power system. Second, the number of movements of control bodies decreases, which increases the likelihood of non-depressurization of fuel rods (Fig. 12).

The proposed method was used in the development of ACS by changing the energy discharge by NPU to ensure an unchanged quantitative measure of stability of the reactor core in the form of an axial offset. Our studies have shown that the use of three control circuits made it possible to reduce the power level of NPU without additional control actions (Fig. 10). The amount of power varied from 1000 MW(e) to 800 MW(e) as opposed to the basic modification to 935 MW(e). In [14], the results of energy release modeling were reported confirming the correctness of the developed methodology. In the basic version, the axial offset oscillation that appeared at each subsequent cycle of the maneuver introduced an additional internal disturbance, which characterizes the depth of immersion of the absorbing rods of the controllers. In the designed ACS variant, this effect is absent while the axial offset is maintained constant, which does not affect the subsequent characteristics of the cyclic change in the power level.

The limitation of the devised method is the need for preliminary calculation of approximation models based on the results of neutron-physical calculation of NPU and subsequent calculation of settings. Such calculations should be carried out depending on the results of modeling the change in reactivity during the fuel campaign. Determining the frequency of calculations of approximation models of individual controllers and finding its optimal value is task for the further study. Such studies will eliminate the limitations of the proposed method.

7. Conclusions

1. To obtain information about the course of nonlinear dynamic processes of the core, an approximation model was built and a transfer function was determined that describes changes in the reactivity of a nuclear power unit in the form of a single function. This function was obtained by a two-dimensional approximation of the dependence of reactivity on the period of the fuel campaign (the beginning or its end) and on the magnitude and sign of the change in the power level (increase or decrease). The solutions found for all cases of load changes showed their compliance at a time interval of ~24 hours (three shifts of work of operational personnel). Based on the obtained transfer functions of the approximation models, the tuning parameters of the power controller are calculated in advance. As the power of NPU changes and the effective days that have passed, the power controller settings are adapted to the new properties of the control object. The advantage of this method is the fact that the proposed approximation model of the object of control can be integrated into the control system of NPU. The synthesis of the object model is performed by determining its structure and coefficients as necessary and depending on the operational parameters of NPU.

2. A control method has been devised and, on its basis, the ACS has been synthesized for changing the energy release by NPU to ensure the specified value of the quantitative measure of stability of the core in the form of an axial offset. For this purpose, the approximation model of the core in the form of a transfer function was integrated into the control system. The results of our study showed that such a model is adequate to the physical-mathematical one over a long-time interval (more than 24 hours). Taking into consideration the properties of the reactor core in the arising nonlinear dynamic processes during operation for the subsequent time interval is carried out with the help of a physical-mathematical model that makes it possible to obtain new coefficients for the existing approximation model.

The ACS consists of three control circuits. The first includes a power unit power regulator and a boric acid concentration regulator. It maintains the set value or changes the energy release of the core in accordance with the instructions of the dispatcher by adjusting the concentration of boric acid in the heat carrier of the first circuit. The second control circuit includes an axial offset regulator and maintains its desired value by changing the position of the absorbing rods. The third control circuit includes a comprehensive controller of the thermal parameter of the second circuit of NPU. It maintains the set value of the heat carrier pressure of the second circuit by controlling the position of the control valves of the turbine unit, which ensures the constancy of the temperature at the inlet to the core of the nuclear power unit. The advantage of the proposed control scheme is its adaptation to the existing control and protection systems of nuclear power plants. It uses standard controllers and channels for measuring technological parameters. Only a change in the information links between the standard controller systems is required. Such a scheme of ACS has made it possible to eliminate the occurrence of xenon oscillations and maintain the axial offset at a given value. The size of the offset ensures the stability of the core not only with stationary energy release but also with the transition from one power level to another.

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