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A MODEL OF A POWER UNIT WITH VVER-1000 AS AN OBJECT OF POWER CONTROL

М.В. Максимов, К.В. Беглов, Т.О. Цисельська. **Модель реакторної установки енергоблоку з ВВЕР-1000 як об'єкта керування потужністю.** Досліджується математична модель енергоблоку ВВЕР-1000 за компромісно–комбінованою програмою регулювання з точки зору розподілу значень найважливіших технологічних параметрів по висоті активної зони (АКЗ) реактора. Розглядається вплив положення органів регулювання системі управління та захисту (ОР СУЗ) на розподіл нейтронного потоку по висоті АКЗ реактора та, відповідно, на розподіл температури палива.
Ключові слова: реактор, АКЗ, ВВЕР-1000, ОР СУЗ, програма регулювання.

М.В. Максимов, К.В. Беглов, Т. А. Цисельская. **Модель реакторной установки энергоблока с ВВЭР-1000 как объекта управления мощностью.** Исследуется математическая модель энергоблока ВВЭР-1000 по компромиссно-комбинированной программе регулирования с точки зрения распределения значений важнейших технологических параметров по высоте активной зоны (АКЗ) реактора. Рассматривается влияние положения органов регулирования системы управления и защиты (ОР СУЗ) на распределение нейтронного потока по высоте АКЗ реактора и, соответственно, на распределение температуры топлива.
Ключевые слова: реактор, АКЗ, ВВЭР-1000, ОР СУЗ, программа регулирования.

M.V. Maksimov, K.V. Beglov, T.A. Tsiselskaya. **A model of a power unit with VVER-1000 as an object of power control.** The mathematical model of a power unit with VVER-1000 is studied by the compromise-combined control program in terms of height distribution of the most important core technological parameters. Namely, influence of control and protection system regulating group location on the neutron flux height distribution and, accordingly, on the fuel temperature distribution, is considered.
Keywords: reactor, core, VVER-1000, control and protection system regulating group, control program.

The major feature of electric power consumption in the power system consists in non-uniformity of daily, weekly and seasonal diagrams of electrical loading.

It is mentioned in numerous works devoted to the problem of operation of contribution cycling power units to energy-producing that the increase of nuclear demands to raise of Nuclear power plant (NPP) equipment maneuverability, i.e. to operate NPPs in the variable mode [1…3].

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The character of circuit work parameters change (pressure, temperature, flux) is important when choosing a control program (CP).

The CP choice depends on many physical, constructive, technical and economic features of a power unit and the mode of its use in the power supply system.

The known CP don't provide an accurate setting of technological parameters.

Hence a new CP was offered in [4]. It was called the "compromises-combined CP".

Practically, realization of any CP is carried out by the automatic power controller of a NPP unit. However, before its implementation, a CP must be checked using a model.

Thus, operation of nuclear units according to the compromise-combined CP must be studied. Therefore the aim of this paper is creation of a simulation model for a power unit with VVER-1000.

For VVER-1000, the mathematical model (model), considering all difficult dynamic processes, is described by a system of non-linear differential equations.

Though point modeling is often used for studying engineering problems, this approach is not adequate because of impossibility to monitor main parameters of reactor, e.g., change of the coolant temperature height distribution, the axial offset height distribution (AO), etc.

Therefore it was accepted to develop a model of reactor, using sectionalization of the core.

The structure of each zone model is identical, but zones differ in geometrical and heat–hydraulic parameters, as well as in their static and dynamic properties.

Structurally, a zone model consists of nine blocks:

— point reactor model linking the neutron flux density with the reactivity;

— model of energy release in the fission material linking the quantity of released fission energy with the neutron flux density;

— model of heat transfer in a fuel rod linking the fuel temperature with the quantity of released energy;

— model of convective heat transfer a fuel rod wall to the coolant linking the coolant temperature with the quantity of released energy;

model of the regulating unit control group efficiency;

— model of influence of the boron coolant concentration on the reactivity;

— model of the reactivity poisoning effect due to xenon;

— model of the fuel temperature reactivity effect;

— model of the coolant temperature reactivity effect;

The last three blocks consider reactivity effects during an industrial reactor operation.

The reactor kinetics is described a system of 7 differential equations considering 6 groups of delayed neutrons [5]:

$$\begin{cases} \dfrac{dn}{dt} = \dfrac{(\rho - \beta_{ef}) \cdot n(t)}{l} + \sum\limits_{i=1}^{6} \lambda_i \cdot C_i \\ \dfrac{dC_i}{dt} = \dfrac{\beta_i \cdot n(t)}{l} - \lambda_i \cdot C_i \end{cases}, \tag{1}$$

where

ρ — reactivity;

β_{ef} — aggregate share of delayed neutrons;

β_i — contribution of the *i*-th group of delayed neutrons;

$n(t)$ — neutron density, n/cm^3;

l — medium lifetime of instantaneous neutrons generation, s;

λ_i —radioactive decay constant for the *i*–th group of predecessors, s^{-1};

C_i — effective concentration of the *i*–th group predecessors, n/cm^3.

The energy release in the core described by the following equation [6]:

$$q_z = \frac{v \cdot V_t \cdot \Sigma_f^5 \cdot E_f^5 \cdot \beta c \cdot k_r \cdot x}{2 \cdot m \cdot n_{tv} \cdot \sin(\beta c \cdot H/2)} \cdot \cos(\beta c \cdot z) \cdot n(t)\,, \tag{2}$$

where

q_z —linear heat rate, W;
v — velocity of thermal neutrons, cm/s;
V_t — fuel volume, cm^3;
Σ_f^5 — macroscopic fission cross-section for U-235, cm^{-1};
E_f^5 — energy of one U-235 fission, J;
βc —coefficient considering features of profiling. For VVER-1000, βc =0,842;
k_r — heat release radius variation factor. k_r=1,35;
x — part of heat release in the fuel rod. In VVER–1000, x=0,96;
z — coordinate of the medium point of each zone, cm;
m — quantity of cartridges in the core, pcs;
H — core height, cm;
n_{tv} — number of fuel rods in the cartridge, pcs.

The heat transfer process the fuel rod wall can be described by the following differential equation

$$\frac{m_{ct} \cdot c_{pct}}{\alpha \cdot F} \cdot \frac{dt_{ct}}{d\tau} + \Delta t_{ct} = \frac{1}{\alpha \cdot F} \cdot \Delta q_z + \Delta t_w^{out}\,, \tag{3}$$

where

c_{pct} —wall heat capacity, J/kg·K;
m_{ct} — fuel rod cladding wall mass, kg;
α — convective heat exchange coefficient, W/m^2·K;
F — heat exchange area, m^2;
t_{ct} — temperature of the fuel rod cladding wall, °C;
Δt_{ct} — cladding wall temperature drop, °C;
Δq_z — energy release gain, W;
Δt_w^{out} — exit coolant temperature gain, °C.

Heat transfer from the fuel rod wall to the coolant can be described by the following differential equation

$$\frac{m_w \cdot c_{pw}}{\alpha \cdot F + g \cdot c_{pw}} \cdot \frac{dt_w^{out}}{d\tau} + \Delta t_w^{out} = \frac{\alpha \cdot F}{\alpha \cdot F + g \cdot c_{pw}} \Delta t_{ct} + \frac{g \cdot c_{pw}}{\alpha \cdot F + g \cdot c_{pw}} \Delta t_w^{in}\,, \tag{4}$$

where

m_w — the mass of the coolant per one fuel rod, kg;
c_{pw} —coolant heat capacity, kJ/kg·K;
g — expense of the coolant per one fuel rod, kg/s;
t_w^{out} — exit coolant temperature, °C;
Δt_w^{in} — inlet coolant temperature gain, °C.

The regulating group efficiency model has been obtained for the Zaporozhye NPP unit № 5 (Fig. 1).

Best of all this dependence is analytically approximated by the cosine dependence

$$\rho_{cps} = a + b \cdot \cos(c \cdot h + d)\,, \tag{5}$$

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where

$\rho_{суз}$ — the reactivity imported by regulating group;

a, b, c, d — the fitted coefficients;

h — group height, cm.

Model of boron concentration coolant influence on the reactivity:

At rather small (less than 6 g/kg) changes of boron concentration in circuit $\Delta\rho_{бор}$ are find using the equation [7]

$$\Delta\rho_{бор} = \alpha_{бор} \cdot (C_1^{(2)} - C_1^{(1)}). \tag{6}$$

$\alpha_{бор}$ for the interval $[C_1^{(1)},\ C_1^{(2)}]$ is determined by the expression:

$$\alpha_{бор} = -\frac{N_A}{A_{бор}} \cdot \gamma_{т.н.} \cdot \theta \cdot \frac{\sigma_a^B}{\sigma_a^5 \cdot N_5^0} \cdot \frac{\bar{\Phi}_{т.н.}}{\bar{\Phi}_{т.}^0} \cdot \frac{V_{т/н}}{V_т}, \tag{7}$$

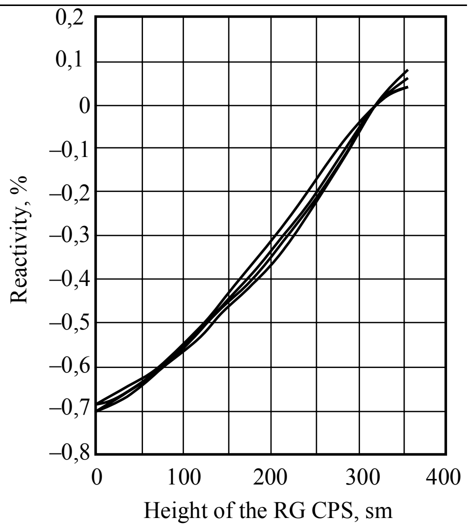

Fig. 1. Efficiency of regulating by the regulating group

where

N_A — the Avogadro number, mol^{-1};

$A_{бор}$ — nuclear mass of boron concentration, ;

$\gamma_{т.н.}$ — coolant density, g/cm^3;

θ — utilization factor of thermal neutrons;

σ_a^B — effective section of boron cross–uptake, cm^2;

σ_a^5 — effective section of ^{235}U cross–uptake, cm^2;

N_5^0 — concentration of ^{235}U atoms, cm^{-3};

$\bar{\Phi}_{т.н.}$ — medium value of thermal flux density penetrating into the boron in the coolant, cm^{-1} c^{-1};

$\bar{\Phi}_{т.}^0$ — value of thermal flux density penetrating into the fuel, cm^{-1} c^{-1};

$V_{т/н}$ — coolant volume in core, cm^3;

$V_т$ — fuel composition volume in core, cm^3.

A dependence for boric acid concentration changing in the primary circuit was obtained in [7]. Hence, the model is described by the following equations.

Injection of boron concentration solution

$$T_1 \cdot \frac{dC_{H_3BO_3}}{d\tau} + \Delta C_{H_3BO_3} = k_1 \cdot \Delta G_{H_3BO_3}. \tag{8}$$

Injection of a pure overhead product

$$T_2 \cdot \frac{dC_{H_3BO_3}}{d\tau} + \Delta C_{H_3BO_3} = k_2 \cdot \Delta G_{H_2O}, \tag{9}$$

where

$C_{H_3BO_3}$ — boron concentration, g/kg;

k_1, k_2 — transfer factors, $\frac{\text{g/kg}}{\text{kg/s}}$;

T_1, T_2 — time constants, s;

$\Delta G_{H_3BO_3}$ — change of boric acid flow, kg/s;

ΔG_{H_2O} — change of pure overhead product flow, kg/s.

The transfer factors differ 2.5 times, and the time constants differ 2 times, for processes of injection and excretion of boron concentration from coolant. Extraction of boron concentration lasts longer.

Model of the reactivity effect due to xenon poisoning.

At any moment, the losses of the reactivity due to xenon poisoning are determined by the following dependence

$$\rho_{\mathrm{Xe}}(t) = -\frac{\sigma_a^{\mathrm{Xe}} \cdot \theta}{\sigma_a^5 N_5(t)} \cdot N_{\mathrm{Xe}}(t), \qquad (10)$$

where

σ_a^{Xe} — microscopic cross-section of thermal capture by ^{135}Xe, cm^2;

σ_a^5 — microscopic cross-section of thermal capture by ^{235}U, cm^2;

$N_5(t)$ — concentration of ^{235}U atoms, cm^{-3};

$N_{\mathrm{Xe}}(t)$ — concentration of ^{135}Xe atoms, cm^{-3};

Θ — thermal neutrons utilization factor for the case of a unpoisoned reactor.

Change of ^{135}Xe concentration can be described by

$$\frac{dN_{\mathrm{Xe}}}{dt} = \gamma_{\mathrm{Xe}} \sigma_f^5 N_5(t)\Phi(t) + \lambda_J N_J(t) - \sigma_a^{\mathrm{Xe}} N_{\mathrm{Xe}}(t)\Phi(t) - \lambda_{\mathrm{Xe}} N_{\mathrm{Xe}}(t), \qquad (11)$$

where

γ_{Xe} — specific yield of Xe nucleus;

σ_f^5 — ^{235}U thermal fission microscopic cross–section, cm^2;

$\Phi(t)$ — neutron flux density, n/cm^2·s;

λ_J — constant of J radioactive decay, 1/s;

$N_J(t)$ — nucleus J concentration, cm^{-3};

λ_{Xe} — constant of Xe radioactive decay, 1/s.

The ^{135}J concentration rate is found as a difference between the ^{135}J generation rate (as immediate fission product) and its decrease (at the expense of β-decay)

$$\frac{dN_J}{dt} = \gamma_J \sigma_f^5 N_5(t)\Phi(t) - \lambda_J N_J(t), \qquad (12)$$

where

γ_{Xe} —specific yield of J nucleus due to fission.

The system of equations (10)...(12) determines the reactor xenon poisoning dynamics.

Model of reactivity effects due to the fuel temperature and the coolant temperature.

The effect of reactor reactivity change due to the change of the temperature of coolant was term the reactor temperature.

Temperature reactivity effect is generally determined by the following equation

$$\rho_t(t_T) = \int_{20^o}^{t_T} \alpha_t(t_T) dt_T, \qquad (13)$$

where $\alpha_t(t_T)$ — temperature reactivity coefficient.

For small changing of coolant temperatures, the temperature reactivity coefficient value can be considered constant, so the effect of reactivity is determined by the following equation

$$\Delta\rho_t = \alpha_t(t_T) \cdot \Delta t_T. \qquad (14)$$

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For operational personnel it would be sufficient to have only one reactor performance characteristic — dependence of the reactivity on the thermal power $\rho = f(N_p)$.

The power reactivity effect is determined as

$$\rho_t(N_P) = \int_0^{N_P} \alpha_N(N_P)\,dN_P\,, \tag{15}$$

where $\rho_t(N_P)$ is quantity of reactivity change due to increase of power from 0 to a given level N_p.

However, dependence $\rho_t(N_P)$ is ambiguous and changes depending on reactor operating conditions. This fact forces us to use in physical accounts by various components temperature reactivity effect. With that end in view all VVER's temperature reactivity effect with high-temperature fuel divide on two material components — fuel temperature reactivity effect and coolant temperature reactivity effect.

The data for calculation of the equations (1), (2), (6), (7), (10) and (11) have been obtained from [1, 5, 6].

The dependences for the temperature reactivity effect and for the power reactivity effect have been obtained for the Zaporozhye NPP unit 5.

In terms of the model described above, an experiment to define basic technological exponents has been made.

As we suppose that regulating units are moved to the level 350...301 cm (considering from the core bottom), change of boron concentration in the primary circuit is ±8 g/kg, departure of coolant temperature on inlet the core is ±5ºC.

Modelling has been made for a model with four zones. The following zones from the top to the bottom are considered: 350...325 sm — I zone (marked by the dark blue line on the diagrams), 325...301 sm — II zone (marked by the green line on the diagrams), 301...175 sm — III zone (marked by the red line on the diagrams), 175...0 sm — IV zone (marked by the blue line on the diagrams). The specified numerical values denote distance from the core bottom.

Power decrease was modelled by moving of regulating units and, then, by injection of boron acid (Fig. 2, *a*, 3, *a*, 4, *a*). The influence of boron injection on specified parameters distribution is shown in Fig. 2, *b*, 3, *b*, 4, *b*. The steps done after the thermal conditions have been obtained, are shown in Fig. 2...4.

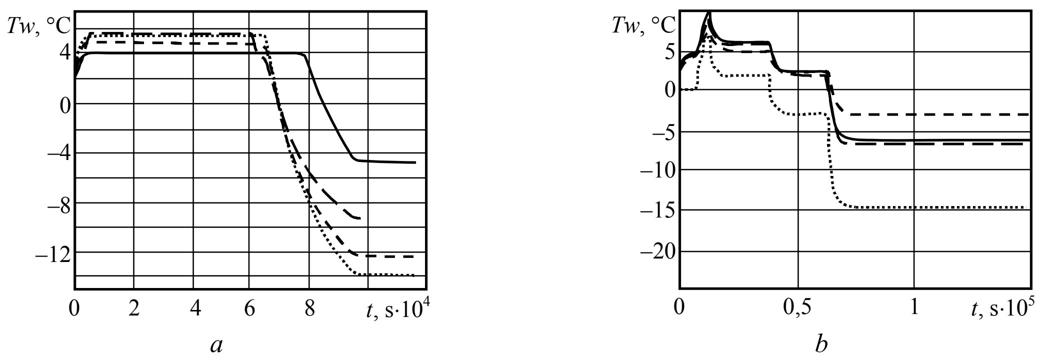
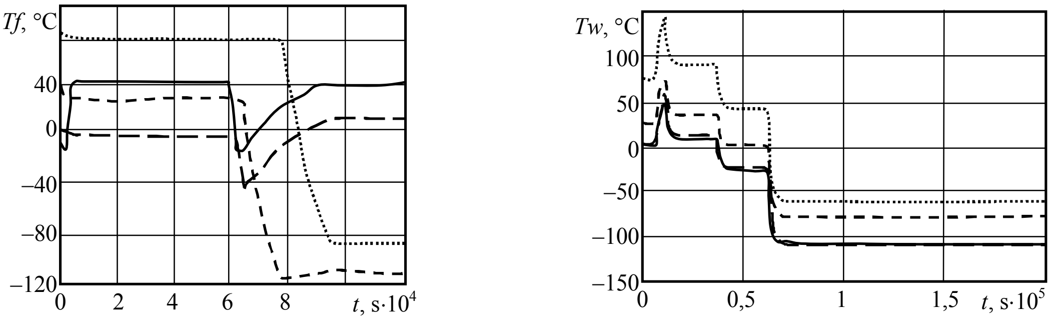


Fig. 2. Coolant temperatures core height distribution

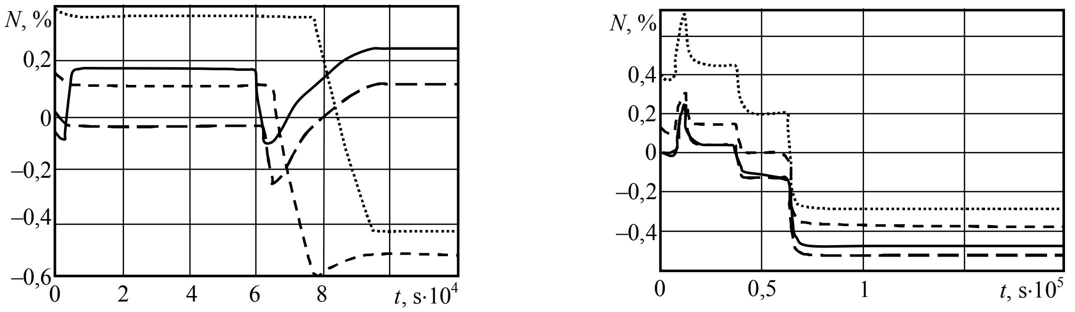
A mathematical model of the VVER-1000 power unit has been realized in the graphical dynamical system modeler and simulator Xcos Scilab package [8].

The presented VVER–1000 core model of differs from the known ones in taking into account the influence of the location of the regulating units on the neutron flux distribution, and, accordingly, on the fuel temperature distribution.



a *b*

Fig. 3. Fuel temperatures core height distribution



a *b*

Fig. 4. Nominal neutron flux core height distribution

If can be seen that the departures of the parameters occur preferentially in the reactor's upper part, in I and II zones, where the rods of regulating units are moved.

The analysis shows that the fuel temperature change in I and II zones is allowed to be no more than 100 °C, when the reactor power change is 10 %.

If we have this allowable change of the fuel temperature, then we can compensate it moving the regulating units.

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