

Simplified dynamic model of a nuclear reactor

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Abstract. The article is devoted to the issues of nuclear energy. It considers a simplified dynamic model simulating the main processes occurring in a nuclear power plant. The basis of the proposed model is a model of heat release in the reactor, simulating the relative neutron power and temperature of the coolant in the hot thread of the circulation circuit. The position of the absorbing assemblies, which determines the reactivity introduced by them, is formed in an operational control model that simulates bringing the assemblies into a pre-start state, displacement of the control assembly under the influence of a power regulator, as well as under the action of protections of various levels. The purge/recharge model describes the dynamics of the concentration of boron absorber during water exchange between the deaerator and the circulation circuit. Models of thermal deaeration, pressure compensation and heat exchange in the circuit simulate the pressure and temperature of the coolant.

1 Introduction

Currently, the importance of new technologies for nuclear power is increasing. Nuclear energy is used in thermal power engineering when energy in the form of heat is obtained from nuclear fuel in reactors. It is used to generate electrical energy in nuclear power plants (NPP), for power plants of large marine vessels, for desalination of seawater.

Nuclear power owes its appearance, first of all, to the nature of the neutron discovered in 1932. Neutrons are part of all atomic nuclei, except the hydrogen nucleus. Bound neutrons in the nucleus exist indefinitely. In their free form, they are short-lived, since they either decay with a half-life of 11.7 minutes, turning into a proton and emitting an electron and a neutrino, or are quickly captured by the nuclei of atoms.

Modern nuclear power engineering is based on the use of energy released during the fission of the natural isotope uranium-235. In nuclear power plants, a controlled nuclear fission reaction is carried out in a nuclear reactor. According to the energy of neutrons producing nuclear fission, thermal and fast neutron reactors are distinguished [1-4].

Numerical modeling methods are widely used in many fields of science and technology. They are also used in the nuclear power industry. There are many computer programs for modeling the processes of the core of a nuclear reactor [5-8]. To describe its properties, simpler models are often used today. A lot of works have been devoted to the issue of calculating nuclear installations [9-14].

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From the point of view of a mathematical model, a nuclear reactor can be considered as a complex conglomerate of interacting processes.

The method of mathematical modeling is used in the work. The proposed model of heat release in the reactor simulates the dynamics of the thermal power of the latter, taking into account the following main effects:

- changes in the position of operational management bodies;
- changes in the concentration of boron absorber in the coolant;
- temperature effect of reactivity;
- changes in the concentration of toxic inhibitors.

2 Main part

The main unit of a nuclear power plant is a nuclear reactor, the scheme of which is shown in Fig. 1. Energy is obtained from nuclear fuel, and then it is transferred to another working body (water, metallic or organic liquid, gas) in the form of heat; then it is converted into electricity according to the same scheme as in conventional thermal power plants.

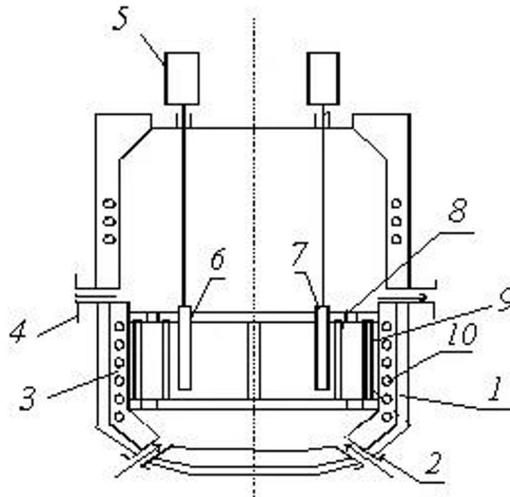


Fig. 1. Nuclear reactor.

They control the process, maintain the reaction, stabilize the power, start and stop the reactor using special movable control rods 6 and 7 made of materials that intensively absorb thermal neutrons. They are driven by the control system 5. The actions of the control rods are manifested in a change in the power of the neutron flux in the core. Water circulates through channels 10, cooling the biological protection concrete

The control rods are made of boron or cadmium, which are thermally, radiation and corrosion resistant, mechanically strong, and have good heat transfer properties.

Inside the massive steel housing 3 there is a basket 8 with fuel elements 9. The coolant enters through the pipeline 2, passes through the core, washes all the fuel elements, heats up and enters the steam generator through the pipeline 4.

The reactor is located inside a thick concrete biological protective device 1, which protects the surrounding space from the flow of neutrons, alpha, beta, gamma radiation.

Fuel elements (fuel rods) are the main part of the reactor. In them, a nuclear reaction occurs directly and heat is released, all other parts serve for insulation, control and heat removal. Structurally, fuel rods can be made with rod, plate, tubular, ball, etc. Most often

they are rod-shaped, up to 1 meter long, 10 mm in diameter. They are usually collected from uranium tablets or from short tubes and plates. The outside of the fuel rods are covered with a corrosion-resistant, thin metal shell. Zirconium, aluminium, magnesium alloys, as well as alloy stainless steel are used for the shell.

The number of circulation circuits determines the type of reactor, the coolant used, its nuclear physical properties, and the degree of radioactivity. The single-circuit circuit can be used in boiling reactors and in reactors with a gas coolant. The most widespread is the two-circuit circuit when using water, gas and organic liquids as a coolant. The three-circuit scheme is used at nuclear power plants with fast neutron reactors using liquid metal heat carriers (sodium, potassium, sodium-potassium alloys).

Nuclear fuel can be uranium-235, uranium-233 and plutonium-232. The raw materials for producing nuclear fuel are natural uranium and thorium. A nuclear reaction of one gram of fissile material (uranium-235) releases energy equivalent to 22×10^3 kWh (19×10^6 call). To obtain this amount of energy, it is necessary to burn 1,900 kg of oil.

With each fission of the uranium-235 nucleus, 2-3 neutrons are released. Of these, only one is used for further reaction, the rest are lost. However, it is possible to use them for the reproduction of nuclear fuel by creating fast neutron reactors. When operating a fast neutron reactor, it is possible to simultaneously obtain approximately 1.7 kg of plutonium-239 per 1 kg of burnt uranium-235. Thus, it is possible to cover the low thermal efficiency of nuclear power plants.

Fast neutron reactors are ten times more efficient (in terms of using nuclear fuel) than fuel neutron reactors. There is no moderator in them, highly enriched nuclear fuel is used. Neutrons flying out of the core are absorbed not by structural materials, but by uranium-238 or thorium-232 located around.

In the future, the main fissile materials for nuclear power plants will be plutonium-239 and uranium-233, obtained respectively from uranium-238 and thorium-232 in fast neutron reactors. The conversion of uranium-238 into plutonium-239 in reactors will increase nuclear fuel resources by about 100 times, and thorium-232 into uranium-233 by 200 times.

In the two-group approximation of neutron kinetics, on which the model is based, the volume concentration of neutrons is described by a system of balanced equations of the form

$$\begin{cases} \frac{dn}{dt} = (\rho_0 - \rho_b - \beta) \frac{n}{t_0} + \frac{N}{t_1} + S - r_p \\ \frac{dN}{dt} = \frac{\beta}{t_0} n - \frac{N}{t_1} \end{cases} \quad (1)$$

where n , N are the volume concentrations of neutrons and sources of delayed neutrons; ρ_0 is the operational reactivity, i.e. reactively introduced by operational control bodies; ρ_b is the reactivity introduced by the boron absorber t_1 is the lifetime of instantaneous and delayed neutrons; β is the proportion of delayed neutrons; S is the rate of generation of seed neutrons; r_p is the rate of neutron absorption toxic inhibitors.

The ρ_b value characterizes the reactivity introduced by the boron absorber and is determined by the ratio

$$\rho_b = 5,72 k_b c_b \quad (2)$$

where: k_b is the efficiency of the boron absorber; c_b is the concentration of the boron absorber in the coolant of the circuit.

The concentration of delayed neutron sources is calculated in an inertial link with a time constant t_1 , at the input of which the neutron concentration multiplied by the coefficient is fed. In turn, the neutron concentration is formed at the output of an inertial link with a time constant t_0/β , the input of which is supplied with an algebraic sum:

- generation rate due to the operational management body;
- absorption rates of toxic bore by the absorber;
- generation rates of delayed neutron sources;
- seed neutron generation rates;
- the rate of absorption by toxic inhibitors.

The reactivity ρ_0 introduced by the operational management bodies is determined by the current position of the absorbing assemblies, the last of which is considered to be the controlling one.

The rate of generation of seed neutrons is calculated on the assumption that the latter occurs during spontaneous fission of U_{238} . The change in the concentration of the latter is described by the ratio

$$N = \frac{\rho}{\mu} N_a \exp\left(-\frac{t \ln 2}{T_{1/2}}\right) \quad (3)$$

where: ρ , μ is the density and molar weight of the nuclear skinny;

N_a is the Avogadro number;

$T_{1/2}$ - half-life.

From here you can get

$$S = \left| \frac{dN}{dt} \right| \frac{\rho}{\mu} \frac{N_a}{T_{1/2}} \ln 2 \quad (4)$$

The absolute neutron power is then converted into relative N_r , which is used in further calculations. The dynamics of the neutron concentration growth is approximated by an exponential dependence, the characteristic time of which corresponds to the reactor period. The construction of this approximation and the calculation of the reactor period can be carried out using a react meter simulator.

The heat released during fission heats the nuclear fuel, which, in turn, heats the coolant due to heat transfer during flow. The heat transfer coefficient is selected so that at a nominal power, a given exchange surface and a given flow velocity, the nominal temperature of the hot coolant (i.e., the coolant in the hot thread of the circulation loop) would be provided. The temperature of the cold coolant (i.e., the coolant in the cold thread of the circulation loop) is formed by the heat exchange model in the circuit considered below. Based on the conditions of the thermal balance, it is possible to describe the dynamics of the temperatures of fuel and hot coolant by the relations of the form

$$\begin{cases} c_f m_f \frac{dt_f}{dt} = P_n - \alpha_r F_r (t_f - t_h) \\ c_h m_h \frac{dt_h}{dt} = \alpha_r F_r (t_f - t_h) - \alpha_b F_b (t_h - t_c) \end{cases} \quad (5)$$

where t_h , t_f , t_c are the temperatures of the coolant, nuclear fuel and feed water;
 c_h , c_f are the heat capacity of the coolant and nuclear fuel;
 m_u , m_f are the masses of the coolant and nuclear fuel;
 a_r , a_b - heat transfer coefficients in the reactor and steam generator;
 Fr , Fb - heat transfer surfaces in the reactor and steam generator.

To stabilize the pressure in the circulation circuit, a steam cushion formed in the pressure compensator is used. In the compensator, the coolant flows are mixed, coming from the purge/recharge deaerator of the circulation circuit through the coarse and fine injection valves, as well as heating of the coolant with electric heaters turned on. The temperature of the coolant coming from the deaerator is considered to be equal to the temperature in it, and the rate of the injection tour corresponds to the temperature of the cold coolant.

The flow rate of the coolant discharge from the compensator into the circulation circuit corresponds to the feed rate and is determined by the level regulator in the compensator. Since in the nominal mode there is a balance between the purge and recharge costs, taking into account the Torricelli formula, we get

$$f_s = \rho_H S \sqrt{2gH} \quad (6)$$

where: S is the cross section of the make-up pipeline; ρ_H is the current density of the coolant; H are the current levels of the coolant in the pressure compensator.

The formulas given in the article (1-6) not only describe the model, but also allow for the necessary thermal calculation.

3 Conclusions

In this paper, a simplified dynamic model was considered that simulates the main processes at a nuclear power plant, including heat generation in a reactor. The proposed model is based on a reactor heat release model that simulates the relative neutron power and the temperature of the coolant in the hot flow of the circulation circuit. Thus, the model allows you to describe many processes occurring in the installation.

References

1. G. Thouvenin, J.M. Ricaud, B. Michel, D. Plancq, P. Thevenin, ALCYONE: the PLEIADES fuel performance code dedicated to multidimensional PWR studies. Top Fuel 2006 International Meeting (2006)
2. A. Logg, K.-A. Mardal, N. Garth Wells Automated Solution of Differential Equations by the Finite Element Method (The FEniCS Book, 2011)
3. E.V. Amosova, A.V. Shishkin, IOP Conference Series: Materials Science and Engineering **262**, 7 (2017)
4. A.I. Kanareikin, E3S Web of Conferences **371**, 05014 (2023)
5. A.I. Kanareikin, E3S Web of Conferences **371**, 03025 (2023)
6. A.I. Kanareikin, E3S Web of Conferences **376**, 01066 (2023)
7. A.I. Kanareikin, E3S Web of Conferences **376**, 01064 (2023)
8. A.I. Kanareikin, AIP Conference Proceedings **2762**, 020007 (2022)

9. A.I. Kanareikin, IOP Conference Series: Earth and Environmental Science **990**, 012012 (2022)
10. A.I. Kanareikin, IOP Conference Series: Earth and Environmental Science **1045**, 012070 (2022)
11. J.D. Hales, Computational Materials Science **99**, 290-297 (2015)
12. Y. Rashid, R. Dunham, R. Montgomery, EPRI Report 1011308 (2004)
13. B. Philip, Journal of Computational Physics 286 143-171 (2015)
14. K.J. Geelhood, W.G. Luscher, C.E. Beyer, J.M. Cuta, FRAPTRAN: a computer code for the transient analysis of oxide fuel rods. (Washington (DC), United States Nuclear Regulatory Commission, 2011)