Using of computer code GEFEST800 at the initial stage of NPP operation with BN-800

E.F. Seleznev[1], D.S. Asatrjan[1], A.A. Belov[1], V.I. Belousov[1], I.S. Chernova[1]
G.N. Manturov[2], A.A. Peregudov[2], M.Y. Semenov[2], A.M. Tsibulya[2]
I.I. Drobyshev[3], I.V. Fedorov[3]
G.U. Dubovoj[4], D.A. Maltsev[4], E.P. Ljapin[4], A.A.Ivanov[4]

[1]Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN)
52, B.Tulskaya, Moscow, 115119, Russia
[2]State Scientific Centre of the Russian Federation – Institute for Physics and Power Engineering (SSC RF - IPPE)1, Bondarenko sq., Obninsk, Kaluga region, 249033, Russia
[3]All-Russian Research Institute for Nuclear Power Plants Operation JSC ("VNIIAES JSC")
[4]Belojarskaja nuclear power plant (BNPP) Zarechnij, Sverdlovsk region, BNPP, 624250, Russia

Abstract

The GEFEST800 code has been developed to carry out neutronic calculations of nuclear power plant operation for sodium cooled fast breeder reactor BN-800 (stationary and transient from minimum controllable power level to full reactor power with drive control rods and fuel burning). The code allows to calculate the following parameters: k_{eff}; maximum reactivity reserve; effective reactivity of control rods and control rod groups; full, specific and linear power of energy release in fuel assemblies; unevenness coefficients of energy release in fuel assemblies and reactor core; damaging irradiation dose for fuel assemblies; burning; reactivity coefficients; effective fraction of delayed neutrons; transient processes characteristics; decay energy release and many other parameters.

The code has diffusion, transport and Monte-Carlo modules. The CONSYST code with ABBN-93 library is used for constants preparation. The code has thermo mechanical module to take into account changes in the size of the cells. The presence of such module allows considering these changes in calculations of reactivity coefficients at different power levels. Services such as calculated parameters control, graphics, data preparation, analysis of calculated results are provided in interactive mode using specially developed graphical shell.

Some results of using the code at the initial stage of NPP operation with BN-800 are presented in the paper.

Key words: fast breeder reactor, neutronic calculations, efficiency, reactivity

1. Introduction

The GEFEST800 code has been developed to carry out neutronic calculations of nuclear power plant operation for sodium cooled fast breeder reactor BN-800[1].

The code has a modular structure. The main modules are: module of neutron constants preparation; module of reactor neutronic calculations in three-dimensional geometry in multi-group diffusion approximation with a point on the fuel assemblies and the point on the fuel pin; module based on the Monte Carlo method for the evaluation of nuclear safety; module of power calculation; module of fuel and boron burn up calculation; module for calculating the neutron flux and the radiation load on the construction materials; module of thermal hydraulic calculation to account for feedback in neutron-physical calculations; module of temperature calculation of structural materials in the SFA in the coolant medium or a gaseous environment; module for calculating the residual heat dissipation; module for calculating adjoin function and evaluations reactor parameters by perturbation theory; module for calculating the effective functional of reactor kinetics; module of non-stationary calculation (direct and inverse problem) in the quasi-static approximation; module of calculation uncertainties evaluation and a group of modules for operation with the fuel archives.

2. Estimation of the reactivity effects

The thermo mechanical calculation module is implemented in code to account for estimated modifications of cell dimensions, allowing these changes to take into account when calculating the temperature reactivity coefficient (TRC) and power reactivity coefficient (PRC) at different power levels.

Reactivity coefficients and effects are responsible for the security of the core of the reactor facility during operation under the influence of external conditions. Special challenge for the determination represents reactivity effects with feedback on the temperature and the geometric parameters of the core. Experimental measurement of the effects values on the existing industrial reactor facilities is difficult due to the limited resources for the registration and the time for experiments. Therefore, the values obtained experimentally have a significant uncertainty. At the same time, data calculation is conjugated with a lot of uncertainties, such as the determination of the temperature field and the fuel data. The developed calculation scheme and implemented in GEFEST800 algorithms are designed to improve the accuracy of the calculation of values of reactivity effects. Program that solves conditionally critical task in the diffusion approximation in the three-dimensional hexagonal grid («hex» module) has been modified to take into account feedbacks in the neutron-physical calculation. It adds possibility of a call the calculation algorithm considering axial movement of the fuel layers relative to each other.

The paper presents the results of calculations of the reactivity coefficients for technological and integral [2] characteristics of the core: power and temperature of the coolant at the inlet (power and temperature coefficients respectively). Also, the calculated values are compared with experimental results obtained during the 45th micro company (MC) at BN-600, and data from the start of BN-800.

The general sequence of the calculation for all effects defined in the code is as follows:

1) Formation of the initial "critical" state;
2) Formation of the initial calculation state values with predetermined process parameters, such as control rod position, power, coolant temperature at the inlet, etc.;
3) Transition to a state with the perturbed parameter values by which the corresponding effect will be determined;
4) Calculation of reactivity effect by the formula:

$$\Delta\rho(x_i; x_i^*) = \left(\frac{1}{K_{ef}^*(x_i^*)} - \frac{1}{K_{ef}(x_i)}\right)K_{кр} \quad (1)$$

where $K_{кр}$ - is the effective multiplication factor (K_{eff}) obtained by solving the conditionally critical problem for the critical state;

x_i - is a technological variable that determines the reactivity effect;

x_i^* - is a perturbed value of a technological variable, according to which the reactivity effect is determined;

$\delta x = \left|x_i^* - x_i\right|$ - the magnitude of the perturbation;

K_{ef} - K_{eff} for the initial calculation state;

K_{ef}^* - K_{eff} for calculation state with the perturbed values of technological parameters;

$\Delta\rho(x_i; x_i^*)$ - reactivity effect at the transition from the initial to the perturbed value of the technological variable.

The values of x_i и x_i^* should be understood as the asymptotic value established after the non-stationary process, since the definition of reactivity effect is carried out in the framework of a sequence of stationary calculations which does not taking into account the behavior of the object during the transition process.

The formation of the state means finding the values of functional fields of the neutron flux, power, temperature, sizes, constants corresponding to this set and values of K_{eff}. The coefficient of reactivity in a code should be understood as reactivity effect caused by a change in considered technological variable per unit of measure or normalized on the unit [2].

The papers devoted to the calculation of reactivity effects for active zones of different type of reactor use the concept of reactivity coefficients on physical parameters of the core, as the partial derivative of the reactivity function of the parameters [2], such as the temperature of the fuel in the calculated cells, the temperature and density of cladding and coolant in the calculation, and so on, by applying, for example, perturbation theory tools to obtain values. The magnitude of the effect in GEFEST is obtained from direct calculation of the core, however, the feedback mechanism may also be viewed as a sequential set of functional components that affect the K_{eff} or components value. One can identify the following among them: temperature-density, nuclear, thermo-mechanical and Doppler component.

Temperature-density component of effects. Change of all feedback components in the program is ultimately caused by the change of thermal-hydraulic parameters of the reactor core. So that calculations of fuel temperatures fields, coolant, fuel cladding and the structure is produced in "mif" program [1]. Also, it takes into account changes in coolant density, which is then converted into a nuclear concentration:

$$\rho_{j,iNa} = \rho^{Na}_{j,i}(T^{Na}{}_{i,j}) * \delta_{j,iNa},$$

where $T^{Na}{}_{i,j}$ - sodium temperature in i-th channel and j-th computational cell;

$\delta_{j,iNa}$ - sodium fraction in i-th channel and j-th computational cell;

$\rho^{Na}_{j,i}$ - sodium density in i-th channel and j-th computational cell;

$\rho_{j,iNa}$ - sodium nuclear concentration in i-th channel and j-th computational cell.

Nuclear and Doppler components. This component is caused by changes in the structure of micro- and macro- cross sections as a result of changes in temperature of nuclides included in the calculation of cells. Preparation of cross sections for GEFEST is held using CONSYST / ABBN system [3]. Version of the group constants library ABBN-93.1a [3] is used as the basis of the constants library.

Thermo-mechanical component. The fact that fuel assemblies (FA) with the fuel pins containing fuel columns of varying heights were spent during the time of operation of the reactor BN-600 brings some difficulties in modeling the reactor. In addition, the fuel column varies the sizes with the media temperature changes in the active zone. If we consider that the coolant temperature at the inlet to the core is 130-140°C while the reactor has increase power from 0,1% power to full power, than the change in the length of the assembly becomes significant. Thus, the movements of the various fuel assemblies relative to each other are at a level of 1%, even during normal operation of the reactor. The length of the fuel part in different FAs may differ up to 10% and is not always possible to combine the fuel boundaries in the fuel assembly with the boundaries of the calculating cells of reactor model. Therefore fuel length in the fuel assembly models may differ from the initial lengths up to 5%. The influence of these uncertainties on the calculation results had not been estimated early. Furthermore, fuel movement in emergency situations in different fuel assemblies with respect to each other lead to deviation growth of FA sizes and reactor model cells, thereby increasing the model inaccuracy. GEFEST analyzes such thermo-mechanical processes as deformation

(expansion) of the lower collector and axial deformation of computational cells in order to account the impact of changes of elements sizes in the core.

Deformation (expansion) of the lower collector. The size of the core cells in a hot state are determined on the basis of free expansion of the pressure collector. Let's suppose that there is model of the reactor for some initial conditions, i.e. the plane and height sizes of the model cell are given, the concentrations of nuclides are prepared and imposed in the store of fuel (FS). Number them all with the index "0". Then the dependence of changing the plane size from changing the lower collector temperature has the form:

$$h = h_0[1+\alpha(T)(T-T_0)],$$

where h_0, h - an assembly location step in the model for sodium temperatures T and T_0 respectively at the inlet;

$\alpha(T)$ - a linear expansion coefficient of the pressure collector;

Therefore the concentration of fuel, cladding and structure should be recalculated like:

$$\rho = \rho_0 \frac{1}{[1+\alpha(T)(T-T_0)]^2} = \rho_0 K_\rho,$$

where ρ, ρ_0 - nuclear concentration, initial and given to the sodium inlet temperature T, where conversion factor of given concentrations on the change of the lower collector temperature looks like:

$$K_\rho = \frac{1}{[1+\alpha(T)(T-T_0)]^2}.$$

If the lower collector temperature changes, then for the coolant we have:

$\delta^*_{Na} = \delta^0_{Na} K_\rho + (1-K_\rho)$.

Axial deformation of computational cells. Axial changing of cells sizes is associated with thermal expansion of the cell material. Let's select three materials between the materials of computational cell:

- fuel;
- fuel cladding material;
- fuel structure material.

Basic characteristics of fuel assemblies such as the cell sizes z_0, the concentration of constructional materials, fuel and coolant are stored in the complex, so all the parameters for the basic state are marked with index "0".

Height sizes of fuel cells in the case of free expansion vary depending on the fuel temperature:

$$z = z_0[1+\alpha_T(T_t, w_{pu})(T_t^i - T_{t0}^i)],$$

where z_0, z – initial (base) height size of the fuel cell and temperature resulted;

T_t^i - temperature of the fuel in the computational cell;

$\alpha_T(T, w_{pu})$ – coefficient of linear expansion of the fuel 1/°C, which depends on the temperature and the content of plutonium w_{pu} [2];

Thus, at first we read out the concentration from the TA and then we multiply them by the value of the given-size K_{zt}^i:

$$K_{zt}^i = 1/[1+\alpha_T(T_t, w_{pu})(T_t^i - T_{t0}^i)],$$

so we obtain the size corresponding to a fuel temperature in the computational cell T_t^i.

If the expansion of the fuel cladding takes place separately from the fuel, so the size of the estimated fuel cell is determined by the fuel size. Nuclides concentrations in the fuel

cladding cells are determined through multiplication by the value of the fuel cladding given-size K^i_{zs} by analogy with the value of K^i_{zt} with the coefficient $\alpha_S(T_S)$ of linear expansion of fuel cladding, 1/°C [2].

If the fuel cladding and the fuel come into contact, then the size transfer factor K^i_{zct} is calculated based on simple balance forces operating in the axial direction as follows:

$$K^i_{ztc} = 1/[\frac{1/K^i_{zt} * Y_t(T_t, bu)A_t + 1/K^i_{zc} * Y_c(T_c)A_c}{Y_t(T_t, bu)A_t + Y_c(T_c)A_c}],$$

where $Y_t(T_t, bu)$ and $Y_c(T_c)$ - the Young's modulus (MPa) of fuel materials and fuel cladding, which depend on temperature and burn up % FIMA. [2];

A_t, A_c - the cross-sectional areas of the fuel and the fuel cladding in the computational cell, m^2.

If the nuclides concentrations of FAs structure are stored at the FS in isolation, so the size of the computational cell is determined by the fuel size, and nuclides concentrations in the fuel structure cell are defined through multiplication by the value of size transfer factor of the fuel structure material K^i_{ztvs}, which is calculated similarly to the value of size transfer factor of the fuel cladding.

Two models of the partition of the core components are used in GEFEST. The physical model consists of the computational cells as physical objects, for example, it is a part of the FAs volume. The physical model may have varying boundaries, which takes into account the form changings of the simulated physical object, such as its thermal extension or displacement in space. The mathematical model is designed to solve mathematical equations, such as neutron transport equations. The array of the height layer sizes of the physical model is contained in the FS file, which contains the data on the fuel assembly. This array may be strictly individual for each of the assemblies. Nuclides concentrations contained in the FS correspond to the physical array of the height layer sizes. At the initial stage, after reading the model from the FS file, models coincide.

Let's consider an assembly having an axial partition corresponding to a physical channel in model (b) and a mathematical partition of the canal in model (a). We perform the constants convolution process from channel *b* to *a* (see Fig. 1).

The process of the cross sections preparation for the mathematical model consists of several steps. At the first step, the cross sections preparation for mathematical layers begins with the transfer of sections from channel *a* to the channel *b*. The disposal cross sections and the diffusion coefficients for the channel *a* cells are determined by the sum of the disposal cross sections and transport cross sections with layer thicknesses weights from the channel *b*:

$$\Sigma^g_{ij} = \frac{\sum_{m=1}^{n} \Sigma^g_m V_m}{\sum_{m=1}^{n} V_m}, \qquad (2)$$

where Σ^g_m - the cross section of group g at the cell *m* of channel *b*;

V_m - the volume of cell *m* of channel *b*;

Σ^g_{ij} - the resulting cross section of the group g at the cell j of channel i.

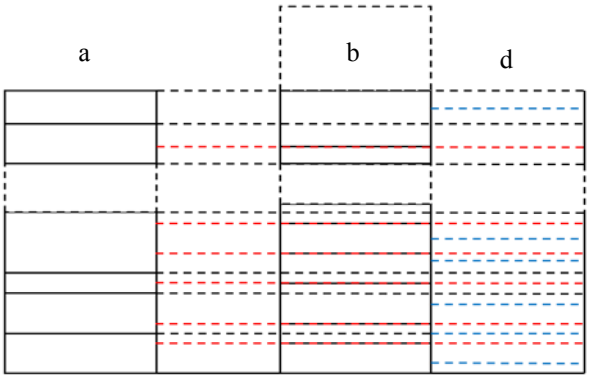

Fig.1 The process of cross sections transfer from the model of channel *b* to *a*

The current coefficients for the computational cells of the reactor mathematical model are defined after determining the disposal cross sections and the transport cross section for the channel *a* by the above-mentioned method.

At the second step, the maximum height layer of each computational cell of *b* is divided by the number of layers corresponding to the ratio of the maximum size of the layer to the minimum size of the layer. Thus, the computational cells of channel *d* are formed. All cross sections in these small cells are known, so one can proceed to the determination of the neutrons flux in the channel. Neutron flux in the channel *d* is determined by solving the one-dimensional neutron transport equation by the sweep method.

At the third and the final step, a determination of cross sections takes place in the cells of a mathematical reactor model with the weights of fine layers volume and neutron flux in them for each group of neutrons:

$$\Sigma_{ij}^{g} = \frac{\sum_{m=1}^{n} \Sigma_{m}^{g} V_{m} \varphi_{m}^{g}}{\sum_{m=1}^{n} V_{m} \varphi_{m}^{g}}, \qquad (3)$$

where φ_{m}^{g} - the value of the neutron flux in the cell *m* of channel *d*.

The calculation on the basis of the mathematical reactor model is carried out after passing through all the channels of the reactor.

Thus, if the thermo mechanical dimensional changes of physical zones of each of the reactor channels are determined in thermo mechanical module and come to the FS with recalculation of nuclides concentration in each physical zone, so the thermo mechanical changes in reactor channels up to the channel destruction can be calculated by means of above-mentioned algorithm with high accuracy. The resulting values are also compared with experimental results obtained during the 45th microcampany (MC) of BN-600 and data from the start of BN-800.

The 45th and 46th MCs of BN-600 core were simulated to test the new technique considering thermo mechanical feedbacks. The choice was made in favor of the given MCs, since there are available experimental data on the temperature and power coefficients of reactivity at various power levels and temperatures of coolant at the inlet to the core for these time intervals. Conditions were prepared and the power and temperature coefficients were calculated according to available data. Calculations were made according to (1) by the following algorithm (for example, the power coefficient): after the preparation of condition corresponding to the power level w_0, the calculation of two perturbed states at power levels $w_0 + \delta w$ and $w_0 - \delta w$ (δw - the perturbations magnitude) were made and the values of

reactivity effects $\Delta\rho_1 = \Delta\rho(w_0; w_0 - \delta w)$ and $\Delta\rho_2 = \Delta\rho(w_0; w_0 + \delta w)$ were obtained. Therefore, in this case the power reactivity coefficient (α_w) %Δ*k*/*k*/%Nnom was defined as:

$$\alpha_w = \frac{\Delta\rho_2 - \Delta\rho_1}{2\delta w}, \qquad (4)$$

The TRC (α_w)%Δ*k*/*k*/°C was determined according to this algorithm. The proportion of delayed neutrons (β_{eff}) can also be calculated, so the result obtained is given for the power coefficient in units cent/% of nominal power (Nnom), and the temperature coefficient in units cent/°C. Tables 1 and 2 presents the comparison of calculation results of the reactivity coefficients by GEFEST both with feedback algorithms (f.on) and without them (f.off) with experimental data. The data averaged over the results of six measurements for the confidence level of 68% were used as experimental data on the BN-600.

Table 1. Results of the calculation of the power reactivity coefficient on the BN-600

№ MC	№	Nnom, %	Deviation (f.off-exp)/exp	Deviation (f.on-exp)/exp
45	1	18	-18%	-15%
	2	60	-20%	-9%
	3	95	-15%	-1%
46	4	18	-5%	-1%
	5	60	23%	17%
	6	95	-15%	9%

Table 2. Results of the calculation of the temperature reactivity coefficient on the BN-600

№ MC	№	Nnom, %	Tnom, °C	Deviation (f.off-exp)/exp	Deviation (f.on-exp)/exp
45	1	60	342	-11%	-10%
	2	95	348	-6%	-3%
46	3	60	354	-6%	4%
	4	95	368	-7%	-5%

A series of calculations of reactivity effects was carried out in order to evaluate the influence of various physical effects on the PRC and TRC values. One of the components of reactivity effects was performed as a variable parameter. The calculations were made as follows: there was some value of the given component in the initial state (the component can be understood as a value or a set of values of any physical quantity $\alpha(x_i)$, for example, field of the fuel temperatures), the other value corresponding to the perturbed value $\alpha(x_i^*)$ was obtained in the perturbed state, then the effect $\Delta\rho(\alpha(x_i); \alpha(x_i)^*)$ was determined as (1), while the other components remained unchanged. These calculations were carried out at different temperature and power levels, which correspond to the experimental data. Dependences on the relative contributions of power and temperature values are shown in Tables 3 and 4.

Analyzing the results of calculations to evaluate the contributions of the various components in the value of the PRC, it is necessary to pay attention to the facts that:

- the main contribution to the effect caused by the change of the fissile material temperature and the axial extension;
- the absolute value of PRC is reduced as the contribution of the components with increasing power. So the relative contribution of the fuel temperature is also reduced, but the

effect of the expansion is growing and makes more than one third of the value of the effect at nominal power.

Table 3. Contribution of the components in the PRC value,%

Power level, % Nnom	18	60	95
PRC component			
Fuel temperature*	72,7	66,3	64,6%
Coolant density	1,8	-0,02	-0,3%
Temperature of cladding materials	1,1	2,4	1,9%
Thermo mechanical component of the axial extension	24,4	31,3	33,8%
Total	100	100	100

Table 4. Contribution of the components in the TRC value,%

Power level, %Nnom	18	60	95
TRC component			
Fuel temperature*	31,7	26,2	22,8
Coolant density	6,1	6,5	5,4
Temperature of cladding materials	4,2	5,2	3,9
Thermo mechanical component of the axial extension	0,6	4,0	7,2
Thermo mechanical component of the radial expansion of the collector	57,4	58,1	60,7
Total	100	100	100
*- a fuel temperature mean the average temperature of the nuclides contained in the fissile material, including the side and down blanket			

The influence of the uncertainties of sodium temperature at the inlet to the core and the control rod positions was also analyzed. Calculations were made with the original positions and sodium temperature and with modified sodium temperature values by 10 ° C and modified immersion depth to 1% of the working stroke in the core. The estimates obtained are shown in Table 5.

Table 5. Estimation of the PRC values deviations with the measurement conditions change,%

Power level, %Nnom	18	60	95
Changes of PRC at T_{in} change for 10°C	2,0	3,0	5,0
Changes of PRC at immersion depth of control rod positions change for 1%	7,7	1,4	0,8

The obtained deviations from the result for PRC indicate the necessity of an accurate formation of the initial calculation conditions for carrying out the comparison with experimental data.

The results on the TRC demonstrate that:

- the main contribution to the TRC value makes radial expansion of the collector and the change of the fuel temperature;
- TRC varies slightly in the power range about nominal;
- the contributions of the main components of the TRC decreases with increasing power, but at the same time the contributions of feedbacks by thermo mechanical components increases.

3. Some results of using of computer code GEFEST800 at the initial stage of NPP operation with BN-800

The ttemperature and power coefficients of reactivity were measured when launching the tests on the NPP with BN-800 on the minimally controllable level. The measurement of the power reactivity effect was also performed in the range of 0-50% of the nominal power by determining the difference between the minimum reactivity reserves at various power levels. The calculations on the model of NPP with BN-800 have been carried out according to presented data. The results are in good agreement with the experimental data, as it is presented in Table 6.

Table 6. Results of the comparison of BN-800 the effects calculations and reactivity coefficients with the experimental data

Name	Deviation from the experiment,%
TRC	5,0
PRC	-3,7
Power reactivity effect,(40-50 %Nnom)	-3,0

The results of the comparison of the calculation with an experimental determination of the control rods effectiveness at the stage of physical start are presented in Table 7.

Table 7. Comparison of calculated (C) and experimental (E) data on evaluation of the control rods effectiveness

Control rods	(C-E)/ E, %	Control rods	(C-E)/ E, %	Control rods	(C-E)/ E, %	Control rods	(C-E)/ E, %
CR1*	-5,4	CR10	2,0	AR1*	0,3	SR3	-4,8
CR2	-5,5	CR11	-0,1	AR2	-0,4	SR4	-2,2
CR3	-3,4	CR12	2,5	PSR1*	-6,5	SR5	2,9
CR4	-2,6	CR13	-3,7	PSR2	-7,7	SR6	-0,0
CR5	4,0	CR14	-5,2	PSR3	-2,5	SR7	3,5
CR6	3,3	CR15	-6,0	PSR 1÷3	-4,7	SR8	3,4
CR7	-3,8	CR16	2,0	SR1*	-2,2	SR9	-0,9
CR8	-6,9	CR1÷4	6,0	SR2	-2,9	SR1÷9	0,5
CR9	1,3						
* cr – control rod; ar – automatically rod; sr – scram rod; psr – passive scram rod							

The eexperiments on the determination of energy release distribution were conducted during the physical start of the BN-800 by γ-scanning method on ^{140}La. For states with a minimum critical mass, the preliminary γ-scanning experiment was conducted in order to assess the "background» ^{140}La activity occurred during the physical start. The main experiment was performed for the starting state of the core, the characteristics of the activity (in a different extent) was measured for the 27 fuel assemblies. Table 8 contains the deviations between the calculated and experimental values of activity.

Table 9 shows a comparison of experimental and calculated height distribution of relative activities for one of the fuel assemblies. The distributions were normalized to a value of activity in the central plane of the core (0.0 mm altitude).

Table 8. Comparison of the experimental activity and the relative calculated activity in γ-scanning experiment for the start of the BN-800

Zone	Cell	(C-E)/C, %	Zone	Cell	(C-E)/C, %	Zone	Cell	(C-E)/C, %
SE*	20-20	0,0	SE	17-26	6,6	BE*	10-10	0,0
SE	17-17	1,0	SE	24-18	-0,1	BE	09-09	-4,1
SE	13-13	4,7	AE*	12-12	-7,2	BE	32-32	-3,9
SE	22-22	1,9	AE	11-11	-0,3	BE	15-28	0,7
SE	25-25	-2,9	AE	16-26	1,6	BE	08-09	-1,5
SE	29-29	-1,6	AE	16-27	-3,0	BE	18-31	0,8
SE	19-23	0,1	AE	15-27	-1,0	BE	24-10	-2,0
SE	18-24	-0,4	AE	26-16	-3,5			
SE	17-25	5,1	AE	27-15	-2,0			
* se – small enrichment; ae – average enrichment; be – big enrichment.								

Table 9. Comparison of the calculated (C) and experimental (E) altitudinal distributions of the activity of FA with number 24-20

Height,mm	(C-E)/C,%	Height,mm	(C-E)/C,%	Height,mm	(C-E)/C,%
-400	-1,7	-100	-3,1	200	2,3
-300	-5,5	0,0	0,0	300	5,3
-200	-2,3	100	1,0	400	1,9

Conclusions

The method of reactivity effects calculation using an algorithm of thermo mechanical feedbacks accounting is realized in GEFEST. The decrease of the calculation results deviations from the experimental data was shown using the algorithm on the results of measurements of the temperature and the power reactivity coefficients of 45th and 46th MCs of BN-600. The contribution of thermo mechanical feedbacks in values of TRC and PRC was demonstrated. The results of calculations on the BN-800 model are in good agreement with the data obtained during the physical start of NPP. This justifies the applicability of the algorithm for further operational calculations. It is planned to develop the work further in the direction of using more detailed fuel physics models and thermo mechanical models.

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