CALCULATION AND EXPERIMENTAL ANALYSIS OF THE BN-800 REACTOR CORE NEUTRONIC PARAMETERS AT THE STAGE OF REACHING FIRST CRITICALITY FOLLOWED BY RATED POWER TESTING

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Abstract. The main task of the measurements at different stages of the reactor start-up (first criticality, first start and further testing of the rated power) is to obtain complete and accurate information on the monitored neutronic parameters of the core. It is essential for subsequent reactor operation and also helps to verify and improve the accuracy of calculating neutronic parameters.
Insertion of the start-up neutron source initiated BN-800 first criticality which continued till the start-up project core formed, including neutronic measurements carried out under conditions of both the minimum critical mass and start loading, at the minimum controllable power level. Then, measurements were performed at different power levels during the no-load stage (when the turbine-generator was connected to the grid, the reactor power reaching 25% of the design level, i.e. until the reactor power start-up) and the on-load stage (when the rated power was reached).
Analysis of the results of the performed measurements showed that experimental and calculated values agree well (within the declared design and experimental uncertainties):

- the minimum critical loading is determined very precisely and the start critical state is predicted with high accuracy;
- agreement between the calculated and experimental values of CR worths is proved;
- regulatory compliance for reactivity balances is confirmed;
- agreement is achieved between the calculated and measured values of fission reaction rate distribution (relative power density) in the core;
- calculated estimations of temperature and power reactivity coefficients, reactivity effect due to fuel burnup and neptunium reactivity effect are confirmed by the measurement results.

Calculation methods used for the experimental analysis are similar to those employed for design justification of the core neutronic parameters. The obtained results of the measurements and of their calculation analysis will be used in cross-verification of the GEFEST-800 computer code designed for the calculation monitoring of BN-800 core operation.

Key Words: BN-800 reactor, first criticality, first start and further testing of the rated power

1. Introduction

The main task of the planned measurements during the reactor first criticality, first start and further testing of the rated power is to obtain the maximum complete and accurate information on the reactor neutronic parameters important for confirmation of its design characteristics. This unique information is not only crucially important for subsequent reactor operation, it also helps to verify and in some cases to improve the accuracy of calculation of neutronic parameters based on the measurement results obtained at the reactor.

The process of reaching BN-800 first criticality was divided into two stages. The first stage, i.e. sub-step B1, consisted in the initial core loading that involved insertion of a start-up neutron source and formation of the minimum critical mass with installed six fixed reactivity compensation rods (FRCR). The second stage, sub-step B2, consisted in the process of reaching the minimum controllable power level (MCPL) at the minimum fuel critical mass, followed by core start loading and the entire scope of neutronic measurements conducted both at the minimum and start loadings.

After the first criticality was achieved, the next stage C started, i.e. the stage of the reactor first power start-up, that consisted in the process of putting the reactor unit into operation, from the moment of reaching the first criticality to the beginning of electric power generation. The stage also included measurements of reactivity effects at the power levels of 5%, 15%, 33%, 50% Nrated. The final stage in the reactor start-up process was stage D, the pilot reactor operation, at which a number of neutronic measurements were performed at the power levels of 67%, 85%, 95%, 98%, 100% Nrated, and some transient and emergency reactor processes and safety system operation were tested.

At the stages of start-up and rated power testing, the experimental programs were accompanied with analytical monitoring by means of the certified computer codes JARFR [1], GEFEST [2], TRIGEX [3] and MMKK [4], with BNAB-93 nuclear data support [5] and the CONSYST system for their preparation [6].

2. Critical Mass Gaining

Specific attention in the course of gaining critical mass is paid to the issues of criticality safety. The reactor sub-criticality control is based on the analysis of variation of the start-up neutron source inverse multiplication value [7] obtained from the results of neutron count rate measurements.

Initially the core was loaded with dummy fuel assemblies (FA). The process of reaching first criticality was started on January 31, 2014, with installation of a start-up neutron source (SNS) into the central core cell. After that the core was loaded step-by-step, by sequential symmetrical replacement of the dummy FAs with the standard ones according to the core load map. The number of FAs to be loaded at each step was calculated by the inverse multiplication curve.

The minimum critical loading estimated by the inverse multiplication curve for the core with six FRCRs was equal to 397 FAs, which completely coincided with the fact of reaching first criticality. The Keff value calculated with the precision MMKK code for this state was equal to 0.9999±0.00002. The analytical models for precise calculations were developed with a detailed description of the FA inner structure (with a separate consideration of fuel columns, CPS control rod absorber, fuel pin and absorber cladding, FA wrappers, coolant). The fuel loaded into FAs and the absorbing material (10B isotope) contents in the CPS rods were in compliance with their passport data.

In order to get the operation reactivity margin required to perform physical measurements, three extra FAs were loaded, with the total number of FAs in the core with the minimum critical mass equal to 400. The calculated effective K-factor was equal to 1.0014±0.00002.

On June 27, 2014, for the first time the reactor reached the minimum controllable power level (MCPL) with the core configuration that had the minimum critical loading (397 + 3 FAs).

In the minimum critically loaded core the CPS rod worth values and spatial power density distribution values were measured by gamma-scanning of the reference FAs. It was followed by arranging the start-up critical loading of the core. The BN-800 reactor with its start-up core reached its MCPL on July 26, 2014. The start-up core configuration map [8] is presented in Figure1.

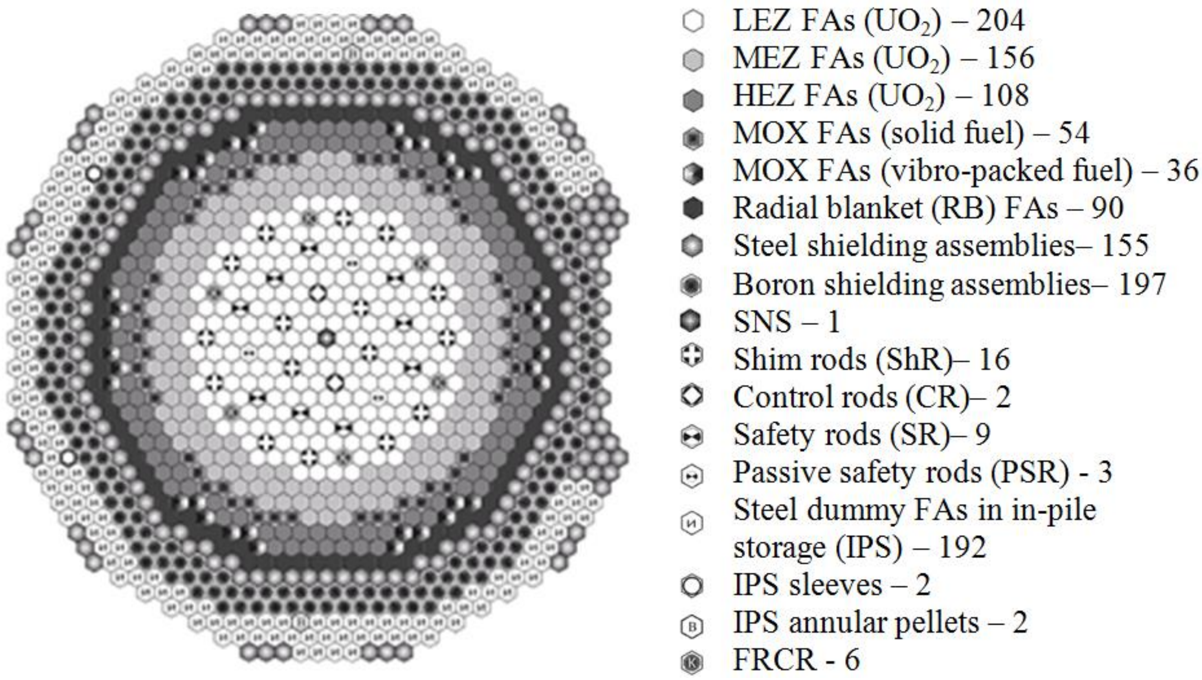

Fig. 1 - Start-up core configuration map

3. CPS Rod Worth

The CPS rod worth values were measured by the “run-away – rod-drop method” [9, 10]. The power loss curve processing time after the end of CPS rod insertion was 180 seconds. The time of SR movement from the upper position to the lower one does not exceed 1 second, the time of CR movement is 85 seconds, the time of ShR and PSR movement is 225 seconds. The estimated worth measurement error for single CPS rods is 5%, for CPS rod groups it is 10%. The comparison of calculated results with the measured values is presented in Figure 2.

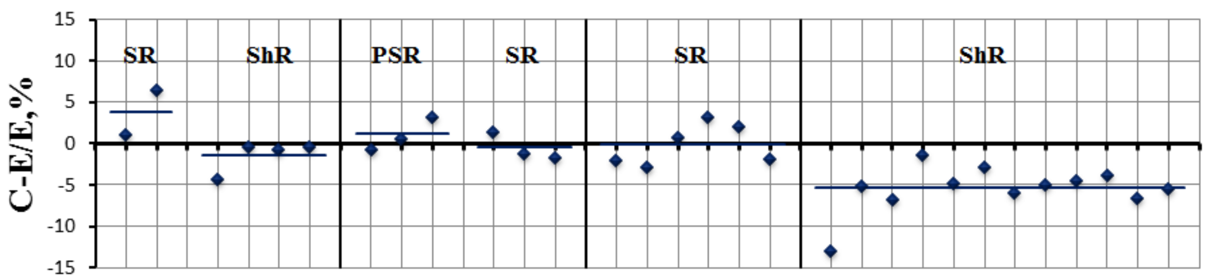

Fig. 2 – Single CPS rod worth values

The comparison of experimental and calculated values revealed the following regularity. For the safety rods that move from the upper position to the lower one within one second, a good agreement between the experimental and calculated values is observed and their discrepancy does not exceed ±3%. However, for the other rods which move during 100-200 seconds, the experimental values turn out to be systematically overestimated by 10-15%. It is mostly obvious at the comparison of experimental data for PSRs and SR-3, 6 and 9. All these rods are located at the same radius, have the same design and 10B contents; however, the PSRs move at the speed of ~200 times lower. Proceeding from that, an assumption was made on a certain dependence of the rod worth measurement results on the duration of its insertion into the core [11]. The comparison of worth values for the two CPS rod types mentioned above made it possible to determine the “time factor” that turned out to be equal to 0.9. That coefficient was used for correction of experimental values of PSR, CR and ShR worth.

Besides single CPS rods, those rod groups for which it was possible were subject to measurement of their worth. The group of CRs, SR-3, 6, 9 and the outer ring of ShR 5-16 were not measured.

Table 1 shows the measured worth values for all CPS rod groups and their comparison with calculated values. It was done by two methods: a) by summing up single rod worth values multiplied by the estimated interference factor, and b) by direct measurement of rod group worth. All the experiment results (except those for SR) have been corrected with regard to the “time factor”.

TABLE 1 – COMPARISON OF WORTH VALUES FOR CPS ROD GROUPS

CPS rod groups	Amount	Movement time, s	Calculation, %Δk/k	(C-E)/E, % total	(C-E)/E, % group
CR	2	85	0.48	+4	–
ShR 1÷4	4	225	1.21	-1	-3
PSR	3	225	1.08	0	+6
SR÷3, 6, 9	3	1	1.06	0	–
SR 1÷9	9	1	2.43	0	+1
SRs minus one (without SR-3)	8	1	2.22	0	-2
ShR 5÷16	12	225	4.44	-6	+1 ± 5*)

*) Obtained from the comparison of two critical states

It should be pointed out that the experimental value for the ShR 5÷16 group was obtained from the comparison of two critical states of the reactor: with minimum loading and in the start-up configuration. These two states differed from each other in the presence of outer ShrR ring and in the number of FAs with MOX fuel in the core.

The ShR 5÷16 group worth and its error were estimated by the least square method [12], with consideration of BNAB-93 nuclear cross-section uncertainties and calculation-experiment (C/E) discrepancies of two critical states (0.15%Δk/k) and their errors (0.2%Δk/k) [12].

4. Spatial Power Density Distrinution

With the aim to analyze spatial power density distribution in the BN-800 start-up core, the La-140 activity in the irradiated FAs was measured by means of gamma-scanning, with the use of semi-conductor high-purity germanium (HPGe) detector.

The first (background) gamma-scanning was performed at the end of the entire program of measurements in the core with minimum critical loading. The La-140 background was primarily built up in the course of reactor initial reaching the MCL and subsequent CPS rod worth measurements. The second (main) gamma-scanning was performed after arrangement of the core start-up configuration, right after the MCL was reached, before implementation of the physical measurement program. In the course of analysis of the main gamma-scanning results, consideration was given to the residual lanthanum background activity determined by calculation and measurements at the first gamma-scanning.

Figure 3 shows discrepancies between the experiment and calculation of relative La-140 activity distribution along the radius of the start-up core. From the figure it is clearly seen that the maximum deviation is equal to 5%, which is in compliance with declared accuracy of calculation under the project. It is also important to note that the results do not demonstrate any pronounced trend in relationship between calculation and experiment, the discrepancies look random. Thus, the conclusion can be made that no pronounced biases were detected in the course of calculation and measurement result processing.

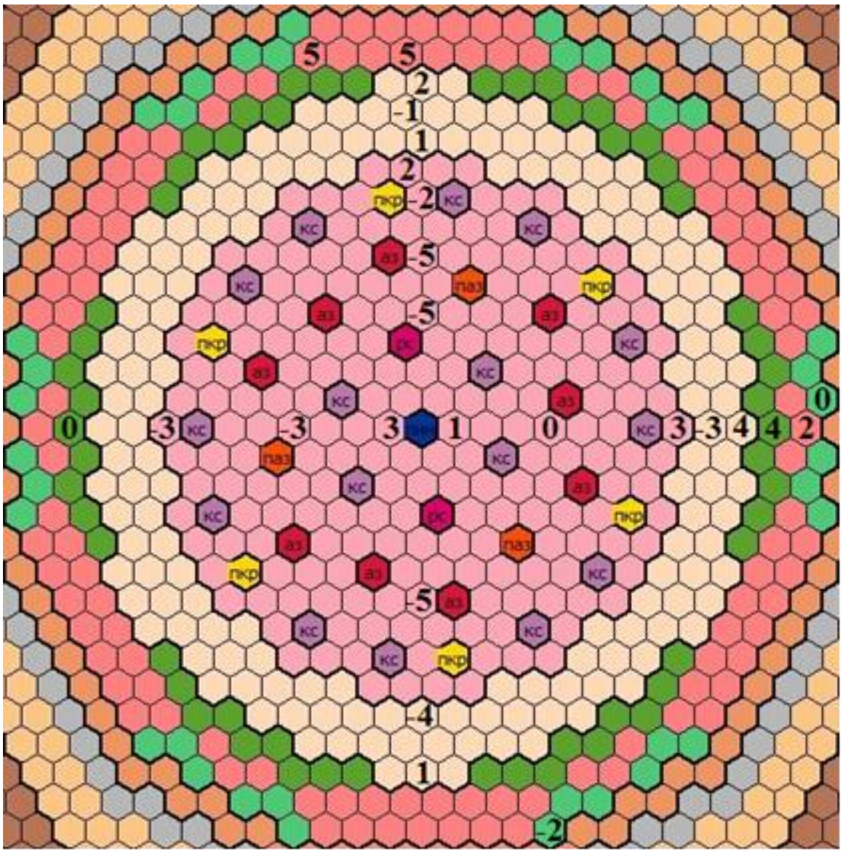

Fig. 3 – Deviation of calculation from the experiment %

5. Reactivity Coefficients

Reactivity effects and coefficients were measured both at the first criticality stage (at the reactor zero power) and at the stages of reactor start-up and rated power testing. The specific feature of the two latter stages consists in the fact that the measurements were performed at non-zero power levels of the reactor, namely, at the level of 5%, 15%, 35%, 50%, 67%, 85% and 98% of rated power.

5.1.Temperature Reactivity Coefficient

Reactivity variations related to temperature variations are described with a differential characteristic, i.e. temperature reactivity coefficient (TRC) that is determined as a reactor reactivity variation at the isotropic variation of its temperature by 1°C. All the other conditions, in particular, pump speed, are maintained constant.

The TRC value was calculated with the use of the perturbation theory of the TRIGEX code (β_{eff} = 6.71E-3). The comparison of calculated and design temperature reactivity coefficient values ($\Delta k/k*10^{-5}$) within the considered temperature range at the first criticality stage is presented in Table 2.

TABLE 2 – COMPARISON OF CALCULATED AND DESIGN TRC VALUES

TRC value	Calculation (for experimental conditions)	Design value
$(\Delta k/k/^{o}C)*10^{-5}$	-3.09	-2.97±0.60

Good agreement (within 4%) of the measured and calculated TRC values is observed, and discrepancy (calculation-experiment)/calculation with the design value does not exceed 8.5%, that is less than the measuring error (±10%). Therefore, a good agreement of design, calculated values and measured values of temperature reactivity coefficient was achieved.

The measurements of temperature (and power) reactivity coefficients were also performed at the stages of power start-up and rated power testing at various levels of reactor power. The TRE (temperature reactivity effect) measuring method at non-zero power levels is the same as at zero levels – by means of minor coolant temperature fluctuations with automatic control of reactor neutron power, but the isotropy conditions of reactor heating in this case are disturbed, different core components have different temperatures, which is inconsistent with the design approach. Approximately half of the reactivity effect measured is related to the Doppler component, which is nonlinearly temperature dependent, and it introduces specific features into the analytical interpretation of the results obtained.

For oxide fuel the Doppler effect (reactivity derivative with respect to temperature (dk_{ef}/dT)) is approximated by the dependence 1/T. Under these conditions the Doppler components of temperature reactivity coefficient are calculated using Doppler constants from the formula

$$K = C_D * \ln(T2/T1)/(T2-T1) \qquad (1)$$

where: C_D – Doppler constant

T1(reloading temperture) = 523K, (Kelvin scale temperature),

T2(inlet temperature) = 627K.

The calculation analysis performed with the reduction of the obtained results to the design conditions has demonstrated an agreement between the calculated values of temperature reactivity coefficient and the measurement results within the error of the given measurements (10%); moreover, the calculation results systematically have a more positive value, which corresponds to the conservative approach in designing.

5.2. Power Reactivity Coefficient

The power reactivity effect (PRE) at the first criticality stage was measured at low power levels (up to 0.4% Nrated) in case of staged reactor power ascension and decrease with its stabilization at individual levels. The coolant temperature at the core inlet for all power variations was maintained unchanged and corresponded to the zero-power temperature.

At the same time in the core design the power reactivity coefficient (PRC) is determined within the power range from zero to the rated one. When gaining the rated power, the average coolant temperature at the core outlet was assumed to be 547°C, the average fuel temperature – 1300°C, the average steel temperature– 480°C.

The PRC Doppler component is nonlinearly dependent on the temperature. As far as the measurements were performed at different power levels, the calculated PRC value needs the procedure of reduction to the state of measurements in the reactor. As a result, the calculated

PRC value reduced to zero power amounts to minus 0.878*10-3 βeff /MW, that within the declared measurement error agrees with the average PRC value.

The power reactivity coefficient was also measured at the stages of first start and reactor rated power testing by way of minor power oscillations at different power levels, with automatic control of the inlet coolant temperature. This approach is also out of line with the design one. Therefore, in the course of analytical interpretation of the measurement results obtained, it is necessary to reduce the calculated results to the measuring conditions again. The analytical analysis has demonstrated an agreement between the calculated valued of power reactivity coefficient at various power levels and the measurement results within the error of the given measurements (±8%).

The design value of PRC amounts to minus $0.37*10^{-5}$ Δk/k/MW, or minus 5.5 10^{-4} βeff/MW, which agrees with the calculated value reduced to the project conditions (minus $5.52*10^{-4}$ βeff/MW).

5.3. Neptunium Reactivity Effect

The neptunium reactivity effect was measured using two methods: after gaining the power (close to the rated one) and after reactor shutdown. The methods of this effect calculation, whenever possible, repeated the technique of its measurement. The analysis has shown an agreement between the calculated data (0.11%Δk/k) and measurement results within the error of the given measurements.

5.4. Fuel Burnup Reactivity Effect

Reactivity loss due to fuel burnup was measured at the stage of power testing close to the rated power level of the reactor during 10 eff. days. The performed analysis has demonstrated an agreement of the calculated values of fuel burnup reactivity effect (-0.018%Δk/k/eff.days.) and the measurement results within 5%, which is less than the measurement error.

5.5. Analysis of the Agreement between the Measurement Results and the Design Characteristics

As the calculation methods accepted in the design practice disagree with the measurement techniques, the measurement results were transformed for their reduction to the design (or close to them) conditions. The analysis performed with regard to these measures has demonstrated an agreement between the measured characteristics and their design values within the measurement error

5.6. Calculation Analysis of Reactivity Balance Results

For reactivity balance of the start-up core it is required to know the total ShR system worth value, however this worth can be measured only in the minimal critical load core, which is impossible to achieve in the start-up core. In order to obtain a similarity of ShR worth measured in the experiments in the start-up core, a “reconstructed” experiment has to be employed, which is obtained using the calculated extrapolation of the experimental data available for the minimal critical load core to the start-up core.

Table 3 presents a comparison of design, refined calculated and measured criticality parameters for the minimal critical mass and start-up core. The design values were calculated using the certified code JARFR, the refining calculation for the experimental conditions was performed using the precision code MMKK. A correction for deviation of actual fuel load that amounts to 0.15%ΔK/K from the design values should be taken into account in the analysis of Keff design values. The “reconstructed” experiment” results are also given in Table 3.

TABLE 3 – CRITICALITY PARAMETERS FOR THE MINIMAL CRITICAL MASS AND START-UP CORE.

Core state	Reconstructed experiment	Refining calculation (MMKK)	Design value
Minimal load (400 FA)	1.0000	1.0014	-
Start-up load (558 FA)	1.000	0.9999	1.000
During reloading (558 FA)	0.971±0.002	0.969	0.972 (+0.003)
	0.972±0.002 *)	0.970 *)	0.973 (+0.003) *)
After SR cocking (558 FA)	0.984±0.002	0.982	0.983 (+0.002)

*) without the most effective CPS rods

As can be seen from the table, the reactor critical state is predicted by the calculation to a high precision. The sub-criticality level after SR cocking and in the course of reloading agrees well with the design values and complies with the requirements of NP-082-07 Rules: no less than 0.01 and 0.02, respectively.

6. Conclusion

The comparison of calculated (for the experiment state) and design values of the basic neutronic characteristics of the BN-800 start-up core is presented in Table 4.

TABLE 4 – COMPARISON OF CALCULATED AND DESIGN VALUES OF NEUTRONIC CHARACTERISTICS

Parameter	Refining calculation	Design values
1. CPS worth, % ΔK/K: -SR (in the core 2CRs, 12 ShRs):	2.4	2.4
- PSR	1.1	1.0
- ShRs 1-4	1.2	-
- ShRs 5-16	4.4	4.1
2. Temperature reactivity coefficient, 10^{-5} ΔK/K/ ºC	-3.09±0.02	-2.92
3. Power reactivity coefficient, 10^{-5} ΔK/K/ MW	-0.58* -0.37**	-0.37**
4. Maximal reactivity margin, % Δk/k	4.85	4.15

*) for power range 0.3÷0.4%N_{rated};
**) for power range 0÷100 %N_{rated};

This table demonstrates a good agreement of refining calculated and design values of CPS rod groups' worth, reactivity coefficients and maximal reactivity margin, which within the declared errors agree with the measurement results. The requirements of PN-082-07 Rules on reactivity balance are fulfilled. No core engineering design adjustment is required

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