MEASUREMENTS OF POWER PROFILE OF THE BN-600 COMMERCIAL FAST REACTOR BY GAMMA-SCANNING AND ANALYTICAL STUDIES OF EXPERIMENTAL DATA

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ABSTRACT

During 25 years of operation of BN-600 fast reactor at the Beloyarsk NPP, complex of analytical and experimental measurements has been developed for control of power rate distribution in the reactor core. Continuous control is performed by computational accompaniment based on three-dimensional multi-group analysis in hexagonal geometry in diffusion approximation. Periodical control is made by measuring of power rate profile in the standard fuel subassemblies of the BN-600 reactor by gamma scanning method on the stages of updating of the reactor core. By now, two cycles of such measurements have been performed when changing for the new reactor core design 01M2 providing 4-fold refueling mode and max fuel burn-up increased up to ~11.1% h.a. In the paper given are brief description of analytical and experimental methods of monitoring of power profile of the BN-600 reactor, results of their comparison and estimation of their precision based on the results of the above studies. It has been demonstrated that the use of 26-group diffusion approximation and GEFEST, JARFR and TRIGEX codes with ABBN-93 nuclear data gives adequate description of power rate distribution among the SAs of the BN-600 reactor core. Conservative estimation of calculation error is 5%. The main concern is evaluation of power profile of peripheral areas of the radial blanket and in-vessel storage, if achieved accuracy of 10-15% is insufficient.

Key words: fast reactor, power rate distribution, experiment, diffusion approximation, calculation, accuracy estimation.

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INTRODUCTION

Relative stability of power profile is one of the advantages of fast neutron reactors. This feature eliminates the need in the special in-vessel system of SA on-line power control. Planning of on-line fuel loading and forecasting of SA irradiation parameters are mainly performed by modeling using up-to-date computer codes. Analytical studies were carried out by various Russian institutions responsible for scientific and technological justification of the core design and BN-600 reactor operation safety using three different computer codes, namely: GEFEST, JARFR and TRIGEX [1].

For the purpose of periodic control of the core, special experimental studies are carried out during normal operation of the reactor and, especially, on the stage of the core updating. Starting from the first criticality of the BN-600 reactor in 1980 till now, two techniques have been used for measurements of distribution of reaction rates and power rates in the fuel subassemblies, namely: 1). SA gamma scanning technique and 2). Needle activation detector technique. The first technique turned out to be more simple and adapted for monitoring SA power rate under conditions of the NPP commercial reactor and this technique has become standard by now. Results of experiments compared with the data of analytical forecast are used for estimation of the core condition, on the one hand, and correctness of analytical results obtained on the stages of design and operation, on the other hand. It is this system that has been developed during over 25 years in the BN-600 power fast reactor at the Beloyarsk NPP.

1. TECHNIQUE OF γ-SCANNING USED FOR MEASUREMENTS OF POWER PROFILE AND EXPERIENCE GAINED IN ITS APPLYING FOR THE BN-600 FAST REACTOR OF BELOYARSK NPP

If fresh SA is irradiated in the reactor core, then some amount of La-140 fission product used for measurements will be accumulated in the SA. Activity of fission products is proportional to the fission rate F_i in this SA and, therefore, it is important from the standpoint of justification of correspondence of SA operating modes to its design parameters. It is this characteristic that is an ultimate goal of the experimental study. The experiment is made to directly determine the intensity of 1596 keV lines of ^{140}La decay (A_i) using semiconductor detector. The measurements are relative, i.e. if the geometry is the same, then measured relative intensity of lines (A_i/A_M) coincides with the relative fission rate (F_i/F_M).

Measurement technique includes three main stages. The first stage implies for the short-duration activation of fresh SAs in the various cells of reactor core, radial blanket (RB) and in-vessel storage (IVS) in the reactor operating at the low (~0.6%) power level. On the second stage starting 60 to 100 hours after the end of SA irradiation, induced relative activity of fission product (^{140}La) is measured by the special device (semiconductor detector). And, finally, on the third stage numerical analysis of obtained results is carried out, discrepancies between analytical and experimental data are revealed and studied and uncertainty of three-dimensional distribution of power rate is estimated taking into account the results of measurements made in the reference points.

Method of measurement of SA induced activity is illustrated by Fig. 1 below. SAs (1) are put in turn in the transfer cell of the reactor at the level of radiation collimator (2) specially designed and installed for the purpose of measurements. In order to take into account radial distribution of power rate, SA might be installed facing detector by the different sides. Measurement of axial power rate distribution in the SA is made by its vertical movement using standard refueling systems. Value of gamma-quantum flux from SA entering detector is controlled by choosing appropriate shape and width of the collimator gap (3). For this purpose, required set of collimators were manufactured. As a rule, measurements of the core fuel subassemblies are performed with the same width of collimator gap. Different collimator is used for the subassemblies of the radial blanket. In order to take into consideration increase of detector counting because of change in geometry, so called special procedure is performed. Its essence is in measurements made on the same SA in two different geometries (before and after collimator replacement). Semiconductor Ge detector (6) is installed from the opposite side of the collimator. In order to decrease background and optimize relationship between pulse gaining rate in La^{140} peak and the dead time of electronic devices, detector is protected by the lead and cadmium filters (4 and 5). It should be noted that rather strict requirements are made to the duration of measurements carried out on the stage of scheduled repair. The above measures make it possible to bring measurement time to the minimum value determined by the standard procedures of SA transport from the reactor cell to the measurement area and back.

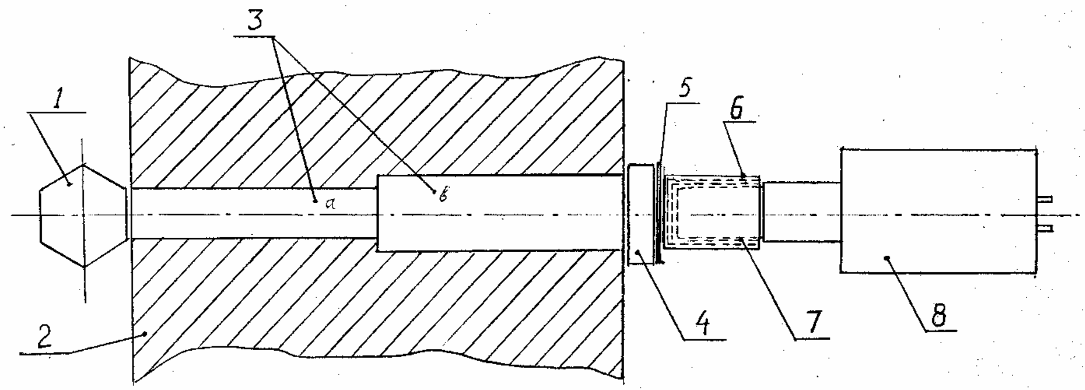


Fig. 1. Measurement of SA induced activity (1 – scanned SA, 2 – transfer cell wall, 3 – collimator, 4 and 5 – radiation filters, 6 - detector, 7 – local shielding, 8 – detector cooling unit)

Single measurement procedure gives the area of 1596 keV photon peak, and the end measured parameter is F proportional to the average fission rate in SA:

$$F_k = \frac{1}{Y_k} \cdot \frac{1}{N_k} \cdot G_k \cdot \sum_{i=1}^{N_k} \eta_{ki} \cdot D_{ik} \cdot \frac{1}{n_{ik}} \cdot \sum_{j=1}^{n_{ik}} \frac{S_{ikj}}{\Delta t_j} \cdot T(t_0, t_{вikj}). \qquad (1)$$

By now, 10 experiments have been carried out using gamma-scanning method on the various stages of updating of the BN-600 reactor core. These experiments have been performed in four cycles:

1). Measurements on the stage of the first criticality and early operation (1980-1982) with two-zone core arrangement;

2). Measurements on updated core with three power profile zones (1987-1988);

3). Measurements under steady state operating conditions with 10% design fuel burn-up (1991 - 1994);

4) Current stage of measurements in the new core with four- fold refueling mode and 11.1% max design fuel burn-up.

Besides, on the current stage, updated measurement technique is used, which is based on newly developed means of measurements and analysis. Generally, in the experiment taking about 5 days, 30 to 40 SAs are scanned including 1 to 3 SAs scanned as well over their height. Core SAs are scanned either on 2 opposite sides, or on all 6 sides, while SAs of the radial blanket and those activated in the in-vessel storage are scanned on all 6 sides. Generalized characteristics of the experiments are presented in Table 2. Results of two latest experiments performed in 2003 and 2005 and their overview are presented below in this paper. Analysis of experience gained in carrying out experimental studies has revealed the following basic sources of measurement uncertainties:

- statistical uncertainty of the number of pulses in the peak of La-140 complete absorption (~1.3% for one measurement);

- uncertainty of processing of instrumentation photon spectrum – determination of peak boundaries, subtraction of background, taking into account change of peak shape depending on the loading of measurement path (~0.3%);
- uncertainty of determination of the dead time of electronic devices and introduction of corresponding correction (up to ~1%);
- uncertainty of determination of geometric factors caused by collimator replacement (up to ~3%);
- stability of instrumentation operation (not worse than 0.5%);
- stability of measurement geometry (up to 3%).
- uncertainty caused by introduction of corrections for decay-accumulation of La-140 under radiation and after the end of irradiation of SA (below 0.3%);
- uncertainty of factor of gamma-quantum absorption by SA materials (up to 2%);
- uncertainty of relative yield of La-140 fission product (it is less than 1% for uranium fuel subassemblies and about 2% when scanning MOX fuel subassemblies).

Table 1. Description and values of parameters in relationship (1)

Parameter designations	Descriptions	Typical values
S_{ikj}	area of photon peak caused by γ-radiation of La^{140}	~ 6000-10000 pulses
Δt_j	time (duration) of measurement	~100 s
n_{ik}	number of measurements	~5
$T(t_o,t_{ikj})$	correction for accumulation-decay of La^{140}; t_o, t_{ikj} – duration of irradiation and time after the end of irradiation, respectively	t_o~ 8 hours t_{ikj}, ~ 60÷200 hours
G_k	geometrical factor for reduction of measurement results to one configuration	1.00 LEZ, MEZ, HEZ ~ 0.1 IRB, IVS ~ 0.02 ERB
D_{ik}	correction for the dead time of electronic devices	~ 1.00÷1.05
N_k	number of controlled SA sides	2 to 6
η_{ki}	factor taking into account attenuation of gamma-quantum flux by SA materials	1.0 LEZ, MEZ, HEZ, IVS ~0.77-0.79 IRB, ERB
Y_k	average yield of Ba^{140} - La^{140} chain	0.0598÷0.0587- UO_2 0.0533 – MOX fuel
k, i, j	indices of SA, SA side and measurement, respectively	

Note: LEZ, MEZ and HEZ – low, medium and high enrichment zones, respectively; IRB and ERB – internal and external radial blankets, respectively; IVS – in-vessel storage

Experience has shown that this method provides the following accuracy of measurement of average over SA cross section fission rate (for 95% confidence probability):

- for core SAs: ~3%;
- for SAs of radial blanket and IVS: ~5-6%;
- for the relative distribution over SAs of radial blanket ~1-3%.

Results of measurements of gamma quanta issued from the different SA sides are used for studying non-uniformity of power rate distribution over SA cross section as well. However, high penetrability of ^{140}La gamma quanta facilitates flattening of experimental distribution and, therefore, this method is more effective for SAs of radial blanket and less effective with respect to the core SAs.

Table 2. Generalized characteristics of gamma scanning experiments performed on the BN-600 reactor

No.	No. of refueling	Date	Irradiation parameters, W, %N_o Δt_{irr} / $\Delta t_{post\text{-}irr.}$	Features of reactor design and operating period	Features Of experiment	Number of SAs in reactor zones LEZ/MEZ/HEZ/ IRB/ERB/IVS*	SA total number	Including those scanned over height
1	First criticality	05.1980	30 % ~ 10 eff. days	Initial loading of the BN-600 reactor with two-zone core design (01)	Measurements on fresh core loading	12 / - / 14 / 2 / 2/ 1	31	3
2	5 Stage 1	04.1982	100% reactor run	Average steady state condition of the initial core 01	Measurements on spent SAs	13 / - / 16/ 3 / 1 / 1	34	-
3	5 Stage 2	07.1982	0.5 % 8.7 hr / -	Average steady state condition of the initial core 01	First measurements at the low power level	18 / - / 16 / 1 / 0 / 2	37	1
4	18	05.1987	0.8 % 6 hr / 80 hr	Start of changing for three-zone core design (01M)	Measurements on different SA designs	9 / 7 / 17 / 1 / 0/ 2	36	4
5	20	07.1988	0.6 % 5.1 hr / 60 hr	Completion of changing for 01M core	2 MOX fuel SAs	11 / 8 / 15 / 2 / 1/ 3	40	4
6	24	04.1991	0.5 % 6 hr / 63 hr	Start of changing for the core design with 10% max fuel burn-up (01M1)	8 MOX fuel SAs	12 / 12 / 16 / 4 / 1/ -	45	2
7	28	04.1993	0.5 % 6 hr / 72 hr	Completion of changing for 01M1 core	Detailed measurements in the radial blanket	8 / 3 / 7 / 7 / 5/ 2	32	-
8	30	05.1994	0.55 % 8.5 hr / 102 hr	Average steady state condition of 01M1 core	Measurement of γ-quanta absorption in SA	2 / 2 / 1 / 7 / 6/ 2	20	-
9	44	05.2003	0.6 % 7.9 hr / 96 hr	Start of changing for the core design with 11.1% max fuel burn-up (01M2)	Updated γ-scanning technique	10 / 5 / 4 / 4 / 4/ 2	29	-
10	48	04.2005	0.6 % 8.0 hr / 67.6 hr	Final stage of changing for 01M2 core	Updated γ-scanning technique	10 / 9 / 4 / 4 / 2/ 3	33	1

* LEZ, MEZ and HEZ – low, medium and high enrichment zones, respectively; IRB and ERB – internal and external radial blankets, respectively, IVS – in-vessel storage.

2. ANALYTICAL STUDIES ON POWER RATE DISTRIBUTION IN THE BN-600 REACTOR. COMPARISON OF ANALYTICAL AND EXPERIMENTAL RESULTS

By now, stable system of software and nuclear data required for analytical studies on fast reactors has been developed in Russia. This system includes up-to-date versions of the above mentioned codes, namely: TRIGEX, JARFR and GEFEST [1]. Methodological basis used by these codes is almost the same; in particular, all these codes are capable of solving neutron transport equation in three-dimensional hexagonal geometry in multi-group (18-26 groups) diffusion approximation. In 2003-2004, adoption of the common system of nuclear data ABBN-93 [2] was completed. In this system, nuclear data were prepared in multi-group approximation (299 multi-groups) and then collapsed into groups. In addition to the above codes, assimilation of another code into the practice of analytical studies on the BN-600 reactor has started recently. This code is based on Monte Carlo method and application of ABBN-93 nuclear data system, namely: MMKKENO code [3]. Now all these codes are intensively verified.

More general systems have been created on the basis of the above codes to meet demands of specific users. Operating code package GEFEST includes large fuel archive saving detailed information on the composition of each SA taking into account fuel burn-up in the reactor. Calculation of power rate in the BN-600 reactor core for the purpose of its operation accompaniment is made with detailed description of reactor fuel inventory to an accuracy of one calculation layer (out of 18 layers) in each one out of 966 fuel subassemblies. Based on TRIGEX code, ModExSys system was developed for analysis of experiments in critical assemblies and power reactors. In addition to the computer codes proper, this system included appropriate databases. Proceeding from the demands in the analysis of experimental data, in particular, those obtained on the BN-600 reactor, TRIGEX code was modified in such a way that it would be possible to make calculations with larger number of physical zones (up to 20000). Based on JARFR, a system including its archives and services was developed for analytical accompaniment of the BN-600 reactor core design. Analytical studies made by the above codes and packages give the basic and most comprehensive information on the BN-600 reactor core neutronics including data on SA power rate.

What the results of the above experiments afford and how can these be used? The direct interpretation of experimental data is hardly possible. One of the reasons is that the measurements are made on less than 10% of the core subassemblies that is fairly insufficient for building and studying purely experimental distribution of power rate and correct normalization to the total thermal power of the reactor. Therefore, measurement results can be used as reference data for comparison of analytical and experimental power profile and further interpretation of revealed discrepancies. In our opinion, three groups of conclusions and recommendations can be drawn on the basis of such comparison.

1. Systematic discrepancies can be either eliminated or minimized by the following measures:
   - modification of calculation technique and application of more correct approximations;
   - introduction of displacements and correcting or multiplying factors;
   - selection of optimum controlling parameters making it possible, for instance, to mutually compensate different components of systematic uncertainties. By way of example, there can be noted the possibility of mutual compensation of mesh uncertainty and diffusion approximation uncertainty in the calculations on the neutron field of the BN-600 reactor, thus decreasing discrepancy between the results of calculation using coarse mesh and experimental data;
2. Numerical values of discrepancies between analytical and experimental data make it possible to correctly estimate the error of calculated power rate in SA which is taken into account, on the

one hand, for evaluation of a margin for max design value (on the stage of checking correspondence of the real values to design parameters) and, on the other hand, for specification of design and operating margins (on the stage of designing, justification and licensing). On this stage, an integral estimation is made on accuracy and reliability of results of calculations by the above code packages including all possible archives and databases.

3. If experimental data exceed max permissible design values, then rearrangement of the core is needed.

In order to adequately evaluate the results of experimental studies carried out on the BN-600 reactor, work was performed on the first stage on the analysis of methodical features of the above codes and their methodical accuracy. Because of limitations on the volume of the paper, only conclusions based on the results of these studies can be given below:

- upon adaptation to the common nuclear data base, results of calculation of SA power using TRIGEX, JARFR and GEFEST codes are in mutual agreement within 1% using 1 point per hexagon of SA, thus making it possible to unambiguously eliminate possible algorithmic and other calculation uncertainties when solving transport equation;
- in spite of some difference in algorithms of taking into account coarse mesh, nevertheless, power of SA in the core and radial blanket is almost equally predicted by these codes, i.e. max discrepancy is mainly 3%, its increase up to 10-15% being only observed in peripheral cells of IVS and ERB;
- calculations made by Monte Carlo method using MMKKENO code made it possible to determine the level of uncertainty of diffusion approximation when evaluating power of the BN-600 reactor core and its main components, namely fission rates of U^{235} and U^{238} (see Fig. 2). Diffusion approximation describes quite adequately power rate distribution in the core (there is ~2% underestimation only in the boundary cells), however it produces significant uncertainty in the field gradient on the core-radial blanket boundary (up to ~ 8-10%) and in the calculation of neutron field in the IVS cells (up to 10÷20%).

Fig. 2. Results of evaluation of correction for multi-group diffusion approximation

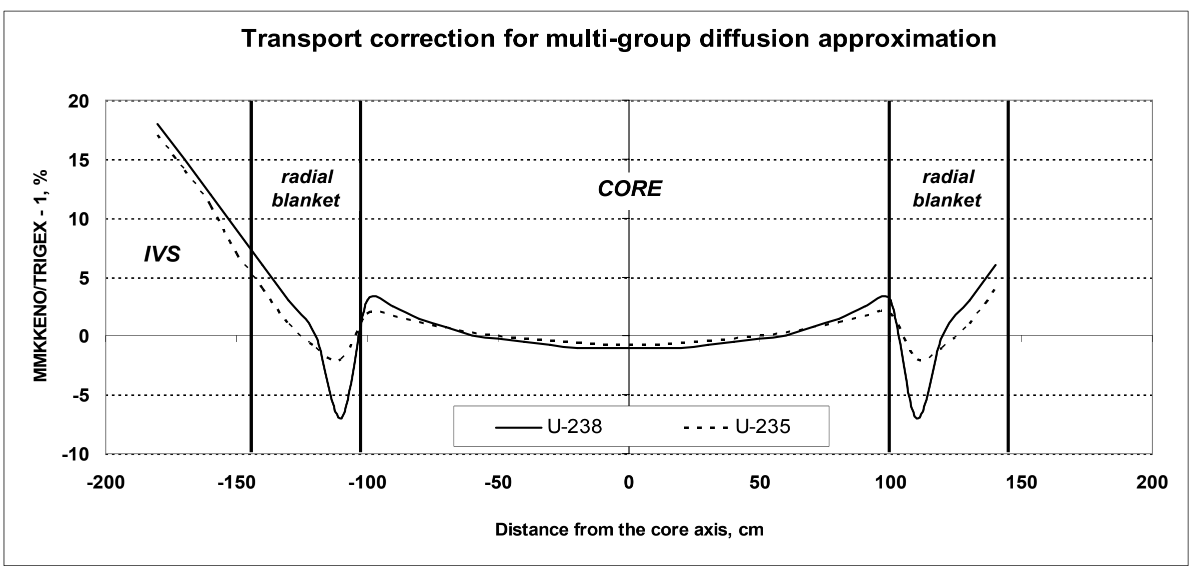


Table 3 and Figs. 3 and 4 show comparison of the results of analytical and latest experimental studies performed in 2003 and 2005. Based on this comparison, the following conclusions can be drawn:

Table 3. Discrepancies (%) between analytical and experimental data after 44 (on the left) and 48 (on the right) refueling cycles

Cells	Zones	GEFEST	JARFR	TRIGEX	MMKKENO
19-19	LEZ	-1.4	-1.0	-1.7	-2.4
19-22	LEZ (SR)	-2.9	-2.9	-3.1	-3.5
19-16	LEZ (CR)	-1.7	-1.4	-0.7	-1.5
22-22	LEZ (SHR)	1.6	-0.1	-0.1	-0.8
16-16	LEZ (SHR)	0.5	0.4	-0.7	-1.7
18-20	LEZ	-0.4	0.2	-0.5	-1.2
20-24	LEZ (SR)	-0.9	-0.6	-0.7	-0.8
19-13	LEZ (SR)	0.8	-0.2	1.9	1.3
12-11	LEZ	1.6	1.2	1.7	1.1
16-13	LEZ	1.3	1.2	2.0	1.2
20-27	MEZ	-1.8	0.4	-2.1	-1.0
18-11	MEZ (SHR)	2.9	-0.3	2.3	1.7
19-10	MEZ (SHR)	3.5	1.3	2.8	3.1
25-25	MEZ	1.6	2.3	0.5	1.7
10-10	MEZ	0.2	0.2	0.1	0.1
18-26	HEZ	-1.1	-0.7	-1.7	-0.2
26-27	HEZ	-4.3	-0.3	-1.6	-0.1
07-07	HEZ	-1.0	0.5	-0.1	0.8
20-09	HEZ	1.8	0.3	2.0	3.2
20-29	IRB	-2.0	2.9	1.4	-5.4
20-08	IRB(EOU)	-0.5	-0.3	3.3	-3.8
29-29	IRB(EOU)	-3.7	-9.5	-1.6	-5.5
05-05	IRB	-3.5	-3.1	-7.2	-7.4
21-07	ERB(EOU)	-1.0	-19.6	3.3	0.9
19-05	ERB	-4.8	-6.5	-2.1	-3.1
31-31	ERB	-8.4	1.6	-7.6	-5.1
19-32	ERB	-5.8	-7.1	-5.2	-5.7
19-33	IVS (HEZ)	-22.6	-16.8	-24.4	-20.0
19-35	IVS (HEZ)	-25.7	-22.1	-25.6	-18.1

Cells	Zones	GEFEST	JARFR	TRIGEX	MMKKENO
19-19	LEZ	0.9	2.4	0.9	0.4
19-23	LEZ (SR)	0.3	0.4	0.8	0.6
19-15	LEZ (CR)	0.4	2.0	1.4	0.7
21-22	LEZ (SHR)	0.5	0.5	-0.2	-0.7
17-16	LEZ	-4.0	-2.2	-3.3	-3.9
19-21	LEZ	-0.5	0.5	-0.3	-0.5
21-24	LEZ (SR)	-1.3	-1.2	-0.9	-1.0
19-13	LEZ (SR)	-0.5	-0.8	-0.2	-1.4
13-12	LEZ (SHR)	2.6	3.3	1.6	0.7
15-13	LEZ	1.4	2.4	1.3	0.5
17-10	MEZ	0.9	0.3	0.3	-0.2
13-08	MEZ	-1.5	-2.1	-2.0	-2.1
21-12	MEZ	1.2	0.6	0.9	0.5
24-25	MEZ (SHR)	1.9	1.1	1.1	2.3
09-10	MEZ (SHR)	1.2	0.3	0.5	0.6
22-27	MEZ (PhS)	-1.4	-1.4	0.3	1.1
20-27	MEZ	0.6	-0.1	0.7	2.4
24-14	MEZ	1.2	-0.3	0.6	0.5
14-07	MEZ	3.0	2.1	2.7	2.9
17-26	HEZ	-1.4	-2.1	-1.5	0.7
25-28	HEZ	-4.6	-5.8	-4.2	-2.7
18-07	IRB	1.7	3.8	10.4	0.5
19-29	IRB	0.8	3.8	8.4	2.1
22-09	IRB	2.1	3.7	9.5	1.9
29-27	IRB	4.0	7.4	11.5	3.7
05-07	IRB	-0.8	-2.9	4.9	-1.6
32-29	ERB	-21.5	-16.1	-14.1	-15.3
23-33	ERB	-19.5	-11.3	-11.8	-11.0
23-07	ERB	-17.0	-13.1	-10.8	-13.1
17-03	ERB	-20.2	-10.8	-15.8	-16.0
19-33	IVS	-17.0	-8.2	-15.0	-9.3
19-35	IVS	-20.4	-7.1	-20.0	-10.8

Fig. 3. Discrepancy between analytical and experimental data obtained after 44th refueling of the BN-600 reactor.

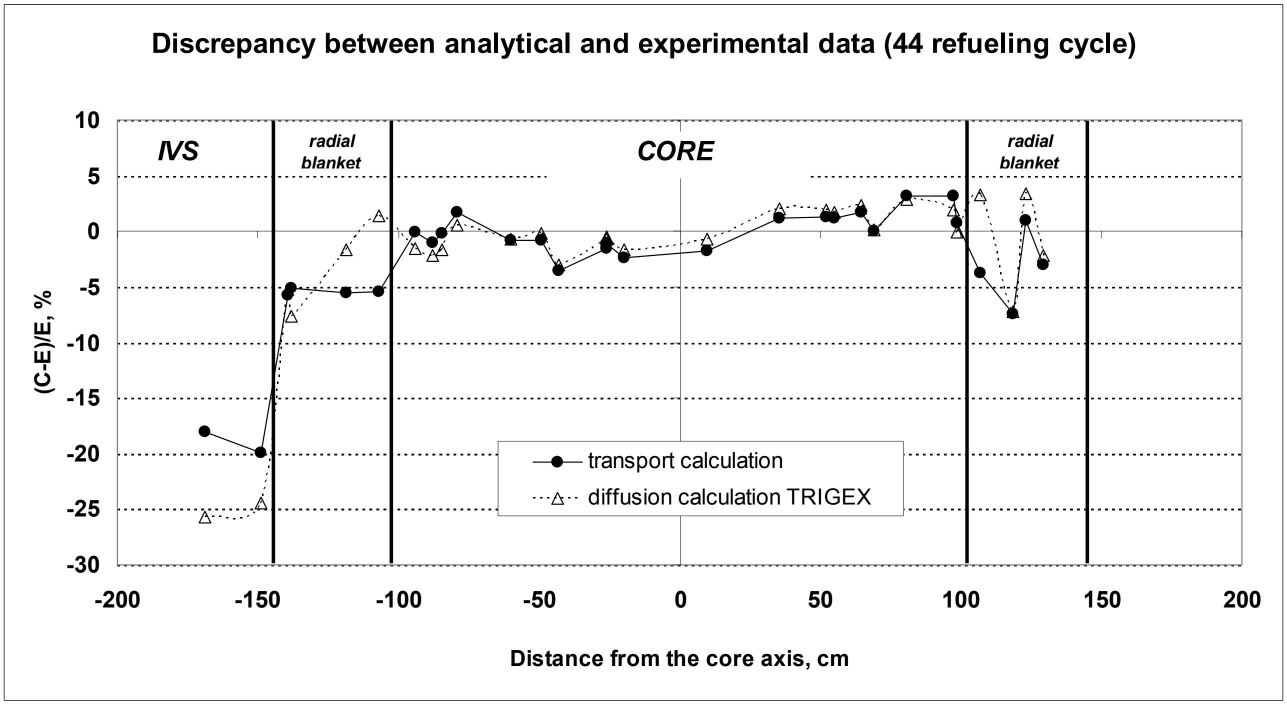
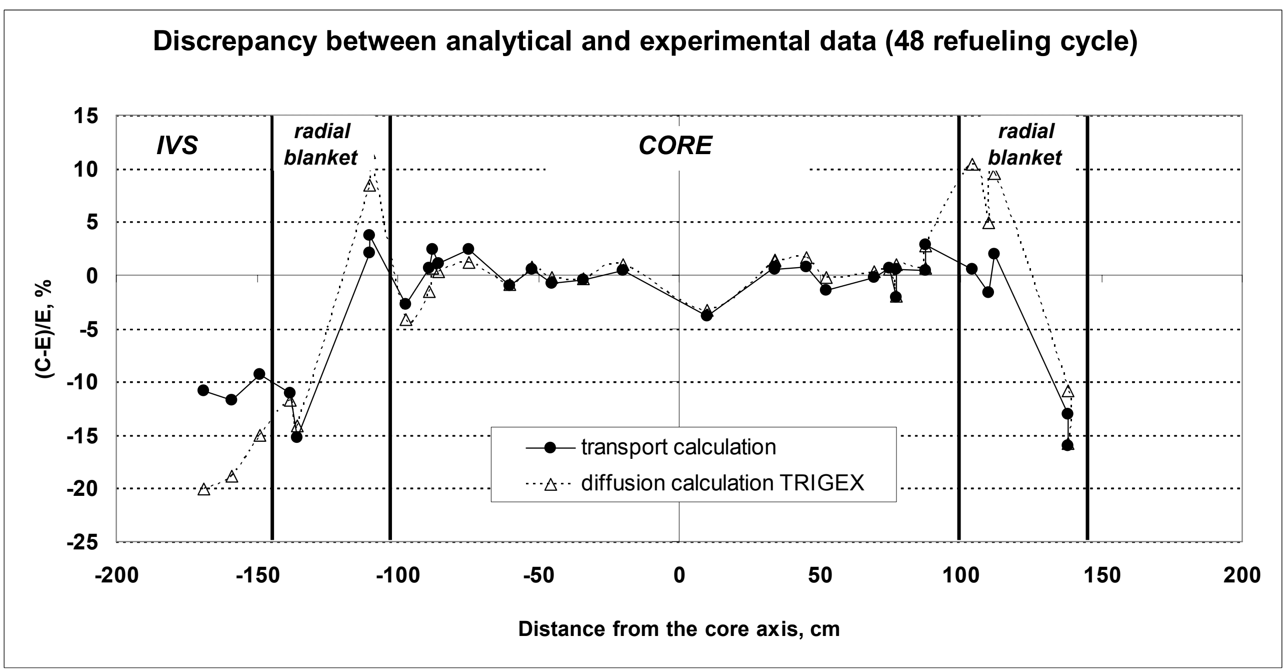


Fig. 4. Discrepancy between calculated results and experimental data obtained after 48th refueling cycle of the BN-600 reactor



- for the core SA, max discrepancy does not exceed 5%, and root-mean-square discrepancy is at the level of ~1.5%, this value corresponding to the accuracy of the experiment;

- for IRB SA, experimental data are in a good agreement with analytical results obtained by GEFEST and JARFR codes (discrepancy within ±7%). Results obtained by TRIGEX code are also within that range after introduction of transport correction in MMKKENO code;
- for SA of ERB and IVS, max discrepancies are, respectively, equal to 10-15% and ~20%. For IVS SA, it could be caused to some extent by the errors of diffusion approximation, and if these errors are taken into account, then the discrepancy for IVS SA decreases down to ~10-15%;
- in the experiment performed in 2005, there was clear indication of the difference between calculated and measured gradient of neutron field distribution in the radial blanket that was not caused by the error of diffusion approximation, so additional analysis was required;
- in the experiment of 2003, many controlled SAs of radial blanket were influenced by EOU hydrogen-containing heterogeneous irradiation (trap-type) devices provided for Co-60 production. This might cause the difference in the nature of discrepancies between results of the experiments in the radial blanket obtained in 2003 and 2005;
- core updating has not resulted in the additional uncertainties in the calculation of power rate distribution among SAs of the BN-600 reactor. Thus, experimental data confirmed design power profile in the reactor SAs to be within the above limits.

CONCLUSION

Analysis of experience gained in applying gamma-scanning method showed its full applicability, convenience and reliability under conditions of NPP with fast reactor. Accuracy of this method is 3-6% (for 95% level) for various SAs.

26-group diffusion approximation and GEFEST, JARFR and TRIGEX codes using ABBN-93 nuclear data give adequate description of power rate distribution among the SAs of the BN-600 reactor core. БН-600. Conservative estimation of calculation error is 5%. The main concern is evaluation of power profile of peripheral areas of the radial blanket and in vessel storage, if achieved accuracy of 10-15% is insufficient.

It should be noted also that all results of this work refer to the average over SA cross section characteristics. Analisys of local values require additional studies and estimation.

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