EPJ Web of Conferences **106**, 02008 (2016)
DOI: 10.1051/epjconf/201610602008

Methodology of Fuel Burn Up Fitting in VVER-1000 Reactor Core by Using New Ex-Vessel Neutron Dosimetry and In-Core Measurements and its Application for Routine Reactor Pressure Vessel Fluence Calculations

Pavel Borodkin[a], Gennady Borodkin, and Nikolay Khrennikov

Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS), 107140 Moscow, Russia

Abstract. Paper describes the new approach of fitting axial fuel burn-up patterns in peripheral fuel assemblies of VVER-1000 type reactors, on the base of ex-core neutron leakage measurements, neutron-physical calculations and in-core SPND measured data. The developed approach uses results of new ex-vessel measurements on different power units through different reactor cycles and their uncertainties to clear the influence of a fitted fuel burn-up profile to the RPV neutron fluence calculations. The new methodology may be recommended to be included in the routine fluence calculations used in RPV lifetime management and may be taken into account during VVER-1000 core burn-up pattern correction.

1. Introduction

To perform the correct VVER reactor pressure vessel (RPV) neutron fluence calculations a reliable neutron source in the reactor core should be used as input data, as it is recommended in the recently developed Russian utility procedure [1]. During analysis of VVER-1000 ex-vessel neutron-activation experimental data derived by SEC NRS it was shown that calculated estimation of axial fuel burn-up distributions, especially in peripheral part of the core, did not correlate to axial ex-vessel fluence rate distributions [2]. Semi-empirical neutron sources prepared from in-core self-power neutron detector (SPND) measurement data have allowed reach a better coherence (up to 10% in upper core part) with experimental data for all the investigated VVER-1000 units. Solving the reverse task of alternative estimation of power output (fuel burn-up) distributions, it was suggested to adjust the core neutron source distribution on the base of the neutron leakage measurements, core calculations and in-core SPD measured data.

The developed approach of fitting fuel burn-up patterns of peripheral fuel assemblies of VVER-1000 type reactors uses results of the new ex-vessel measurements and their uncertainties. Especially, during ex-vessel measurements on Rostov-2, Balakovo-1 and 2 power units, where modified types of fuel assemblies (TVS-2M) with extended fuel height are used, and on Novovoronezh-5, where core loadings with reduced neutron leakage are implemented and were investigated for long

[a] Corresponding author: pborodkin@secnrs.ru

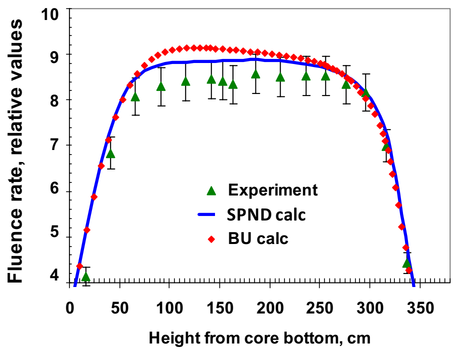

Figure 1. Comparison of calculated (two methods) and measured fluence rates on azimuth of 55 degrees.

term operation, the influence of the ex-vessel fluence distributions on the axial burn-up patterns of core periphery was analysed. Applying to these measurements, the deterministic neutron transport calculations (3D synthesis by DORT code coupled with the BUGLE-96T library [3]) were performed. The results of the new extended ex-vessel measurements allow to estimate more accurate fuel burn-up shifts in separate fuel assemblies by using the developed methodology.

2. Routine Neutron Fluence Calculations

Recently developed Russian utility procedure [1] established the main requirements for the RPV neutron fluence calculation procedure. In accordance to [1] a reliable neutron source in the reactor core should be used for the routine neutron fluence calculations. The input data for the preparation of time-integrated neutron source distributions are drawn from the results of the neutron-physical calculations performed for the project and actual reactor core loadings. The time-depended neutron source characterized by rate and density of fissions determines power distribution in the core volume, and the time-integrated neutron source, considered as number of fissions, results in fuel burn-up. The fuel burn-up distribution, especially in peripheral parts of the core, correlates with time-integrated neutron leakage (neutron fluence) and its distribution in ex-core region.

Such principle is used for routine fluence calculation procedure by discrete ordinates two-dimensional code DORT with the BUGLE-96T library [3] (47 neutron groups). Input data for the preparation of time-integrated neutron source distributions (used for DORT calculations) are usually based on a shift of the fuel burn-up distributed over the fuel assemblies for one operating cycle. The neutron source associated with the distribution of the fuel burn-up shift in the fuel assemblies corresponds to power output distribution over reactor core during operating cycle. Ex-vessel measurements are used for validation of the calculated fluence rates.

During analysis of VVER-1000 ex-vessel neutron-activation experimental data derived by SEC NRS it was shown that the calculated estimation of axial fuel burn-up distributions, especially in peripheral part of core, did not correlate to axial ex-vessel fluence rate distributions [2]. Time-to-time steady-state core power distributions measurements from the in-core monitoring system were used for alternative analysis of neutron source distribution. The source distribution, on the basis of SPND data for (r,z) geometry of DORT model, was calculated the same manner as for the standard procedure, where instead of fuel burn-up distributions, the integral power distributions obtained on SPND were used. Ex-vessel fast neutron FRs were calculated using SPND-sources in case of several VVER-1000 reactors. The calculated axial FR distributions were compared with experimental data (Figs. 1, 2 – for Balakovo-3 power unit).

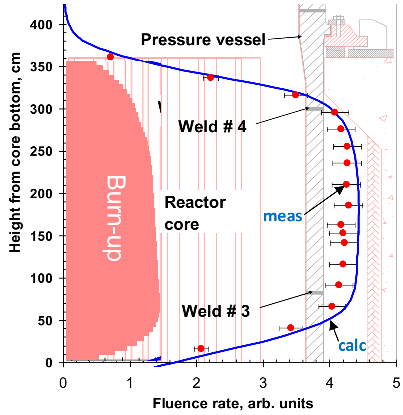

Figure 2. Comparison of corrected fluence rate calculation, ex-vessel measurements and initial calculated BU distribution.

The shape of the calculated distribution has considerably become close to experimental one. Semi-empirical neutron sources prepared from in-core SPND measurement data allowed reach a better coherence (up to 10% in upper core part) with experimental data in case of all investigated VVER-1000 units, that was shown in [2].

A new approach for the adjustment (fitting) of the time-integrated neutron source distributions, and, hence, fuel burn-up patterns in peripheral fuel assemblies, is proposed on the base of the ex-core neutron leakage measurements, neutron-physical calculations and in-core SPND measured data.

One notes that the Russian utility procedure [1] also includes the recommendation to use SPND based neutron source for routing fluence calculations.

3. Methodology of Fuel Burn up Fitting in VVER-1000 Reactor Core

Taking into account that fuel burn-up distributions in operating VVER-1000 may be now evaluated by analytical methods (calculations) only, it is needed to find the way for validation of the calculational evaluations. Let's return to the analysis of a neutron source in the reactor core. If there are some results of ex-vessel (especially, height distributed) neutron activation measurements, it is possible to solve a inverse problem, i.e. using results of such measurements it is possible to estimate (adjust) neutron-source distributions (and hence, burn-up distributions) in a reactor core. These distributions could be compared with results of neutron-physical calculations to prove last. The methodology of neutron-source distributions (and hence, burn-up distributions) fitting are close to the other methods of the adjustment of many parametric values (i.e. neutron spectra from the measured reaction rates).

BU distributions in fuel assembly from the standard neutron physical calculations may be used as initial BU function (BUcalc). Measurements of in-core power distributions based on SPND data is used as correction factor (SPND meas). Preparation of the average source distribution, which is used for neutron transport calculations, is the final point of the adjustment based on correction of initial BU distributions. Criterion of iteration process is the good coherence of the calculated (C) and the measured (M) ex-vessel neutron leakage distributions. The main points to achieve this methodology of fuel burn up fitting are:

a) reliable and sufficient measurements;
b) improved neutron transport calculations models;
c) procedure of core neutron source preparation;

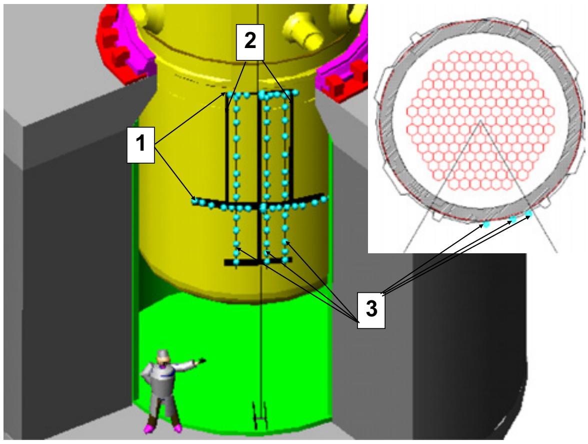

Figure 3. New ex-vessel installation of NADs on VVER-1000 reactor and axial chain positions for 60 degrees sector: 1– azimuthal rods with NAD; 2 – support racks of the construction; 3 – three chains with NAD.

d) method of the adjustment (fitting) which should have:
   1) initial and corrected value of the fitted parameter;
   2) criterion of iteration process.

3.1 Ex-vessel Measurements

Many vertical and azimuthal racks with neutron-activation dosimeters (NAD) can be placed in ex-vessel cavity to receive a detail structure of FR distributions (up to continuous shape if using a wire). Thus, the placement of several vertical racks in different azimuth directions allows to estimate accurately fuel burn-up shifts in separate fuel assembly by means of use of weight functions for each peripheral assembly and of superposition of assembly contributions to ex-vessel FR distributions. The new procedure of ex-vessel neutron activation measurements were developed and applied on the some VVER-1000 reactors power units.

Figure 3 illustrates the dosimeters rack mounting.

The rack is strictly blocked on the three axes (radius, height and azimuth). A crosswise rack consists from one vertical rod, two azimuthal rods and two support racks. Support racks were done to fix horizon of two azimuthal rods. The upper azimuthal rod gives allowance to hang on three or more chains with NADs.

Neutron-activation detectors (dosimeters) set includes ^{54}Fe, ^{58}Ni, ^{46}Ti, ^{63}Cu, ^{93}Nb, ^{237}Np and ^{238}U isotope foils. The analyzed threshold dosimetry reactions, (n,p), (n,α), (n,n') and (n,f), give information about fast neutron spectra and allow to evaluate fast neutron fluence and FR. To derive reaction rate (RR) values (or fluence rate) from activity measured data, decay factors depending from reaction products half-life, duration of irradiations (operating cycle period) and local flux variations were taken into account.

From the measured results we can have the experimental value of the FR which should be determined in ex-vessel cavity for different angles θ and different heights Z:$-\Phi_{meas}(\theta, Z)$.

3.2 Neutron Transport Calculations

Three-dimensional (3D) synthesis method based on results of DORT calculations was used for neutron fluence rate evaluations. Analytical justification of 3D synthesis method by comparison with precise

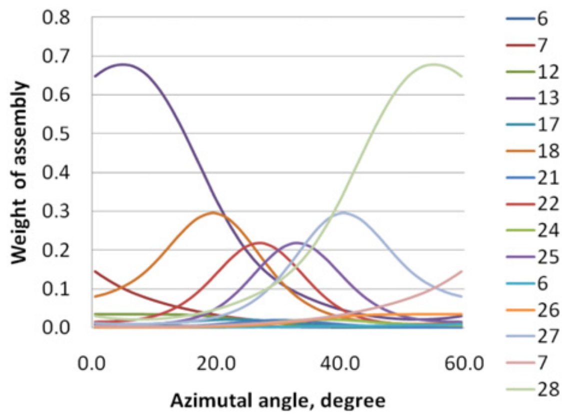

Figure 4. "Weights" of fuel assemblies calculated by DORT for different azimuthal angles in ex-vessel cavity.

3D calculations based on Monte Carlo codes was carried out many times. Calculational results are values of FR in neutron energy groups. Integrated over the fast neutron energy groups FRs are used in the present analysis. Relative space FR distributions correspond completely to fluence distributions. The calculated results correspond also to the (r,z) DORT mesh which use in the model preparation. For the comparison with measured results the calculated values were obtained in the same coordinates of the ex-vessel cavity, where the NADs were placed: different angles θ and different heights Z. The calculated value of FR could be written as follow: $-\Phi(\theta, Z)$.

3.3 Procedure of Core Neutron Source Preparation

An axial neutron source distribution in (r,z) DORT model is prepared from partial axial source distributions of each fuel assemblies. It was shown that the fluence on the RPV of VVER-1000 is influenced by only the first or second peripheral rows of fuel assemblies. The input data for preparation of time-integrated neutron source distributions is usually based on a shift of fuel burn-up distributed over each fuel assembly (i) during an operating cycle. The fuel burn-up is determined from the neutron-physical calculations for each row for the used mesh (Z). The shift of the fuel burn-up for i assembly on each Z layer could be written as follow: $\Delta\rho_i(z) = \rho_i^{end}(z) - \rho_i^{begin}(z)$.

To prepare azimuthally averaged axial source distribution in (r,z) model, partial assemblies distributions are used with "weighting" contributions to NADs positions. The "weight" of each fuel assembly was calculated previously by separate DORT calculations. "Weights" for two peripheral rows of fuel assemblies (i) were calculated for different angles in ex-vessel positions. The "weighting" function for different azimutal angles θ in the ex-vessel cavity is shown on Fig. 4. The example of the weighting function in first peripheral assembly row for the one azimuthal direction is shown on Fig. 5.

From the other way the axial neutron source distribution in (r,z) DORT model allows to make the neutron source distribution for each cell in Z mesh. Each cell from the source distribution depending of the coordinate of Z mesh also gives its own influence to the ex-vessel FR distribution. From the (r,z) separate calculation we could have the "weighting" function of each layer (h) in source distribution for different height Z in ex-vessel cavity. From the neutron transport calculations we have two functions of weights, which we can use for the further fitting procedure:
$w_i(\theta)$– "weight" function for different azimuthal angles θ for each fuel assembly i;
$w(h, z)$– "weight" function for different axial cell Z for each height source layer h (h and z are the same function from the Z mesh used in the DORT model).

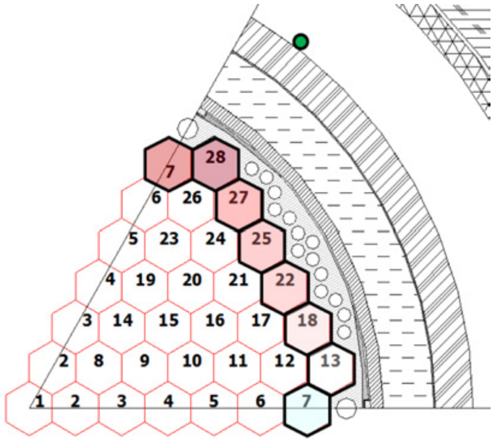

Figure 5. Example of weight distribution for first row and for one azimuthal direction.

3.4 Method of the Adjustment (Fitting)

We assumed that each peripheral fuel assembly (i) give its own response to fluence rate axial distribution on the different angle θ position. The FR from the i fuel assembly to the n azimuthal angle θ could be expressed as follows:

$$\varphi_i(\theta) = \int_0^{Hcore} \Delta\rho_i(z) * w_i(\theta) dz \begin{vmatrix} w(h_1, z_1) \ldots\ldots\ldots w(h_n, z_1) \\ \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots \\ w(h_1, z_n) \ldots\ldots\ldots w(h_n, z_n) \end{vmatrix}; \quad (1)$$

If we have a different number of fluence rate axial distribution, the dependence of each assembly response (for n – assemblies) to the different (k) azimuthal positions could be expressed by a system of linear equations:

$$\begin{bmatrix} \varphi_i(z_1, \theta_0) \\ \cdots \\ \varphi_n(z_H, \theta_0) \end{bmatrix} \ldots\ldots\ldots \begin{bmatrix} \varphi_i(z_1, \theta_k) \\ \cdots \\ \varphi_n(z_H, \theta_k) \end{bmatrix}; \quad (2)$$

Also the same system could be imagined for each azimuthal distribution for different (l) height positions:

$$\begin{bmatrix} \varphi_i(z_0, \theta_0) \\ \cdots \\ \varphi_n(z_0, \theta_{60}) \end{bmatrix} \ldots\ldots\ldots \begin{bmatrix} \varphi_i(z_l, \theta_0) \\ \cdots \\ \varphi_n(z_l, \theta_{60}) \end{bmatrix}. \quad (3)$$

The full calculated fluence rate distribution for one separate azimuthal angle θ_k (or separate height Z_l) with the response of each fuel assembly could be expressed as follows:

$$\Phi(\theta_k) = \sum_i \varphi_i(\theta_k) \text{ for each axial distribution (k);}$$

$$\Phi(z_l) = \sum_i \varphi_i(z_l) \text{ for each azimuthal distribution (l).}$$

Table 1. NPP units with VVER-1000 where analyses were performed.

NPP	Unit	Reactor type	Cycle, year of measurements	Teff, days	Core loading
Balakovo	№3	VVER-1000/320	18 2009–2010	451	TVS-2M with blankets (104% Nnom)
Balakovo	№4	VVER-1000/320	15 2009–2010	420	TVS-2M with blankets (104% Nnom)
Kalinin	№1	VVER-1000/338	28 2012–1014	360	Loading with TVSA и TVSA-PLUS
Novovoronezh	№5	VVER-1000/187	29 2012–1013	307	Standard core loading (project 187) burn up assemblies
Rostov	№ 2	VVER-1000/320	2 2011–2012	305,7	Full core loading with TVS-2M (core height 370)

The fitting parameters $\Delta\rho_i(z)$ are determined with non-linear least squares fit to the accurate value. The fitting was adjusted to minimize χ^2 values, defined as:

$$\chi^2 = \sum_l \sum_k \left(\frac{\Phi_{meas}(\theta, Z) - \Phi(\theta, Z)}{\varepsilon} \right)^2; \tag{4}$$

where ε is experimental error from neutron-activation measurements.

For initial fitting parameter we can use the calculated shifts of fuel burn up for each fuel assembly. But as it was shown the calculated value of fuel burn up shifts did not correlate to ex-vessel fluence rate distributions. The systematic discrepancies (more than 25%) of axial calculational and experimental FR distributions in upper part of core level were observed. The iteration process to minimize $\chi 2$ values could be very long, and consist of a lot of iterations for each fuel assembly. But from last investigation it was shown that preparation of core neutron source could be done by using in-core measurements coupled with neutron-physical core calculations. So if for the second iteration we use the assembly power distribution from the in-core monitoring system, the iteration process would be shorter and the criterion of iteration could be closer.

4. Application to the Routine Fluence Calculations

Experimental investigations using new technique were carried out on NNP power units with VVER-1000 type reactors. In the paper, experimental data obtained on VVER-1000 of different NPP units during different fuel cycles are analyzed. The information of power units where such technique were testified and characteristic of fuel cycles are presented in the Table 1.

During ex-vessel measurements on the Rostov-2, Balakovo-3 and 4, Kalinin-1 power units, where modified types of fuel assemblies (TVS-2M, TVSA-PLUS) with extended fuel height are used, and on Novovoronezh-5, which has reduced VVER-1000 core loadings were investigated.

For the test of the new technique the measurements were performed for such conditions of operated power units:

– power: 100% and 104% from nominal;
– reactor types: VVER-1000/320, VVER-1000/338, VVER-1000/187;
– reactor cores: full core for 320 (338) project – 163 fuel assemblies, full core for 187 project: - 151 fuel assemblies;
– fuel assemblies types: TVS-2, TVS-2M (with and without blanket), TVSA, TVSA-PLUS, TVS-187;
– fuel assemblies on periphery of the reactor core: fresh, burn-up, deep burn-up.

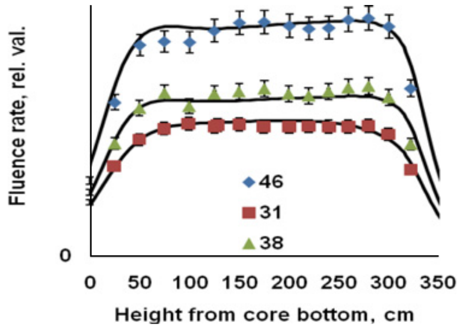

Figure 6. Comparison of FR axial distributions for Balakovo 4 power unit for different azimuthal positions.

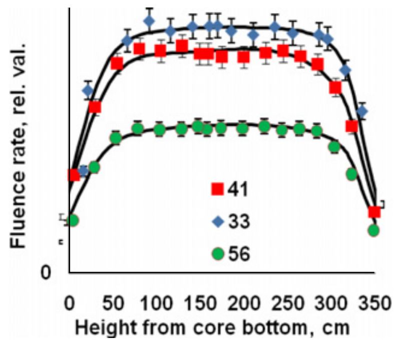

Figure 7. Comparison of FR axial distributions for NV-5 power unit for different azimuthal positions.

Calculational-experimental research was performed for all types of VVER-1000 which are under operation and for all kinds of operation conditions, which could be reflected to the RPV fast neutron fluence distribution. The main focus was give to axial FR distributions in different azimuthal directions, and to the core neutron source preparation, using the new methodology with fitting burn up shifts by using ex-vessel and in-core measurements. The new methodology was applied also to performed experimental results. The iteration process has been applied for the monitor reaction of ^{54}Fe(n,p) (threshold 3.0 Mev) which have ε, approximately, 5–7% (which were proposed as numerical convergence criterion). Figures 6, 7 demonstrate the comparison of calculated (with using of fitted burn-up shifts) and measured fluence rates.

The analog basic trend of the ex-vessel distributions was appeared in application of other measured reaction rates which focused on fast neutron spectrum. The comparison of measured activities shows good agreement of the three distributions (whole fast neutron region) (Fig. 8). It means that spectral index ^{93}Nb(n,n')RR/^{54}Fe(n,p)RR/^{63}Cu(n,α)RR (or FR(> 1 MeV)/(FR(>3 MeV)/FR(> 6 MeV)) is non height-dependent value for all core height levels. The Fig. 9 demonstrates the more detail ex-vessel measured height distribution that could be achieved by using the new technique for local effects analyzes in the fluence distribution for VVER-1000 RPV.

New estimation of ex-vessel fluence rate distributions shows some new features of the fast neutron fluence distributions: maximum value of fast neutron fluence on VVER-1000 RPV could be in the upper part of the reactor core; fast neutron fluence value for the weld # 4, 5 and reactor support structures could be underestimated during standard calculations.

These facts should be taken into account during VVER-1000 RPV neutron fluence assessments and their use in RPV life-time management.

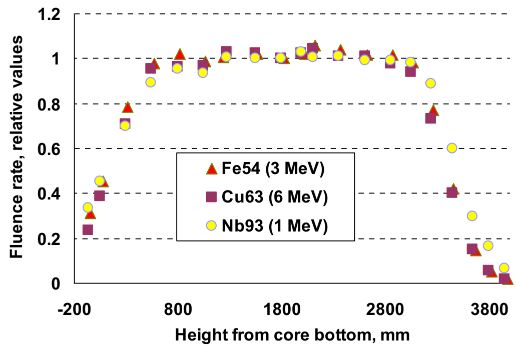

Figure 8. Comparison of relative values of measured FR for different reaction in fast neutron region for ex-vessel positions.

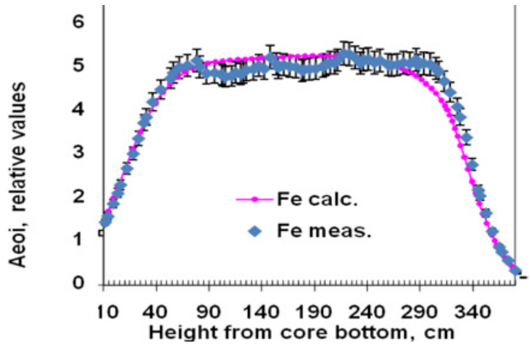

Figure 9. Comparison of FR with measured detail axial distributions by using new technique for local effects analyzes.

5. Conclusion

The approach of fitting axial fuel burn-up patterns in peripheral fuel assemblies of VVER-1000 type reactors in application with the results of new ex-vessel measurements, their uncertainties and influence of fitted fuel burn-up to the RPV fluence calculations has been developed. Especially, during ex-vessel measurements on the Rostov-2, Balakovo-3 and 4, Kalinin-1 power units, where modified types of fuel assemblies (TVS-2M, TVSA-PLUS) with extended fuel height are used, and on Novovoronezh-5, where reduced neutron leakage were investigated, the influence of the ex-vessel fluence distributions on the height burn-up patterns of core periphery was analyzed. Applying to these measurements, the deterministic neutron field parameters calculations (3D synthesis by DORT code coupled with the BUGLE-96T library) were performed.

The results of the new extended ex-vessel measurements allow to estimate more accurately axial fuel burn-up shifts in separate fuel assemblies by using the developed methodology.

References

[1] "The procedure for accounting and prediction of RPV fast neutron fluence for VVER type reactors (RD EO 1.1.2.29.0913-2012)", Concern "Rosenergoatom", 2012

[2] P. Borodkin, G. Borodkin, N. Khrennikov "Application of Ex-vessel Neutron Dosimetry Combined with In-core Measurements for Correction of Neutron Source Used for RPV Fluence

Calculations", 14th International Symposium on Reactor Dosimetry, Bretton Woods, New Hampshire, 22-27 May, 2011

[3] G. Borodkin and B. Boehmer, "Validation of 3D Synthesis RPV Neutron Fluence Calculations using VVER-1000 Ex-Vessel Reference Dosimetry Results", 2000 ANS Annual Meeting, June 4–8, 2000, San Diego, California, Transactions of the ANS, volume 82, 2000, pp. 223–225