STUDIES FOR IMPROVEMENT OF VVER-440 NEUTRON FLUENCE DETERMINATION

K. Ilieva, S. Belousov, T. Apostolov,

INRNE-BAS

Abstract

For assessment of radiation embrittlement and prediction of reactor vessel lifetime with reasonable conservatism a "best estimated" neutron fluence is necessary. New studies purposed to improve the fluence determination are presented: 1) study on the reliability of multigroup presentation of the neutron cross sections, and 2) impact of negative gradient of reactor power in the periphery assemblies on the neutron fluence evaluation. The results of these studies are base for improvement of neutron fluence determination methodology applied by the INRNE, BAS at Kozloduy NPP.

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1. Introduction

The assessment of radiation embrittlement and prediction of reactor pressure vessel (RPV) lifetime with reasonable conservatism has to be based on the "best estimated" neutron fluence. Under "best estimated" fluence we understand that estimation in which the accessible for evaluation parameters of influence as well as sources of uncertainty are accounted. The more precise determination of neutron fluence is a base for reduction of the conservatism in neutron fluence evaluation.

Comparisons of measured and calculated responses in neutron fluence verification tasks have shown unsatisfied consistency of the data [1-4]. The coefficient of attenuation of neutron fluxes through the vessel was determined for the VVER-440 Mockups: Mockup1 - standard loading in direction of maximum exposure; Mockup2 - dummy cassettes' loading in direction of minimum exposure; Mockup3 - dummy cassettes' loading in direction of maximal exposure [1-2]. The comparison of the calculated and experimental attenuation, through the RPV, coefficients of integral neutron fluxes with low energy limits of 1, 2 and 3 MeV shows that the calculated attenuation underestimates the experimental one by a value that varies from 2% to 17%. The measurements carried out by Cu, Fe, Nb activation detectors irradiated behind the VVER-440 reactor vessel at Units 1-4, KNPP [3,4] showed consistency within 20% of measured to calculated results.

One possible reason for the above mentioned discrepancy was addressed to the multigroup presentation of neutron cross sections' energy dependence in neutron transport calculation. In particular, it was suggested that the multigroup presentation does not describe adequately the neutron interaction with iron nuclei for the resonance energy range [2].

Another reason for discrepancy could be the negative gradient of reactor power in the periphery assemblies that is not accounted in routine neutron fluence calculation [4]. The radial gradient of the power in the periphery assemblies of reactor core has an important influence on the flux and shape of energy spectrum of neutrons, which go out from reactor core.

The studies presented below have been purposed to assess the influence of multigroup approximation of neutron cross sections' energy dependence on neutron flux results. The impact of reactor power gradient on the neutron fluence value is quantitatively analyzed too.

2. Calculations

2.1. Multigroup and continuous neutron cross sections

The calculations have been performed for VVER-440 type of reactor with dummy cassettes loading scheme (36 dummy cassettes in the periphery of reactor core) as it is applied on the Units 1-3 of Kozloduy NPP. Comparative calculations have been performed in multigroup approximation and continuous presentation of the energy dependence of neutron cross-sections, both based on ENDF/B-VI file of estimated neutron data. Two methods: 1) DO - discrete ordinates' method by applying the DORT code [5] and problem oriented neutron multigroup cross-sections library BGL440 [6], and 2) MC –Monte Carlo method by MCNP code [7] and continuous energy dependence of neutron cross-sections in the library DLC189, have been used. The calculations have been performed in two-dimension geometry with uniform distributed neutron source (Fig. 1). The relative error at the 1σ level of MCNP calculation has been less than 1 % for the flux and 2% for the activity.

The azimuth distribution of the integral neutron fluxes with low energy limit of 0.5, 1, and 3 MeV as well as saturated ^{54}Mn activity of iron detector have been calculated for the positions of interest: behind the barrel, onto RPV, at 1/4, 2/4 and 3/4 thickness in RPV, and behind the vessel. The iron detector is one of applied as NPP ex-vessel detectors for fluence verification.

2.2. Reactor power negative gradient impact

The calculations of space power distribution and fuel burnup for both, assembly-wise and pin-wise presentation of reactor core were performed for dummy cassettes' loading applied for 21th cycle of Unit 1. The assembly wise calculation was carried out in three dimension geometry by the PYTHIA diffusion code [8]. The pin-wise distribution was obtained by DERAB code [9]. The libraries of effective cross sections needed for these calculations were prepared with NESSEL-4 code [10].

Calculations of neutron flux and spectrum with energy above 0.5 MeV, as well as of niobium, iron and copper ex-vessel detectors' activities, from the reactions ^{93}Nb(n,n')^{93m}Nb, ^{54}Fe(n,p)^{54}Mn and ^{63}Cu(n,α)^{60}Co respectively, were performed in assembly-wise and pin-wise distribution of neutron source.

Three dimension neutron flux and energy spectrum calculation was performed by ASYNT code [11] using three dimension adjoint solution synthesized from two -dimension **(r,t)** and **(r,z)** , and one-dimension **(r)** solution of the adjoint kinetic equation of neutron transport. Each one of them was determined by discrete ordinate code DORT. The BGL440 was applied in these calculations. The macro cross sections in P3 approximation for given geometry and material composition was generated by GIP code [5]. The data for volume neutron source appropriate for ASYNT calculation were prepared from diffusion codes' output files by the interface software package DOSRC [12]. The activity of neutron detectors were determined by IRDF-90 data [13].

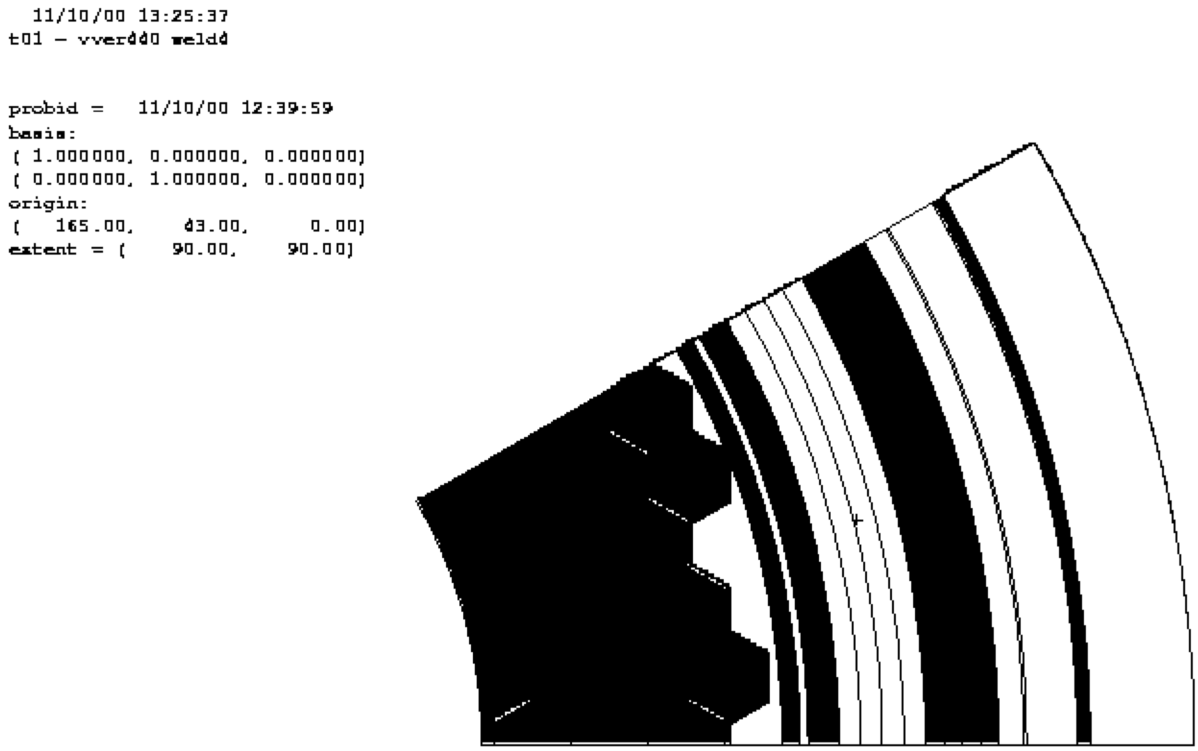

Fig. 1. Two dimension geometry model of VVER-440 with dummy cassettes loading

Table 1. Group structure of BGL440 [6] for energy region above 0.5 MeV

Group	Low-limit	Up-limit
1	14.1910	17.3320
2	12.2140	14.1910
3	10.0000	12.2140
4	8.60710	10.0000
5	7.40820	8.60710
6	6.06530	7.40820
7	4.96590	6.06530
8	3.67880	4.96590
9	3.01190	3.67880
10	2.72530	3.01190
11	2.46600	2.72530
12	2.36530	2.46600
13	2.34570	2.36530
14	2.23130	2.34570
15	1.92050	2.23130
16	1.65300	1.92050
17	1.35340	1.65300
18	1.00260	1.35340
19	.820850	1.00260
20	.742740	.820850
21	.608100	.742740
22	.497870	.608100

3.Results

3.1. Multigroup and continuous neutron cross sections

The comparison between both, DO and MC, applied methods is presented by the differences of responses' values (DO/MC-1)% in Table 2. The "mean" value is the azimuth mean value averaged over the 30-degree sector of mirror symmetry. The "max" value corresponds to the maximum exposure in azimuth direction, 13 degree.

Table 2. Comparison of DO and MC results, VVER-440, dummy cassette's' loading

Position	Type of response	(DO/MC-1)% mean	(DO/MC-1)% max
Behind Barrel	Flux E>0.5 MeV	+1.88	+1.04
	Flux E>1.0 MeV	+1.50	+1.20
	Flux E>3.0 MeV	-3.06	-4.21
	^{54}Mn-activity	-1.70	-3.97
Onto RPV	Flux E>0.5 MeV	-6.42	-7.87
	Flux E>1.0 MeV	-6.70	-9.51
	Flux E>3.0 MeV	-7.53	-9.34
	^{54}Mn-activity	-6.12	-8.77
1/4T	Flux E>0.5 MeV	-3.76	-4.96
	Flux E>1.0 MeV	-5.34	-8.24
	Flux E>3.0 MeV	-7.60	-9.51
	^{54}Mn-activity	-6.18	-9.88
2/4T	Flux E>0.5 MeV	-3.07	-5.73
	Flux E>1.0 MeV	-5.60	-8.45
	Flux E>3.0 MeV	-8.05	-9.55
	^{54}Mn-activity	-6.68	-12.0
3/4T	Flux E>0.5 MeV	-3.05	-5.32
	Flux E>1.0 MeV	-6.11	-8.17
	Flux E>3.0 MeV	-8.44	-10.9
	^{54}Mn-activity	-6.45	-9.99
Behind RPV	Flux E>0.5 MeV	-3.45	-5.57
	Flux E>1.0 MeV	-7.02	-8.69
	Flux E>3.0 MeV	-7.95	-9.56
	^{54}Mn-activity	-5.82	-9.90

The comparison between DO and MC flux values shows that:

-There is a good consistency (within 3%) of the two methods relevant to the absolute flux behind the barrel. Although behind the barrel the differences between DO and MC results practically are not significant there is a tendency in its increasing for the higher energy region. This tendency shows that there are more fast neutrons in the MC spectrum than in DO one.

- The difference between DO and MC results arises after the transmission of neutrons through the water in downcomer. On the RPV in direction of maximum exposure the MC value of neutron flux with energy above 0.5 MeV and ^{54}Mn activity is about 8% greater than the DO value (see Fig. 2.1-2).

- The difference between DO and MC results for neutron flux attenuation through the RPV is not significant (Fig. 3.1-2). This shows that for neutron transport calculation in the RPV steel, the multigroup neutron cross sections of iron lead to the same results as the continuos energy dependence presentation of the neutron cross sections.

- The MC value of ^{54}Mn activity behind the RPV is about 10% greater than the DO value.

For better understanding of MC and DO discrepancies (significant in downcomer) additional calculation was performed for one-dimensional model of the task. In this model the reactor core was presented as a cylinder, the dummy cassettes and the baffle as cylindrical shells with the same cross section square as that in the two dimensional model. The application of this one-dimensional model allowed to avoid any discrepancies aroused from the different geometry description in the two-dimensional DO and MC calculations. The attenuation values obtained by DO and MC for one-dimensional calculation were the same as those for two-dimensional calculation. This means that the two-dimensional discrete geometry modeling in DORT is good enough compared to the exact geometry representation in MCNP.

3.2. Reactor power negative gradient impact

The more detailed pin-wise calculation of reactor core permits to obtain the neutron source negative gradient in the core periphery. The mean power value (assembly-wise value) of a periphery assembly is of about 50-54% greater than the minimal pin power value within the same assembly. The accounting of this source gradient leads to decreasing of neutron flux/fluence onto the vessel, which is about 10% and varies slowly on the azimuth direction. On other hand it leads to relatively increasing of the portion of more fast neutrons in the energy spectrum.

The comparison of the results is presented by the ratio Ass/Pin of flux/activity values calculated by assembly-wise source to those calculated by pin-wise source on Fig. 4. The ratio values for the flux onto the vessel, in 1/4 depth of reactor vessel (1/4T), as well as the activity of niobium (Nb93), iron (Mn54) and copper (Co60) detectors behind the vessel are shown. The azimuth interval corresponds to 30 degree sector of mirror symmetry of reactor core loading . The lower limit of integral neutron flux is 0.5 MeV.

The azimuth interval corresponds to 30-degree sector of mirror symmetry of reactor core loading. The lower limit of integral neutron flux is 0.5 MeV. The different along the azimuth flux/activity decreasing is related as to the lower source from the reactor core periphery so to the hardness of the energy spectrum. Comparison of neutron spectra onto RPV in azimuth direction of maximum exposure obtained by both, pin-wise and assembly-wise source, is presented in Fig. 5.

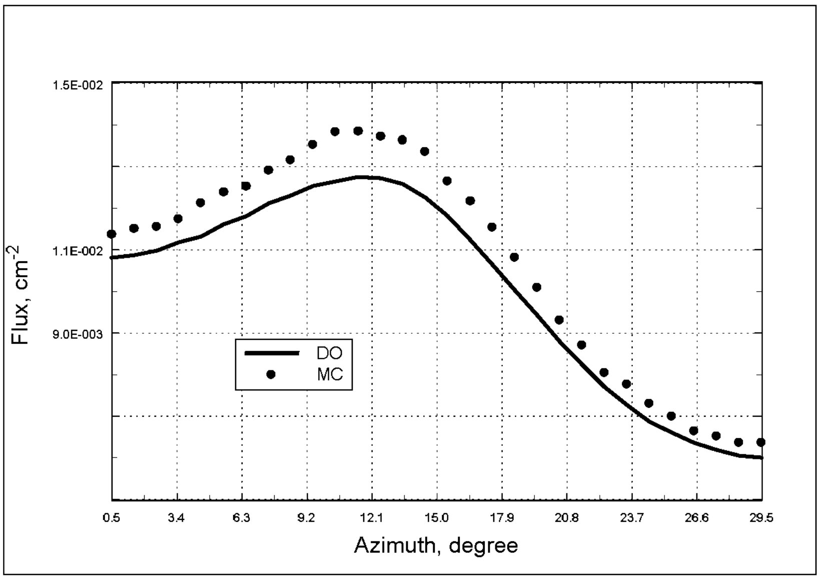

Fig. 2.1. Flux, E> 0.5 MeV, azimuth distribution on RPV

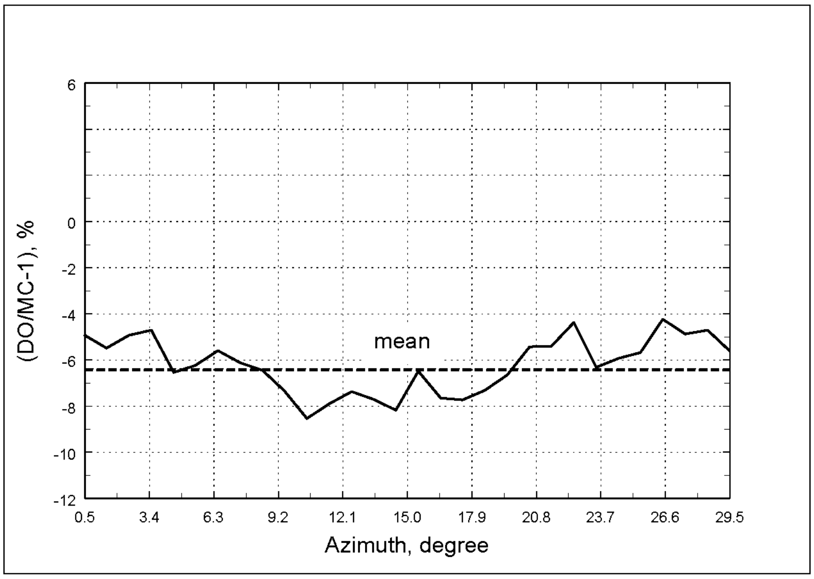

Fig. 2.2. Difference between DO and MC results for flux, E> 0.5 MeV, azimuth distribution on RPV

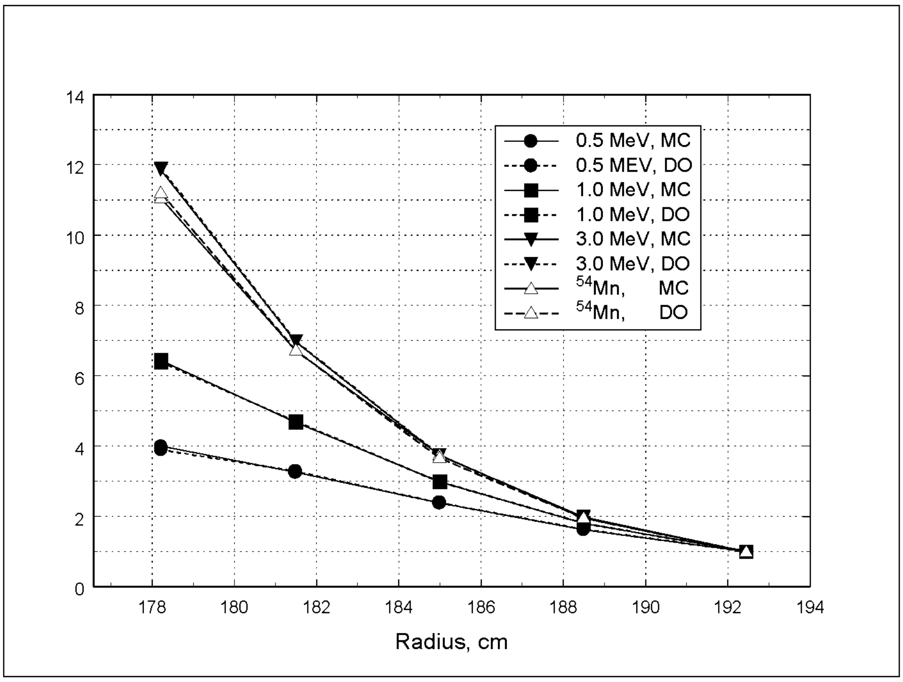

Fig. 3.1. Radial distribution of relative integral neutron fluxes and ^{54}Mn activity
Radial distribution of the coefficient of response (integral neutron fluxes and ^{54}Mn activity) attenuation toward the position behind the RPV

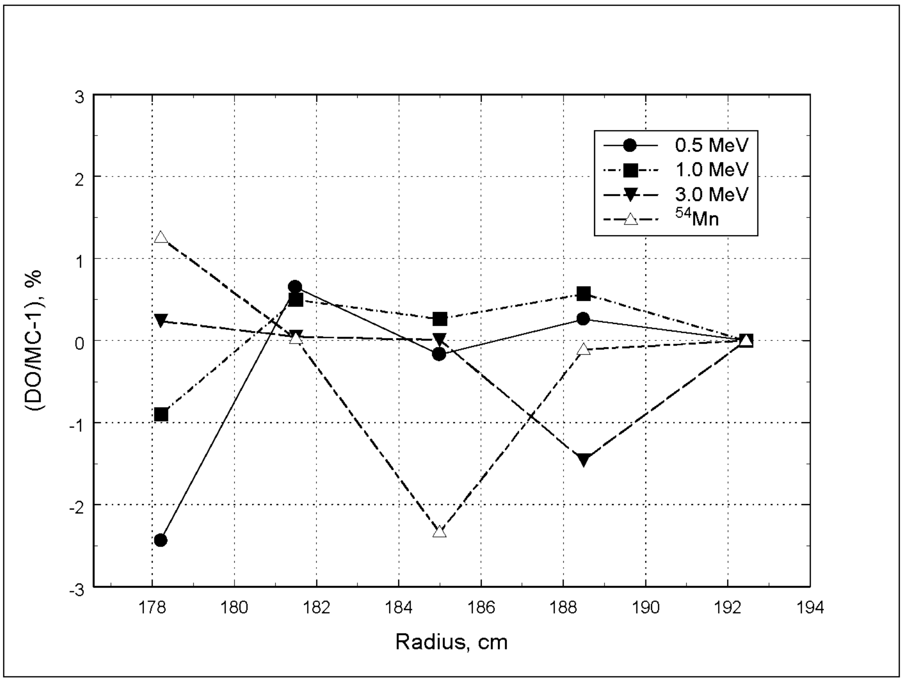

Fig. 3.2. Difference between DO and MC results for attenuation radial distribution of relative integral neutron fluxes and ^{54}Mn activity

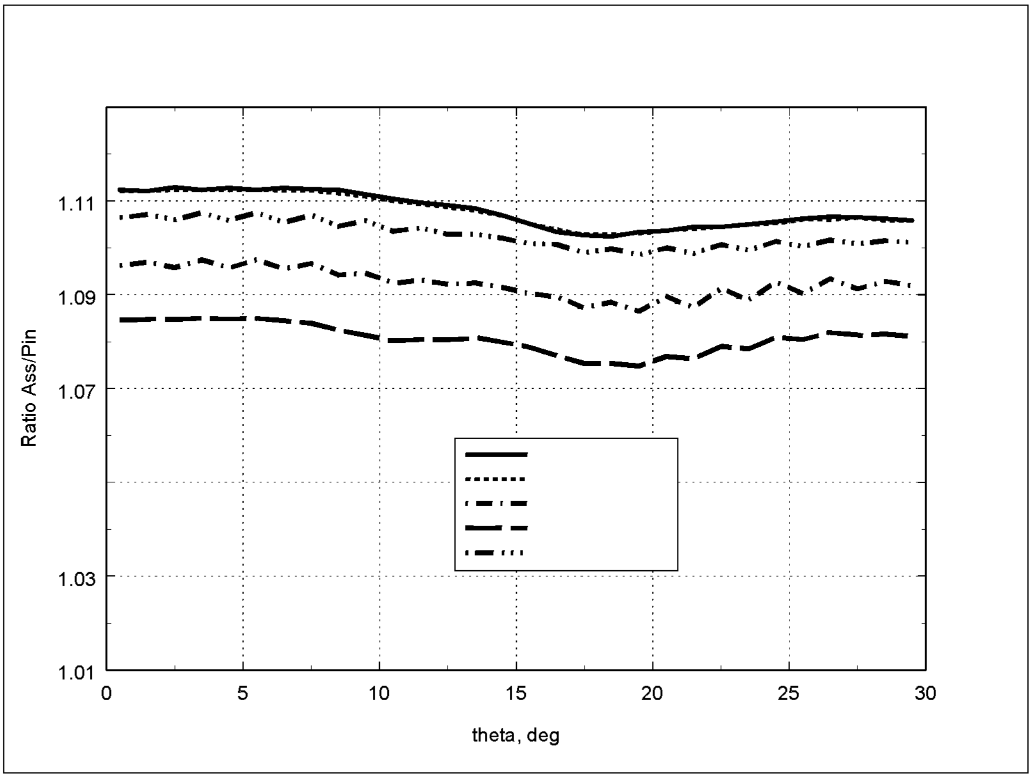

Fig. 4. Ratio Ass/Pin of neutron flux/activity values for VVER-440, dummy cassettes.

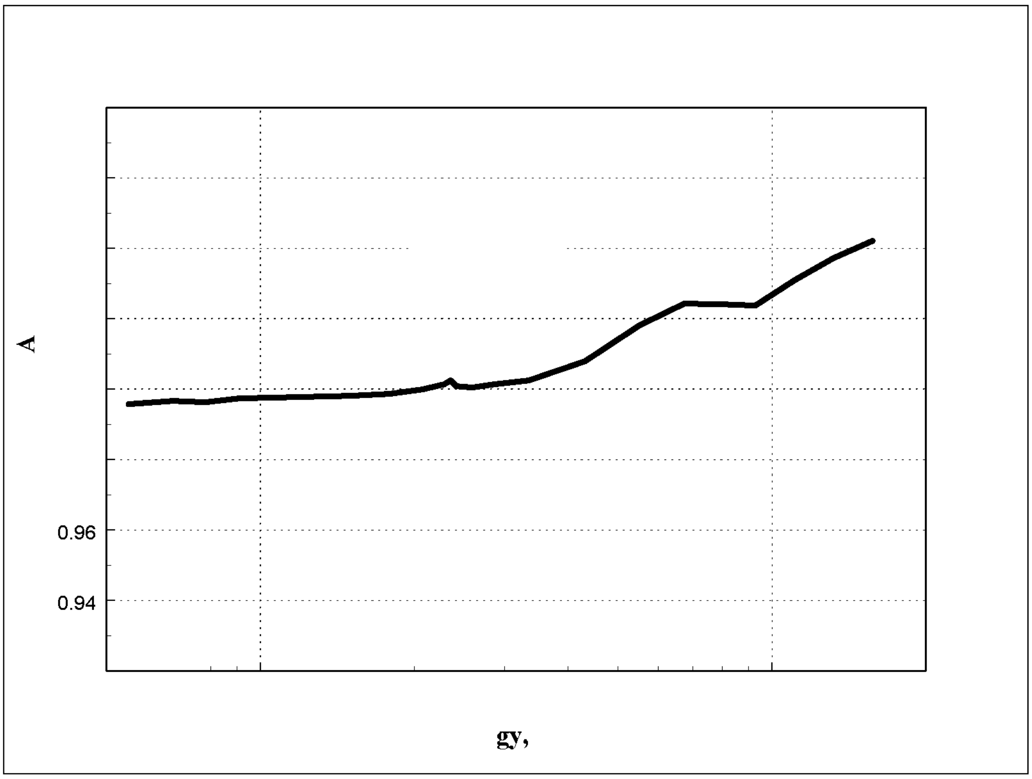

Fig. 5. Pin/Assembly ratio of VVER-440 RPV neutron spectra

4. Conclusions

Studies on the impact of multigroup presentation of the neutron cross sections as well as of negative gradient of reactor power in the periphery assemblies on the neutron fluence evaluation have been carried out.

The attenuation of neutron flux through the water in the downcomer of VVER-440 calculated by MCNP with continuous energy dependence of neutron cross sections (MC model) is 8% less than the attenuation calculated by DORT and multigroup neutron cross sections (DO model).

The good consistency of DO and MC results for the attenuation through the RPV shows that the calculation with multigroup approximation of neutron cross sections for iron leads to the same results as those obtained by continuos presentation of the neutron cross sections energy dependence. In this sense, it seems the suggestion that the discrepancy between experimental and calculational data is expected to be related to the multigroup approximation for iron data was not confirmed at least for RPV attenuation.

For better understanding of the impact of multigroup approximation to the calculation of neutron transport through the downcomer, a more detailed study has to be done.

The obtained results and conclusions are purposed to improve the calculational methodology of BE fluence for VVER-440 type of reactor. They will be implemented in the INRNE Neutron Fluence Methodology and applied for incoming comprehensive neutron fluence analyses at Kozloduy NPP.

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