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NEUTRON FLUENCE CALCULATIONS FOR THE REACTOR VESSEL

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ABSTRACT

The application of the coupled 3D static neutronphysical code and Monte Carlo calculations for the determination of the fast neutron flux in the reactor pressure vessel is examined. The possible ways for coupling the two systems are described and their advantages and disadvantages are methodologically investigated. The fast neutron fluxes calculated by the different coupling methods are presented and compared.

1. INTRODUCTION

The lifetime of a pressurized water reactor directly depends on the neutron irradiation of the reactor pressure vessel. The embrittlement of the material of the reactor vessel is due primarily to the fast neutron flux and the produced crystal defects. The possible extension of the reactor lifetime needs the neutronphysical investigations of how the new types of the fuel elements and new loading patterns influence the neutron flux attaining the reactor vessel and the detectors.

Predicting the possible lifetime needs calculations for the cycles in the past and for the planned cycles in the future. The calculations are also needed for the evaluation of the measurements. The calculations for a VVER-440 reactor vessel (in Paks NPP) were made by coupling the KARATE code system and the MCNP code. The KARATE program is a three dimensional, static, coupled thermohydraulic and neutronphysical code system (Gadó, 1994; Gadó, 1990) routinely used in Paks NPP. Following the cycles the burnup can be examined by the KARATE calculations, but the reactor regions far from the core can be modeled in detail with the required accuracy only in the Monte Carlo calculations. Coupling of the two systems, i.e. the determination of the source for the MCNP calculations by the KARATE program, can be realized in several ways and all of them need methodological investigations. Some of the possible coupling ways, their advantages and disadvantages are described in Section 2. The methodological investigations of the possible couplings are presented in Section 3. The MCNP input model for the Paks NPP VVER-440 reactor and the KARATE inputs are detailed in Section 4. The fast neutron flux in the detectors and in the reactor vessel is determined

with the different couplings of the KARATE-MCNP systems; the results are summarized in Section 5.

The presented calculations are the first step of our project. In the future these calculations will be tested by using the dosimetry measurements applied in the reactor.

2. CALCULATIONAL METHODS

Generally, in the recent days the calculational method for the neutron irradiation of the reactor pressure vessel is using a static reactor calculation code together a Monte Carlo and/or S_n code. One possible way to couple the codes that the source for the Monte Carlo calculation is determined by the deterministic code (Dekens, 1996; Emert, 1997). The Monte Carlo calculation is made with a source determined for a given state of the reactor core and this calculation is repeated in several time steps in order to determine the integral of the flux above 0.5 or 1 MeV in the reactor vessel. The source can also be determined from an other Monte Carlo calculations (Milgram, 1997). The other possibility of coupling is the implementation of the variance reduction in the Monte Carlo calculation with the help of a S_n code (Wagner, 1997; Taro Ueki, 1999; Giffard, 1999).

Our calculations were made by using the KARATE program system for generating the source for the MCNP Monte Carlo code. The KARATE code system consists of 4 levels of calculations:

1st level: Multigroup spectral calculations, the goal of which is to determine the cross sections for the 2- or 4-group calculations and the scalar flux and current spectra for the Monte Carlo calculations. The 2- and 4-group constants are calculated by the MULTICELL program, and the spectra are determined by the 1-dimensional COLA program with cylindrical geometry.

2nd level: 4-group, 2-dimensional fine mesh reflector albedo calculations, which evaluate the core-reflector albedo matrices. In these calculations the orientation and distance of the core from the steel structural elements are respected according to the detailed position dependent model. At the same time these albedo matrices don't give account of the change of the multigroup spectra in the macrogroup. In this respect the COLA model seems more suitable but it needs approaches considering the position dependent effects.
Therefore the global calculations (see 4th level) using the 4-group fine-mesh calculation albedo matrices have limited accuracy at the vicinity of the reflector. The deviation is reduced by fitting the reflector albedos in the KARATE system.

3rd level: 2-group, two-dimensional fine-mesh calculations, which determine the linear power distribution of the fuel rods. From these calculations the fission source of the Monte Carlo calculations for every level of the fuel rods and for every isotope can be determined. The sources of three isotopes (U-235, U-238, Pu-239) are calculated explicitly while the source of the rest actinides is lumped. The few-group constants are determined at the 1st level, the albedo matrices at the boundary of the core and the reflector are determined at the 2nd level.

4th level: 2-group, nodal (GLOBUS) calculations, which calculate the power distribution of the assembly levels and the inhomogeneous boundary conditions (given

flux at the boundary) for the 3rd level of the calculations. In addition these calculations determine the 2-group fluxes and currents at the boundary of the core and the reflector or at an inner boundary in order to generate boundary conditions for the Monte Carlo calculations with the help of using multigroup spectra. The few-group constants are determined at the 1st level, the albedo matrices at the boundary of the core and the reflector are determined at the 2nd level.

Three methods to determine the source by the KARATE system for the Monte Carlo calculations have been worked out. They are presented in the followings.

A. Surface boundary condition for the Monte Carlo calculations at the edge of the core.

From the neutron distribution at the boundary of the core and the reflector depending on the position, energy and flight angle calculated by the KARATE system the source of the Monte Carlo calculations can be determined, which represents the appropriate boundary conditions. This type of coupling has been used in several previous investigations (Lux, 1983).

A surface dependent, multigroup, linearly anisotropic boundary condition can be determined from the spectra of the 2-group node faces of the 4th level and from the spectra of the 1st level of the KARATE calculations. The unnormalized multigroup scalar flux and current (1st level) must be normalized, which can be made with knowing the 2-group scalar fluxes and net currents (4th level). The normalization of the scalar flux is unambiguous, but the current can be normalized by two ways. Supposing that the currents of the 4th calculational level are correct then the multigroup currents (calculated by the COLA program) must be normalized to the 2-group currents. Though the 2-group currents give account of the geometrical difference of the various faces they originate from the only 4-group calculations of the 2nd level. So it seems to be justified to multiply the multigroup currents of the macrogroups with the same constants as was used for the flux normalization. It means that in the respect of the anisotropy the results of the COLA calculations are accepted. In this case the effect of the position dependence is neglected.

The advantage of this procedure that the Monte Carlo calculations must be made in the medium with known composition and with no fissile material. Its serious disadvantage is that the angular distribution on the surface and the energy spectra can be determined with difficulty at the edge of the core.

B. Surface boundary condition for the Monte Carlo calculations inside the core.

In this case the core is divided into two parts which are separated by an inside surface. From the distribution of the neutrons moving through this inside boundary of the core towards the reflector depending on the position, energy and flight angle calculated by the KARATE system the source of the Monte Carlo calculations can be determined, which represents the appropriate boundary conditions. In this case the fissions caused by the neutrons moving between the inside boundary and the reflector must be also modeled. Between the inside surface and the boundary of the core the geometry and the material composition are described rod by rod. The isotopic compositions are determined by the KARATE program. This way the neutrons arriving at the boundary of the core emerged during the Monte Carlo calculation from fissions, i.e. the accuracy of their energy and

angular distributions is determined by the accuracy of the cross section libraries, and the original source-neutrons have enough collisions to develop a real distribution at the boundary. Therefore the radiation of the reactor vessel is less sensitive to the angular distribution of the source neutrons in the Monte Carlo calculation as in case A. The position of the inside source surface is determined by methodological investigations before the calculations (see section 3).

The disadvantage of this method is that the Monte Carlo calculation must be made in a region with burned fissionable material. Since the fissions must be taken into account, the neutrons have to be followed also on low energy region inside and in the near vicinity of the reactor. In case A the neutrons having energy below 0.5 or 1 MeV can be neglected. This means that the used CPU increases.

C. Volumetric source for the Monte Carlo calculations in the core

In the outer region of the core a fission source calculated by the KARATE program is given as a source for the Monte Carlo calculation. In this case during the Monte Carlo calculation the fissions must not be taken into account in this region, so the neutrons having energy below 0.5 or 1 MeV should not be followed. The source is determined by the 3rd level of the KARATE calculation for every axial level of every fuel rod and for the three isotopes (U-235, U-238, Pu-239) separately and the rest of heavy isotopes together. The angular distribution of the neutrons come from the fission is isotropic, their energy distributions can be given for all isotopes separately. In this region of the core the geometry, the material composition and the source is defined rod by rod. The isotopic composition is made also by the KARATE calculations.

The main advantage of this method is that it uses well-defined results of the pin-by-pin KARATE calculations (isotope concentrations and fission density), and the energy and angular distributions of the neutrons crossing the boundary of the core are developed during the Monte Carlo simulation. The disadvantage of the method is the more complicated input of the Monte Carlo calculation than it was in case B. This method also needs a methodological investigation which points out how wide part of the outer core must be modeled in detail in the Monte Carlo calculations that the boundary condition used on the inner surface of the core has no effect to the results.

3. MCNP METHODOLOGICAL INVESTIGATIONS

The methods B and C written in section 2 need methodological investigations, which point out what part of the outer core must be modeled in detail in the Monte Carlo calculations. In case B inside this region absorbing boundary condition should be used, but in case C it has to be also investigated what type of boundary conditions can be applied inside the core.

For the investigations Monte Carlo calculations were used in a simplified geometrical model. This model is the same for the cases B and C. An axially infinite, homogeneous lattice is defined from fuel element cells. Since the radius of the core and the outer elements is big enough, the methodological investigations are made in slab geometry. This is shown on the Figure 3.1. The z-axis is the radial direction of the core.

In this direction the baffle plate, water, core basket, water, core barrel, water gap, thermal shield, reactor vessel, cavity and the concrete regions are taken into account. On the sides of the slab parallel to the z axis reflective boundary condition is applied. In case B (inside surface source) the source is given on the surface signed with A on the Fig.3.1, and in case C (volumetric source) the source is given in the fuel rods found in the region between the surface A and the baffle plate with the thickness L and supposing constant volumetric distribution. In case B on the surface A black boundary condition must be applied, in case C the boundary condition depends on the result of the investigation. For both cases the flux is determined in the energy intervals 0.625 eV, 1 keV, 0.1 MeV, 0.5 MeV, 1 MeV, 20 MeV for the basket, for the core barrel, for the outer surface of the thermal shield, in the middle and on the outer surface of the reactor vessel.

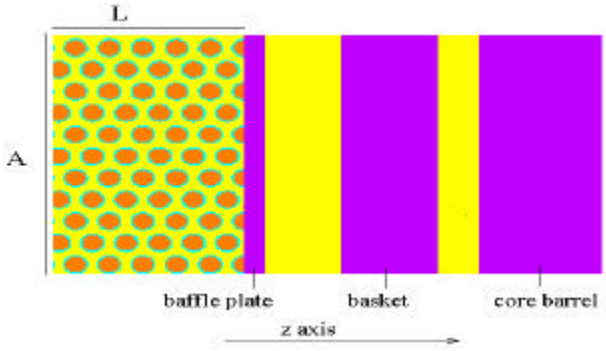

Fig. 3.1. Geometry for the methodological investigations

In case B (inside surface source) the effect of the angular distribution of the source calculated from the flux determined by the KARATE COLA program was investigated. The L was 14.7 cm, i.e. in the calculation there were only the very outer assemblies. The COLA angular dependent neutron flux can be written as a sum of one isotropic and one linearly anisotropic terms, and both have own energy spectrum, i.e. the degree of the anisotropy depends on the energy. (In the Monte Carlo calculations the two terms correspond to one linearly and one quadratically anisotropic ones.) The angular dependent neutron flux was calculated by the COLA module in 70 energy groups, on the surface farther in with one assembly from the reflector, for the Russian assemblies with enrichment 2.4 and 3.6%. Using these neutron fluxes Monte Carlo calculations were made with 4 different sources. There are two sources with the different enrichments and two ones using only the isotropic component of the neutron flux. It was pointed out that the deviation between the calculated fluxes with the 4 sources for the 6 energy groups and 5 surfaces (written above) was not more than 3.5% with the standard deviation of app. 1% of the calculations. The average deviation was less than 2%.

Since the change of the angular distribution of the source covers the predictable error of the angular distribution and the technological uncertainties are possibly more, an isotropic distribution can be used in case B (inner surface source) used by the Monte Carlo calculations. In this method B the calculations must be made in a region with burned fissionable material and the fissions are taken into account in the Monte Carlo calculations, thus the neutrons have to be followed also on low energy region inside and in the near vicinity of the core. For the determination of the possible energy cut-off the

ratio of the back-scattered neutrons crossed the inner and outer surfaces of the basket and the core barrel was calculated. It pointed out that the energy cut-off can be applied outside the core barrel since the ratio of the back-scattered neutrons from here is less than 1 %.

In case C (volumetric source) the thickness of L has to be determined so that the flux calculated in the 6 energy groups and at the core basket, at the core barrel, on the outer surface of the thermal shield, in the middle and on the outer surface of the reactor vessel does not depend on whether the boundary condition given on the surface A is reflective or black. In this case the deviation between the two results calculated with the different boundary conditions was app. 10 % when L was equal to 14.7 cm, i.e. using only one row of assemblies in the edge of the core is not enough correct in the calculations. When L = 2×14.7 cm, the maximal deviation is less than 2 %. The consequence is that using a volumetric source defined in two rows of assemblies at the edge of the core the consideration of the inner core is not required.

4. DESCRIPTION OF THE MCNP AND KARATE CALCULATIONS

4.1 The MCNP input model

The Monte Carlo calculations were made by using MCNP4B (Breismeister, 1997) with continuous energy representation and with the libraries ENDF/B-V and ENDF/B-VI. In the followings the geometrical input of the MCNP will be summarized, then the definitions of the different neutron sources and the used possibilities of the variance reduction will be described.

The geometry of the region outside of the core : In the outer geometry the reactor is modeled from the edge of the core to the outer surface of the concrete surrounding the reactor pressure vessel. The investigated regions are: water, baffle plate, water, core basket, water, core barrel, detector chains, water gap, thermal shield, reactor vessel, cavity and the concrete with the ionization chambers. The calculated area is a 60-degree sector of the core (see Fig.4.1). On the two sides of the 60-degree region reflective boundary condition is applied in the recent work. On the outer surface of the concrete black boundary condition is given. The material and geometrical characteristics of the outer part do not change in axial direction.

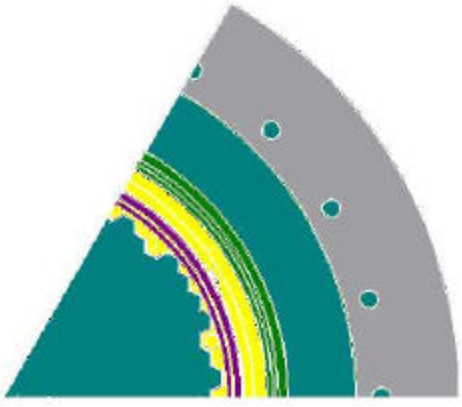

Fig. 4.1. The reactor model

The lattice pitch is 14.7 cm, the axial height of the core 250 cm. The calculated area is extended axially upwards and downwards with 1 m. Inside the core where the fuel is in these regions there is water elsewhere vacuum. Outside the core the model written in the above section is extended with 1 m in the axial direction. On the top and bottom of the extended region black boundary condition is applied.

The material composition of the outside part of the reactor : The region between the baffle plate and the reactor vessel consist of water and stainless steel.

The Charpy-s in the detector chains : Around the core 6 irradiation places can be found, in which there are 2-2 detector chains, that is series of detectors at different axial positions. The detectors are several kinds of activation foils and Charpy impact test specimens made of reactor vessel material which are used for the determination of the ductile – brittle transition temperature. The detector chains can be found next to the core barrel with a stainless steel cover. The position of one irradiation place in the 60-degree sector of the core can be seen in Fig. 4.2.

The data of the core : The calculations were made using standard Russian fuel elements without enrichment zoning. In the VVER-440 core there are 349 hexagonal assemblies consisting of 126 fuel rods. The enrichment of the fuel pellets U-235 metal/(U-235 + U-238 metal) w/o: 1.6, 2.4 and 3.6.

The sources used in the Monte Carlo calculations : The neutron sources are given in different places of the core corresponding to the different models A, B and C described in Section 2. In all cases the region inside the source regions is black according to the model, the neutrons scattered back to this region are not followed. The energy spectra for the surface sources and the number of emerging neutrons for the volumetric source calculated by the KARATE program are normalized to the power of the reactor core. In case of the surface sources the surfaces are replaced by very thin cells in order to simplify the geometry model of the Monte Carlo calculations. In the calculations the flux is determined above 0.5 MeV, in the following we refer to it as the neutron fluence.

1. Outer surface source : The source is given on the outer surface of the core in the water gap inside the baffle plate. A part of the geometrical model belonging to the outer surface source can be seen in Figure 4.2. The outer concrete is not shown in this figure.

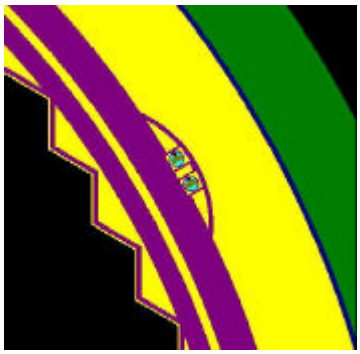

Figure 4.2. Outer surface source

In the geometrical model the core consists of hexagonal regions (assembly and water gap), and the outer edges of the outermost assemblies construct the outer edge of the core. The neutrons are started from the water gaps which can be found here. One source cell is a water gap belonging to the outer side of a hexagon divided into 10 axial layers. Every source cell has its own energy and angular dependent spectrum.

For the source definition sampling of the neutron depending on the position, the energy and the angle should be given:
Position: At first one cell is sampled, which is one axial level of one edge of an assembly. The probability of sampling the cell is the integral of the normalized energy spectra. Inside the cell the position dependent distribution is uniform.
Energy: In the sampled cell the neutrons are started according to the 70-group energy spectrum belonging to the given cell.
Flight angle: The distribution of the flight angle depends on the energy and the position (sampled cell). The MCNP does not support the two level dependence of the source variable. Since the angular dependence calculated by the KARATE is determined as an energy dependent combination of a linearly and a quadratically anisotropic part, therefore two separate calculations are made, one uses a linearly anisotropic angular dependent source, the other one uses a quadratically anisotropic source. The energy dependence is changing from cells to cells. The results of the two calculations should be summarized in a suitable way to obtain the real flux for the calculated regions.

2. Inner surface source : The source is given inside the core, farther in with one assembly row from the edge of the core. The assembly row is modeled in detail, as one can see in Figure 4.3.

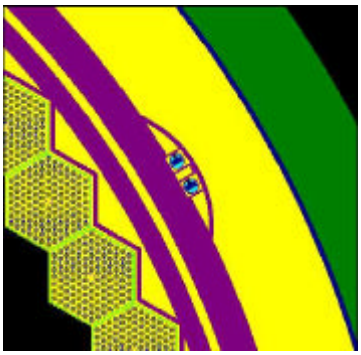

Figure 4.3. Detailed geometry of the inner surface source close to the detector chains

The sampling of the neutrons on the inner surface is similar to the sampling on the outer surface. Very thin cells are defined at the inner edges of the assembly row, which contain vacuum. Every edge is divided into 10 axial levels, and a source cell belongs for every level of every edge. Sampling of the neutron depending on the position, the energy and the angle is the following:
Position: same as previously
Energy: same as previously
Flight angle: The methodological investigations pointed out that in this model the quadratically anisotropic part of the source can be neglected according to the fissions and scattering in the outer assembly row. Thus only one calculation should be made with

linearly anisotropic source. The final result is calculated considering the total number of the neutrons. The fissions are taken into account in the Monte Carlo calculations.

3. Volumetric source : The source is given inside the core, two assembly rows are modeled in detail. Every fuel rod is divided into 10 axial levels, and the source cell is one axial level of a fuel rod. The sampling of the cells depends on the number of the neutrons emerged in the cells from the fission according to the KARATE calculations. Inside the cell the position dependent distribution is uniform and the angular dependent distribution is isotropic. The energy dependence of the source is produced from the fission spectra of the fissionable isotopes. In the Monte Carlo calculations the fissions are treated as absorptions. A part of the geometrical model of the volumetric source can be seen in Figure 4.4. One can see the positions of the assemblies with different enrichments.

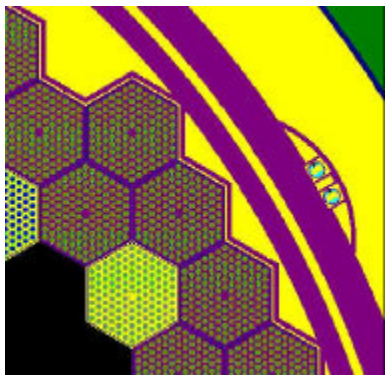

Figure 4.4. A part of the geometry of the volumetric source close to the detector chains.

Variance reduction : Reducing the CPU time which is needed for the results with the required accuracy, several variance reduction methods were used. In every calculation the splitting/Russian roulette was applied. It means that the importance of each region should be set in a proper way, the too thick regions were divided into more parts. This procedure is known very effective especially in those cases, when a strongly absorbing region is placed between the source and the investigated region as it is in our case. Since in our calculations the energy interval above 0.5 MeV is interesting, almost in all of the calculations the energy cutoff with 0.5 MeV was used. In case of the inner source it could be used for only the regions outside the core barrel as it was pointed out in the methodological investigations.

4.2 KARATE calculations

For the investigation of the several connections between the KARATE and MCNP 7 different KARATE-MCNP calculations were made. The procedure of the KARATE calculation and its levels were described in Section 2.

All the calculations were made for the starting cycle of the Paks NPP, unburned fuel elements, zero power and unpoisoned state. The temperature was 260 ^{o}C. All the calculations concerned the same core and the same geometry and material composition.

The 3rd level (fine-mesh) of the KARATE calculations has no high precision near the reflector, because it was not important for the calculations made up till now as the maximum linear power is not here. Thus the fine mesh calculations use only one type of albedo matrices, which is a matrix in an average distance from the stainless steel structural elements.

In the nodal calculations (4th level of the KARATE calculations) the reflector albedo matrices could be different according to the orientation and the distance from the structural elements, but the approach applied at the 3rd level of calculations (i.e. the same albedo) can be used, too. In this case for the identification of the calculated results the „**s**" sign is used for shorting the „simplified albedo".

The values of the albedos were fitted with using measurements. This fitting caused an increasing from −7 to 46 % in the values of the albedos and a 5 % changing in the power of the outside assemblies. For the identification of the calculated results in these cases the „**r**" sign is used for shorting the „real albedo". For methodological investigations the calculations were made with different albedos but without the above fitting. In this case the „**nf**" abbreviation was used for shorting „no fitting". In these calculations it was supposed that the deviation of the calculations and the measurements is caused by the different flow in the outside assemblies.

The unnormalized multigroup scalar flux and net current (1st level) must be normalized, which can be made with knowing the 2-group scalar fluxes and net currents (4th level). The normalization of the scalar flux is unambiguous, but the current can be normalized by two ways. Supposing that the currents of the 4th calculational level are correct then the multigroup currents (calculated by the COLA program) must be normalized to the 2-group currents. For the identification of the calculated results in this case the „**c**" sign is used for shorting the „normalization according to the current".

In addition the multigroup currents at the macro groups can be multiplied with the same constants as was used for the flux normalization. It means that in the respect of the anisotropy the results of the COLA calculations are accepted. For the identification of the calculated results in this case the „**f**" sign is used for shorting the „normalization according to the flux".

The effect of the linearly anisotropic boundary condition in the Monte Carlo calculations was replaced by a source in a very thin volume. In this case the number of the neutrons crossing the surface must be taken into account, so in this source the linearly anisotropic part is equivalent to the isotropic flux boundary condition and the quadratically anisotropic part is equivalent to the linearly anisotropic boundary condition (see the other part of the report).

The 7 types of calculations carried out are described in the followings:

Outer surface sources (case A in Section 2):

- **osrc:** surface boundary condition for the Monte Carlo calculation at the edge of the core (on the outer surface = „**os**"), real (different at the different edges and fitted)

albedo („**r**”) in the GLOBUS calculations (4th, nodal level), normalization of the multigroup current to the few-group current („**c**”).

- **ossc:** surface boundary condition for the Monte Carlo calculation at the edge of the core (on the outer surface = „**os**”), simplified (same and no fitted) albedo („**s**”) in the GLOBUS calculations (4th, nodal level), normalization of the multigroup current to the few-group current („**c**”).
- **ossf:** surface boundary condition for the Monte Carlo calculation at the edge of the core (on the outer surface = „**os**”), simplified (same and not fitted) albedo („**s**”) in the GLOBUS calculations (4th, nodal level), anisotropy from the COLA calculation („**f**”).
- **osnff:** surface boundary condition for the Monte Carlo calculation at the edge of the core (on the outer surface = „**os**”), different but not fitted albedo („**nf**”) in the GLOBUS calculations (4th, nodal level), anisotropy from the COLA calculation („**f**”).

The spectra were calculated by the COLA program in cylindrical geometry with given temperature and boron concentration. The multigroup flux and current spectra valid at the core-reflector boundary were used. The enrichment is equal to the enrichment of the assembly at the border, i.e. 3.6% or 2.4%.

Inner surface sources (case B in section 2):

- **isnf:** surface boundary condition for the Monte Carlo calculation inside the core (on the inner surface = „**is**”), different but not fitted albedo („**nf**”) in the GLOBUS calculations (4th, nodal level). The inner surface is in the distance of one assembly lattice pitch from the edge of the core.
- **iss:** surface boundary condition for the Monte Carlo calculation inside the core (on the inner surface = „**is**”), simplified (same and not fitted) albedo („**s**”) in the GLOBUS calculations (4th, nodal level). The inner surface is in the distance of one assembly lattice pitch from the edge of the core.

In Section 3 the methodological investigations pointed out that if the inner boundary is at least one assembly lattice pitch from the core-reflector boundary then the uncertainty of the angular dependent distribution of the entering neutrons causes a negligible error in the final result and changing the energy dependent distribution made also no effective deviation. Because of this in the coupled KARATE and Monte Carlo calculations the anisotropy was not too important. In cases **isnf** and **iss** the multigroup flux was normalized to the few-group flux and the multigroup current to the few-group current, then the multigroup incoming current was determined from the normalized multigroup flux and net current. Then the multigroup scalar flux was determined which leads to the same incoming current. This isotropic flux boundary condition was prescribed for the Monte Carlo calculations. The spectra were calculated by the COLA code.

Volumetric source (case C in section 2):

- **vols**: volumetric source for the Monte Carlo calculations („**vol**”). The inner surface is in the distance of 2 assembly lattice pitches from the core-reflector boundary, inside the inner surface black boundary condition is applied. Simplified (same and not fitted) albedo („**s**”) in the GLOBUS calculations (4th, nodal level) was used.

In Section 3 the methodological investigations pointed out that if on the inner boundary positioned in the distance of 2 assembly lattice pitches from the core-reflector boundary the black boundary condition is applied, it hardly modifies the results of the calculations with the volumetric source.

5. COMPARISON OF THE RESULTS CALCULATED BY THE DIFFERENT METHODS

The fluence was determined by the different source cases (see previous section)

- for the Charpy-s in the detector chains;
- for the one cm thick **region** in the water outside the core barrel, in the thermal shield, in the quarter and half of the reactor vessel and in the outside of the vessel, averaged axially and azimuthally;
- for the one cm thick region in the different positions of the reactor (for example: in the water outside the core barrel, in the thermal shield and in the half of the reactor vessel), divided into 12 uniform **sector**s azimuthally and averaged axially;
- for the one cm thick region in the different positions of the reactor (for example: in the water outside the core barrel, in the thermal shield and in the half of the reactor vessel), in 11 different axial **level**s with 1 cm height and averaged azimuthally.

The results are summarized in the next figures.

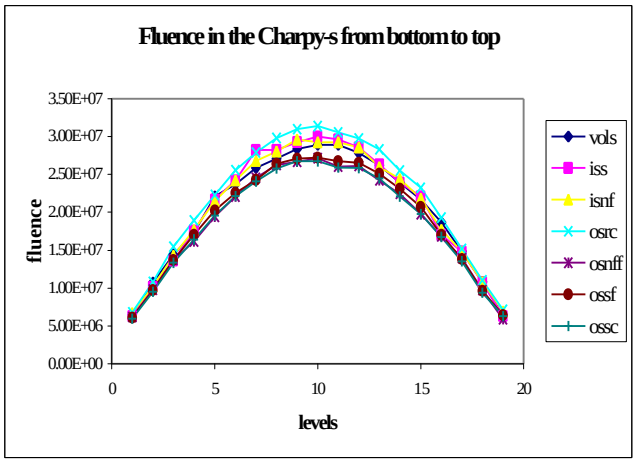

Fig.5.1. The fluence calculated in the Charpy-s

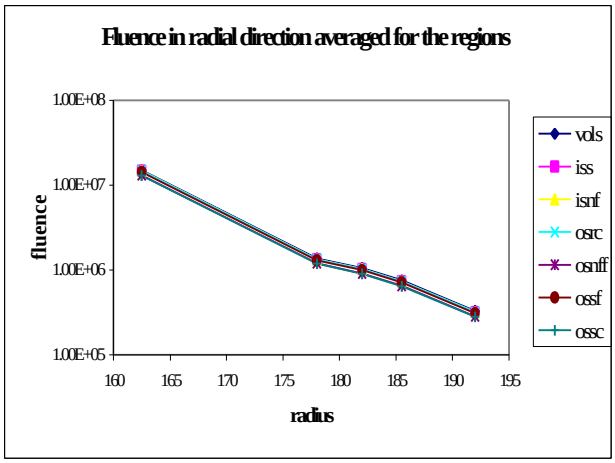

Fig.5.2. The fluence in radial direction averaged for the regions

The global lead factors can be determined from the fluences averaged on the regions. The definition of the global lead factor is $L = L_0 / L_n$,where L_0 is an average fluence in a given region and L_n is an average fluence in a chosen other region. In our case the given region is a 1-cm thick water region next to the core barrel and the other regions are the 1-cm thick layers in the thermal shield, at the quarter, half and outside of the reactor vessel. The global lead factors hardly deviate from each other according to the

facts written above, and one can see in the table 5.1 and in the figure 5.3. In the table the relative statistical errors are less than 5×10^{-3}.

Table 5.1. The global lead factors

	vols	iss	isnf	osrc	osnff	ossf	ossc
thermal shield	10.93	10.99	11.00	10.96	10.92	11.10	10.90
reac.vess. quarter	14.18	14.37	14.39	14.33	14.29	14.52	14.26
reac.vess. half	19.75	20.05	20.09	20.04	19.96	20.29	19.93
reac.vess. outside	45.26	46.04	46.17	46.02	45.88	46.61	45.78

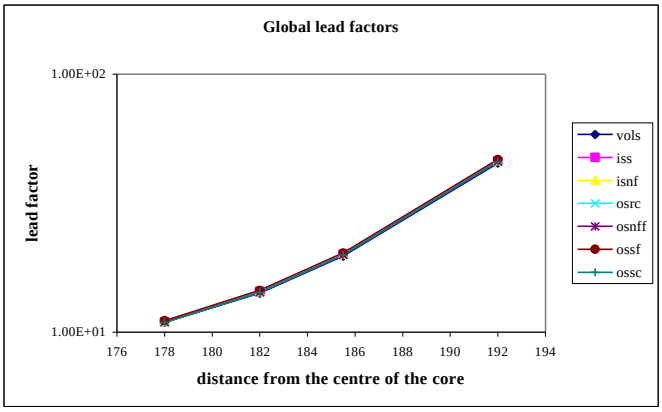

Fig.5.3. The global lead factors

In the figures 5.4.-5.5. the axial change of the azimuthally averaged fluence can be seen for the 1-cm thick regions in 11 axial positions. For the sake of the briefness only two cases are presented from the calculated different positions. The deviation of the volumetric source and inner surface source cases is within the statistical error.

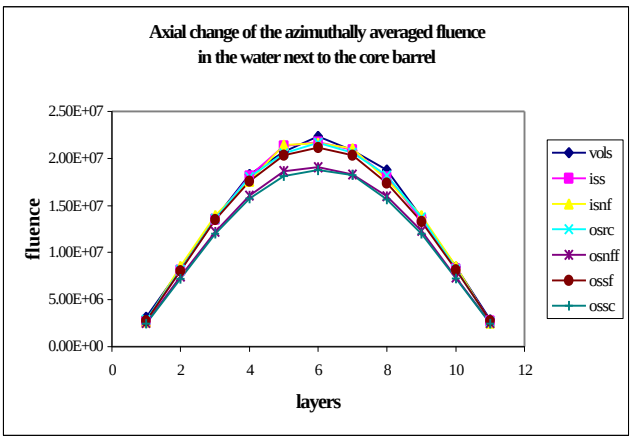

Fig.5.4. Axial change of the azimuthally averaged fluence in the water next to the barrel.

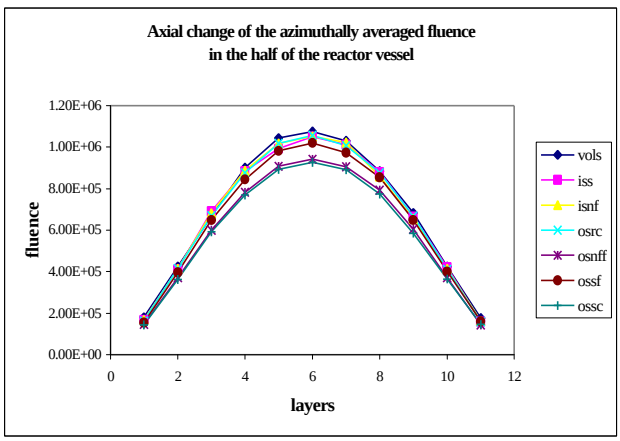

Fig.5.5. Axial change of the azimuthally averaged fluence in the half of the reactor.

In the figures 5.6.-5.7. the azimuthal change of the axially averaged fluence can be seen in two different regions with the help of 12 azimuthal sectors. In this case the

deviations of the results calculated by the volumetric source and inner surface source cases are also within the statistical errors.

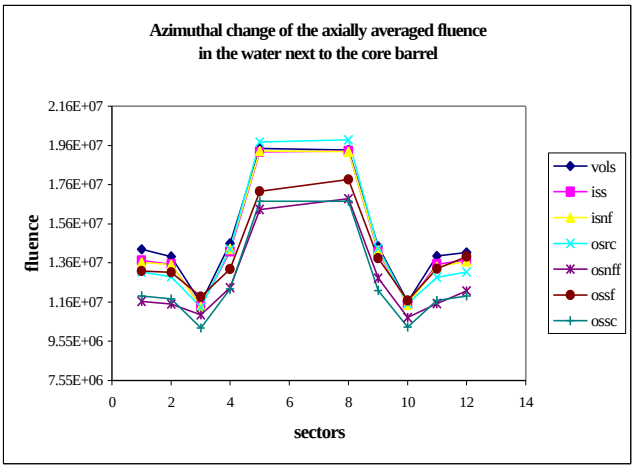

Fig.5.6. Azimuthal change of the axially averaged fluence in the water next to the barrel.

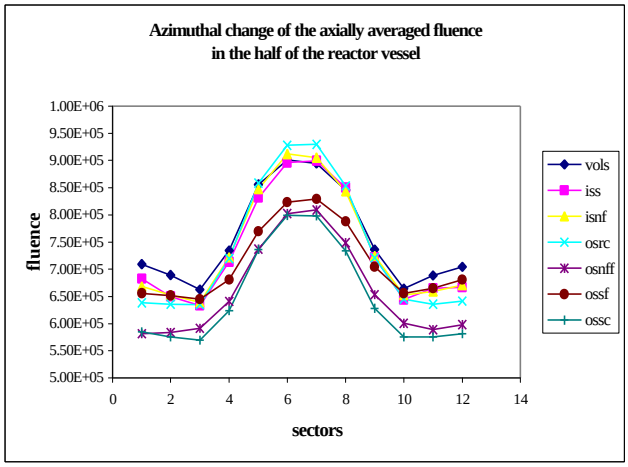

Fig.5.7. Azimuthal change of the axially averaged fluence in the half of the reactor vessel.

Based on the physical considerations the results calculated by the inner surface sources and the volumetric source can be considered the most accurate and they really correspond well to each other (**iss**, **isnf** and **vols** cases). This assumption and the results described above will be tested in a later phase of our project using the dosimetry measurements applied in the reactor and by full core calculations.

6. CONCLUSION

The detailed input model of the MCNP characterizing the VVER-440 reactor is developed which makes the calculation of the neutron fluence needed for the investigation of the reactor vessel embrittlement possible in the following cases:

- fluence above the energy 0.5 MeV or 1.0 MeV in different positions of the vessel;
- the fluence in the Charpy-s.

Three different types of possible coupling of the KARATE code system and the MCNP were elaborated. The advantages and disadvantages of the coupling methods were investigated. Therefore Monte Carlo methodological investigations were carried out in slab geometry and the fast neutron flux in the detectors and in the reactor vessel was determined by the different methods coupling the KARATE-MCNP systems.

The investigations pointed out that the results calculated by the volumetric source and the inner surface source are in good agreement. In case of the inner surface source the selection of the albedos used in the KARATE calculations and the angular distribution of the neutron source have negligible effects. The results calculated by the outer surface sources, except the **osrc** case, are rather below the results of the volumetric and inner surface sources. The results calculated by the **osrc** source overestimates only a little the

results of the volumetric and inner surface sources. This supports the applied fitting of the reflector albedos in the KARATE system.

Taking into account that the rate of the used CPU time is 1:5:8 using outer surface / volumetric / inner surface sources for the same accuracy (8 hours for Charpy-s by osrc method on Pentium II PC, 500 MHz), the calculation with the volumetric source should be chosen for the dosimetry calculations of the reactor vessel. The **osrc** method can be used for preliminary investigations because of its suitable accuracy and higher speed.

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