ABSTRACT

The “MIDICORE” VVER-1000 core periphery power distribution benchmark was proposed by Mr. Mikoláš on the 20th Symposium of AER in Finland in 2010. This MIDICORE benchmark is a 2D calculation benchmark based on the VVER-1000 reactor core cold state geometry with taking into account the geometry of explicit radial reflector. The main task of the benchmark is to test the pin by pin power distribution in selected fuel assemblies at the periphery of the VVER-1000 core. In this paper we present our results (k_{eff}, integral fission power) calculated by MCNP and the KARATE code system in KFKI-AEKI and the comparison to the preliminary reference Monte Carlo calculation results made by NRI, Rez.

1. INTRODUCTION

In the eighties, the complex reactor physical calculations have been organised into large codes. Since then, a considerable development of the calculation models has been carried out and new type of fuels came into practice. Units of such large codes should be tested which gives rise to a wider range of tests in a benchmark collection used for a given reactor type. This strategy was also applied to VVER-specific benchmarking. Benchmark problems were classified into numerical, experimental and operational problems. Some benchmarks considered already within the programmes of Atomic Energy Research (AER) association on VVER reactor physics and reactor safety are available via internet at location: http://aerbench.kfki.hu. Recently, some versatile benchmark tasks were specifically defined for the investigation of larger heterogeneities in the VVER core [1-2].

The diffusion approximation is often not sufficient, if heterogeneities are considerable. Moreover, generating homogenised macroscopic cross section data for the diffusion equations, homogenisation errors occur in the case, if the appropriate neighbourhood effects are not taken into account. Another source of errors is the energy discretisation within a few
energy groups. Last but not least the numerical modelling of core periphery is complicated due to the difficulty of the definition of boundary conditions. Testing of code accuracy at this field can help to increase the credibility of the power distribution calculated.

Relatively large difference of fuel pin power value is used to be observed in VVER-1000 NPP when this value is determined by codes based on pin to pin diffusion difference method on one side and by codes based on nodal diffusion method with pin power reconstruction on the other side. Therefore the validation of the applied calculation methods is necessary, although the distribution of the fuel pin power is not directly measured. To study this phenomenon, a mathematical benchmark representing a simplified sector of VVER-1000 core was defined by ŠKODA JS a.s. in cooperation with ÚJV Řež a.s.[3]. The main considerations of the problem-definition are the followings:

- typical part of the VVER1000 reflector with the all the known heterogeneities,
- only a small cut of the VVER1000 core is used to limit the calculation burden.

The MIDICORE benchmark is a 2D calculation benchmark based on the VVER-1000 reactor core cold state. It consists of 37 fresh fuel assemblies with 4 different enrichments in 60 degree symmetry. The details about core basket and the geometry of explicit radial reflector are given, too. The problem has been solved by different codes [3-4].

In this contribution first the benchmark will be presented then the linear pin power calculation methodology of KARATE will be explained. Finally our reference solution prepared by MCNP and the preliminary result of KARATE will be given and compared to the available results.

2. OUTLINE OF THE MIDICORE BENCHMARK

A short description of the 2D mathematical benchmark is presented below. The details can be found in [3]. The horizontal cross section of the benchmark problem is hexagonal and it is restricted only on a small part of the core at the basket near the periphery as denoted by bold dashed lines in Fig. 1. It is based on the cold state geometry of the VVER1000 reactor vessel. The inner edge of the reactor vessel presents the outer boundary of the model in the radial direction. The core segment with the fuel assemblies (FA’s) names and numbering can be seen in Fig. 1.
The reflection boundary conditions are used on the azimuthal surfaces numbered as 1 and 2; while the total absorption boundary condition (leakage to the vacuum) is used on the cylindrical outer boundary numbered as 3 (see the bold dashed lines in Fig. 1.). The reflection boundary conditions are used in axial directions.

Material composition is homogeneous in axial direction, except the steel-water mixture at a small part of the reflector. On the arc of the basket there are grooves on its outer surface. That is why two different models are introduced. In the reference model a periodic steel/water structure 20/38 mm has be used, while in the standard model this structure is smeared over the volume. There are several cylindrical holes in the basket. Its diameter is 70 mm and strict position can be read from the Fig. 1. The distances of the holes centre from the inner edge of basket are 75 mm and 150 mm, respectively. Finally a 3 mm water gap can be found between the edge of core basket and the edge of FA which has to be modelled, too. The reflector with the basket can be divided into with reflector hexagon. Their names are non-standard one, however the remaining FA’s have standard Russian names and their details are defined in Figs. 2.a-2.d. and in [3]. The cluster control rods are not inserted into the FA’s and the guide tubes are filled with water.
The main issue of this MIDICORE benchmark is to give some reference for validation of pin by pin power distribution at the periphery of the VVER-1000 core calculated by standard core calculation method using few-group diffusion approximation.

Two solutions are required. In the first case the standard method used in the everyday practice has to be applied. In the second case reflector assemblies presented in Figure 1 have to be considered in details in the calculations. The results to be reported:

- $k_{\text{eff}}$
- Integral fission power of FA’s No. 1-10. (normalized for the full core: 37 FA’s)
- Pin by pin power distributions in FA No. 6 - 9 (normalized for the assembly: average relative pin powers equals 1 in each FA’s)
3. LINEAR PIN POWER CALCULATION METHODOLOGY IN KARATE CODE SYSTEM

The KARATE-440 code system version 5.1 applicable for the calculation of VVER-440 reactors is based on the ENDF/B-VI nuclear data library [5-6]. The main goal of the calculation is core reload design, however, certain problems amenable to a static code can be analysed by KARATE-440. Accordingly, stationary neutron physics and thermal hydraulics models have been implemented. These models are capable of following burnup and slow Xenon transient processes but do not allow for calculating faster transients demanded in a safety analysis. The program serves economic core reload design so that the limitations demanded by the safety analysis should be observed. The reload limitations demanded by the safety analysis are also available from the calculations.

KARATE 440 involves all the libraries and computer programs, which are needed to perform fuel cycle calculations and fuel cycle design. The intra assembly power distribution is also determined. The libraries need refreshment if a new fuel type is being used or if the parameter range of an existing fuel is being extended. The calculation is grouped into 3 levels. A level is connected to the higher one through parameterised data libraries. These libraries provide a part of the input data for the higher level. A level is connected to the lower one also, usually boundary condition is provided for a “Lupe”-like calculation. The levels involved in KARATE include:
- cell or assembly spectral calculation by using the 1D COLA and 2D MULTICELL transport modules,
- assembly level to provide homogenized assembly library and to calculate pin powers in selected assemblies by using a 2D 2 or 4 group diffusion calculation,
- global level to determine criticality parameters and power distributions by using a 2 group nodal method.

Core calculations are made with the GLOBUS nodal code using the homogenized few group cross sections of assemblies, then as a result of the core calculations (with flux boundary conditions) inhomogeneous type fine mesh diffusion calculations are carried out for the assembly and its vicinity with the SADR code. Figure 3 presents the calculated domain.
As the goal of the work is the validation of the KARATE-440 code system concerning the linear pin power calculation near the basket, in solving the test case we applied the same methods and procedures as in case of routine power plant calculations. The difficulties of the calculations in the reflector region are as follows.

- Very complicated structure (geometry and composition data) of the basket. Even the Monte-Carlo calculations can lead to different results due to the different input data created by different users. There are different technical data available. Even the Monte-Carlo calculations must be validated against measurements.
- Complicated spectral and 3D spatial effects of the neutron transport: diffusion approximation in the very heterogeneous region is not satisfactory without some special treatment. 3D effects must be taken into account. Usually core design calculations are based on the few-group diffusion approximation. Fine mesh calculations taking into account the flux tilt caused by the environment of the calculated region are usually two-dimensional.

In KARATE, the reflector parts and the control assembly regions are excluded from the diffusion type calculations, and represented by albedo matrices. The elements of the albedo matrices $[\alpha_{g'g}]$ are the reflection probabilities for neutrons entering the excluded region in group $g'$ and returning to the fuel assemblies in group $g$, thus for the partial current that enters the control assembly from the fuel $[J^+]$ the returning partial current $[J^-]$ can be expressed as

$$J_g^- = \sum_{g'\geq g} \alpha_{g'g} J_g^{+}$$

According to the results of the methodological investigations, the albedo matrix elements can be considered as a function of the soluble boric acid concentration $[C_B]$ the moderator density $[\rho_m]$ and the position of the edge.
In the KARATE calculations, first 1D multigroup transport calculations of the different coupler regions and the neighbor assemblies are performed by using the COLA module, and the obtained two-group albedo matrices are parameterized. Finally a correction factor D is determined by using MCNP 3D results.

\[ \alpha_{\text{eff}} = \alpha_{\text{eff}}(C_B, \rho_m) \]

\[ J_{\text{MCNP}}^- = D \alpha_c(p) J_{\text{MCNP}}^+ \]

In case of the fuel pin few group XS library and the fuel assembly library new calculations were necessary for the simplified fuel assembly description in the benchmark. Correspondingly new albedo matrices were developed for reflector albedos according to the specification.

### 4. Calculation Results

On the basis of the benchmark specification Monte Carlo and KARATE calculations were carried out. All presented Monte Carlo calculations have been performed with MCNP5 ver1.40 code [7]. The ENDF/B6 library was chosen as the basic data library (14c for uranium isotopes, .62c and .66c for the others) but for the Fe, Co and Ni isotopes the ENDF/B-V data were chosen. Possible source convergence problems were checked by increasing the number of active and passive cycles as well as the number of neutrons per cycles. No significant change of the evaluated quantities was found. It is worth to mention, that our model has been validated against VVER-1000 benchmarks with good results [1-2].

In our calculation the standard model was used. For that case different MCNP input decks were developed in three institutions separately: Rez [3] RRC Kurchatov [4] and AEKI. The calculated effective multiplication factors are shown in Table 1.

<table>
<thead>
<tr>
<th></th>
<th>( k_{\text{eff}} )</th>
<th>( \sigma )</th>
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<tbody>
<tr>
<td>SKODA-REZ</td>
<td>1.04538</td>
<td>0.00003</td>
</tr>
<tr>
<td>RRC Kurchatov</td>
<td>1.039516</td>
<td>-</td>
</tr>
<tr>
<td>KFKI AEKI</td>
<td>1.04411</td>
<td>0.0001</td>
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The difference among the assembly integrated power distribution is less than 0.46% the average value was 0.3%. Concerning the pin power distributions the difference was less than 1%. The highest discrepancies could be found near the basket. Generally, no significant change was found.

In case of KARATE calculation the first step was the few group library preparation for the nodal and pin by pin model. They were performed by the MULTICELL code. The calculated infinite multiplication factors for four different assemblies with our MCNP results are presented in Table 2. In the second step new albedo set were also evaluated.

<table>
<thead>
<tr>
<th>Name of FA</th>
<th>$K_{\text{inf}}$ calculated by MCNP</th>
<th>St.Dev of MCNP</th>
<th>$K_{\text{inf}}$ calculated by MULTICELL</th>
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<tbody>
<tr>
<td>A200</td>
<td>1.07861</td>
<td>3.9E-4</td>
<td>1.07167</td>
</tr>
<tr>
<td>A40E6</td>
<td>1.22281</td>
<td>4.0E-4</td>
<td>1.21420</td>
</tr>
<tr>
<td>P36E9</td>
<td>1.16005</td>
<td>4.0E-4</td>
<td>1.15152</td>
</tr>
<tr>
<td>P40E9</td>
<td>1.19049</td>
<td>4.0E-4</td>
<td>1.18061</td>
</tr>
</tbody>
</table>

The underestimation of the MULTICELL is significant and it would be hard to attribute to different nuclear data used by both codes. The assemblies where gadolinium pins can be found the underestimation is more pronounced. On the other hand the MULTICELL code gave rather good results against measurements containing gadolinium perturbation [8].

The discrepancies found in the preparation of the FA library together with the deviation of the power distributions predicted by KARATE and MCNP in Fig. 4.a and 4.b.(higher leakage indication of KARATE) explain the $k_{\text{eff}}$ deviation.

**Figure 5. a:** MCNP reference model. $K_{\text{eff}}=1.04520$

**Figure 5. b:** KARATE(GLOBUS) model. $K_{\text{eff}}=1.03274$
Figure 6, MCNP result for assembly 6, A200

Figure 7, MCNP result for assembly 7, P36E9
Figure 6, 7 and 8 show the pin by pin distribution calculated by MCNP.

5. SUMMARY

The MIDICORE 2D mathematical benchmark was solved by the MCNP and KARATE code system. Rather good agreement was gained in case of Monte Carlo simulation comparing the results of three different data sets.

In case of KARATE the same methods and approximations were used as in case of VVER-440 NPP applications. The underprediction of $k_{\text{eff}}$ was already observed in the library preparation. Though the KARATE code system was widely validated against VVER-440 measurements, further investigations are necessary to understand the sources of errors in this benchmark.

6. REFERENCES


