THE KARATE PROGRAM SYSTEM

Cs. Hegedűs, Gy. Hegyi, G. Hordósy, A. Keresztúri, M. Makai, Cs. Maráczy, F. Telbisz, E. Temesvári, P. Vértes

KFKI Atomic Energy Research Institute

P.O.Box 49, Budapest, Hungary H-1525

maraczy@sunserv.kfki.hu

ABSTRACT

The purpose of the KARATE code system is the calculation of neutron physical and thermal hydraulic processes in the VVER reactor core under normal, start-up and slow transient conditions. The code system is used routinely by the Paks Nuclear Power Plant. A number of changes have been made recently to apply ENDF/B-VI based nuclear data, to model reliably the connecting part of the follower and the absorber which has an important role in calculating the maximum linear pin power in VVER-440 cores. Validation results of the code system are illustrated using Monte Carlo calculations and measurements on the ZR-6 critical assembly and the Paks NPP. Some special features of the KARATE system are also described as the ability to handle assemblies with burnup tilt, possibility of calculating kinetic parameters via special perturbations and the application of the code for reactors operating in supercritical thermohydraulic conditions.

1. THE KARATE CODE SYSTEM AND ITS VALIDATION FOR VVER-440 REACTORS

The system consists of three calculational levels: the multigroup transport calculations of diffusion type constants and control rod albedoes, the fine mesh diffusion calculations inside the assemblies and the reflector regions and the nodal calculations of the reactor core. Joint neutron physical and thermal hydraulics calculations are performed in KARATE. The general principles are as follows:

- Three-level modeling: cell, assembly and global calculations make up KARATE. The cell level includes pin cell neutron physics and the fuel model. The assembly level comprises fuel assembly neutron physics and subchannel analysis. The global level involves a nodal neutron physics code and channel analysis.
- Bi-directional calculations are available: boundary conditions can be derived for fine mesh calculations and group constants are prepared for the next, coarser level.
- Consequent parametrization: the group constants are placed in parametrized libraries and the

library parameters are selected after a careful analysis. Those parametrized libraries connect the levels.

The aim of the multigroup calculations is the generation of the few-group fine-mesh diffusion-type constants for each type of the lattice cells of the further fine-mesh assembly calculations. The multigroup calculations are performed on the basis of the ENDF/B-VI nuclear data library. A region containing a number of fuel cells and heterogeneities can be calculated by solving the multigroup collision probability equations by the MULTICELL code. The collision probabilities are obtained as a combination of the cell transmission and escape probabilities and the collision probabilities inside the cylindrical cells. The MULTICELL code is used to generate few-group cross section libraries. For the assemblies with heterogeneous surroundings the non-asymptotic behavior is assured artificially by means of an extra layer outside the assembly. This way the parameter SI (Spectral Index) can be changed. Figure 1 shows the advantage of using SI as a parameter over the application of asymptotic 4-group diffusion constants at the reflector of the ZR-6 47/3 critical assembly.

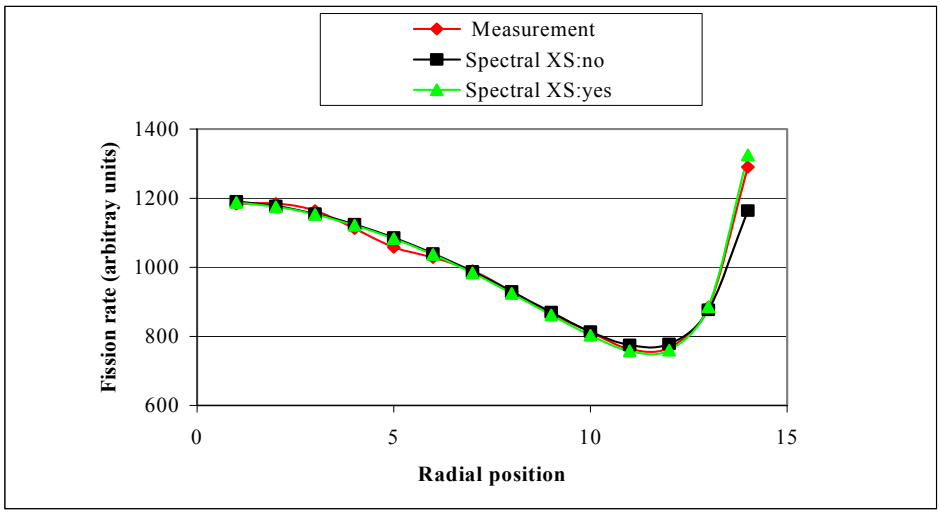

Figure 1

Measured and calculated fission rate distribution of the ZR-6 47/3 critical configuration along a line from the core center to the reflector.

The COLA code is applied for the cells containing absorber or water hole surrounded by fuel cells. Generalized first-flight collision probabilities taking into account the linearly anisotropic scattering are applied in cylindrical geometry. Boundary condition is a prescribed multigroup flux distribution on the boundary of the calculated domain. The COLA code is used to determine control rod albedoes when the cylindrical geometry is applicable, but to model reliably the thin slices of the connecting part of the follower and the absorber Monte Carlo calculations were also applied.

The fuel assembly characteristics can be calculated by solving the two-group fine-mesh diffusion equation in hexagonal geometry for the given assembly or for the assembly and its surroundings (one third of each neighbor assembly) with the SADR code. The irregularity between the assemblies is also taken into account in the difference scheme. The boundary condition can be the traditional white boundary condition on the edge of the fuel assembly, or it can be a given flux distribution on the boundary of the calculation. The latter possibility allows the determination of the assembly response-matrix. In this case it is reasonable not to choose the boundary of the calculation just on the edge of the assembly. This type of boundary condition leads to an inhomogeneous set of equations with prescribed k-eff instead of the traditional eigenvalue problem. The aim of the fuel assembly calculations is twofold. In the first type of the calculations one provides the global calculations with two-group homogenized constants or response-matrices, whereas the second one determines the pin power peaking factor using the results from the global calculations in the form of boundary condition.

In the first case only one axial layer is calculated with predetermined thermohydraulic characteristics of the coolant. The burnup may be calculated pin by pin. Applying proper boundary conditions during the burnup calculations the possible burnup tilt of the assembly can be simulated.

The second type assembly calculation obtains the boundary condition from the global calculation and solves an inhomogeneous problem with a given flux on the boundary. All axial layers are calculated and the third direction is taken into account by an axial buckling originated also from the global calculation. The burnup and the concentrations of some isotopes are calculated pin by pin. In this way the complete history of one assembly can be followed in details.

To calculate the fluence of the reactor vessel three methods were elaborated to couple the KARATE system and the MCNP Monte Carlo code: surface boundary condition at the edge of the core, surface boundary condition inside the core and volumetric source in the core. The advantages and disadvantages of these coupling methods together with calculational results can be found in [1].

A new algorithm realized in the COREMICRO module has been elaborated to calculate the pin-wise power distribution over the whole horizontal cross section of the reactor core. The layer characteristics are calculated by solving the few-group fine-mesh diffusion equation in hexagonal two-dimensional geometry for the given area and its surroundings (reflector). All axial layers are calculated and the third direction is taken into account by an axial buckling originated from the global calculation.

A benchmark problem was formulated to consider the characteristics of the VVER-440 fuel assembly with enrichment zoning. The aim of the benchmark is the study of the space dependence of the power distribution near to a control assembly. It is important to model reliably the connecting part of the absorber and the follower, which can cause high power peaking in the nearby fuel pins. The investigated system of the benchmark is an infinite core made up from VVER-440 assemblies with enrichment zoning and a partially inserted control assembly. The infinite array is to be modeled using mirror boundary conditions on the vertical faces of the basic unit of the infinite core. The horizontal cross section of this basic unit is shown in Figure 2.

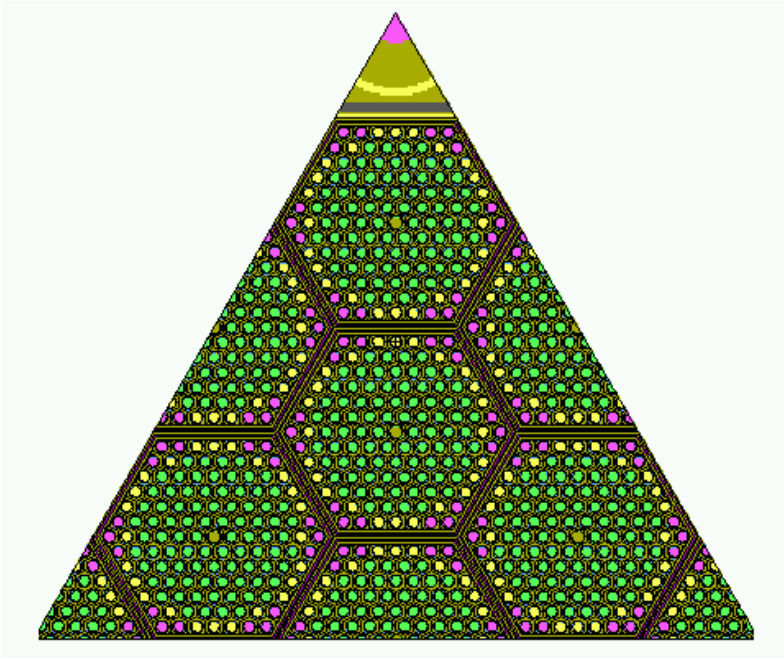

Figure 2

Horizontal cross section of the basic unit of the VVER-440 control assembly benchmark.

The peaking factors for fuel pins next to the absorber can be seen in Figure 3 as a function of axial position. The boron steel absorber is inserted in layers 33-40. The power peaking in layers 27-28 is the consequence of the neighbor connecting part with high water content.

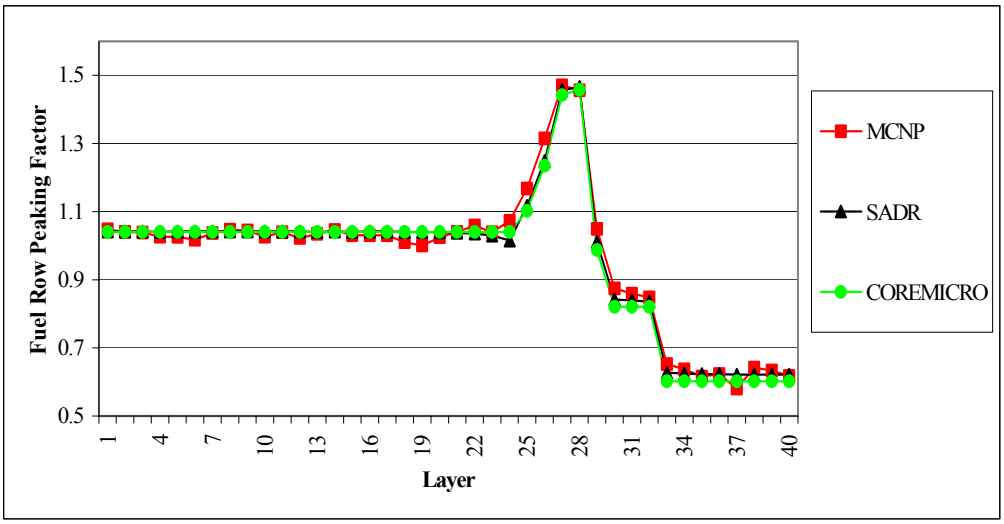

Figure 3

Comparison of peaking factors for fuel pins next to the absorber as a function of axial layers. VVER-440 absorber benchmark calculation.

The global reactor neutron physics calculations are performed in two energy groups by means of the GLOBUS 3D nodal code in hexagonal geometry. The nodes are the assemblies subdivided into axial layers. The unknowns are the partial currents on each node boundary [2]. The nodes are described either by homogeneous cross-sections or by response-matrices available from the assembly calculation. In the first case the response-matrices, which couple the partial currents for a given node, are calculated analytically.

The continuous insertion of control assemblies in the KARATE calculations requires the use of mixed nodes with different material properties. For this purpose a response matrix mixing method was elaborated based on assumptions on the partial current distributions on the side faces of the hexagons resulting an equivalent response matrix [3]. This method enabled applying the original iteration scheme with high speed. Figure 4 shows the continuity of k_{eff} near to the 180 cm rod position.

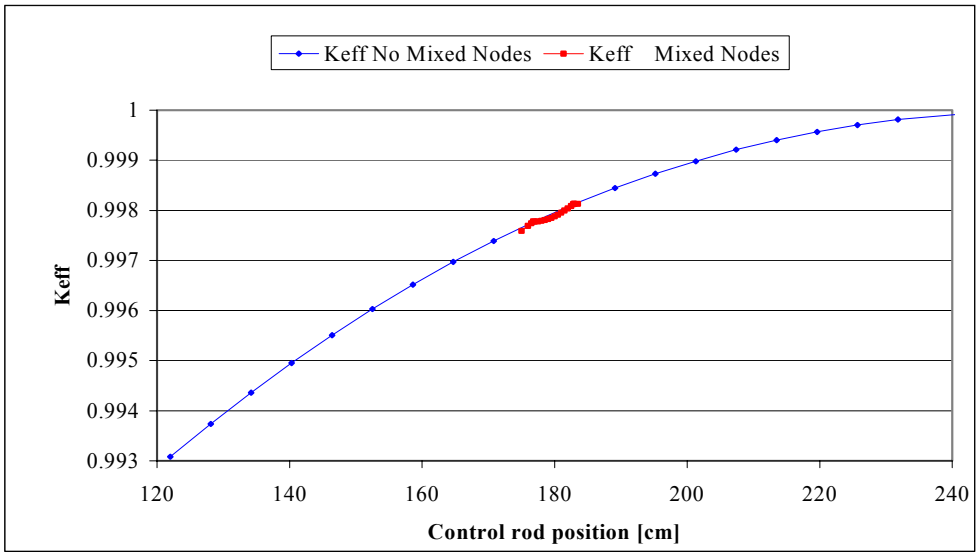

Figure 4

Effective multiplication factor vs. control rod group position.

Using the ENDF/B-VI based few group libraries the critical boron concentrations of the Paks NPP units were reproduced with good accuracy without any cross section adjustment. A typical boron letdown curve for cycle 10 of unit 3 can be seen in Figure 5.

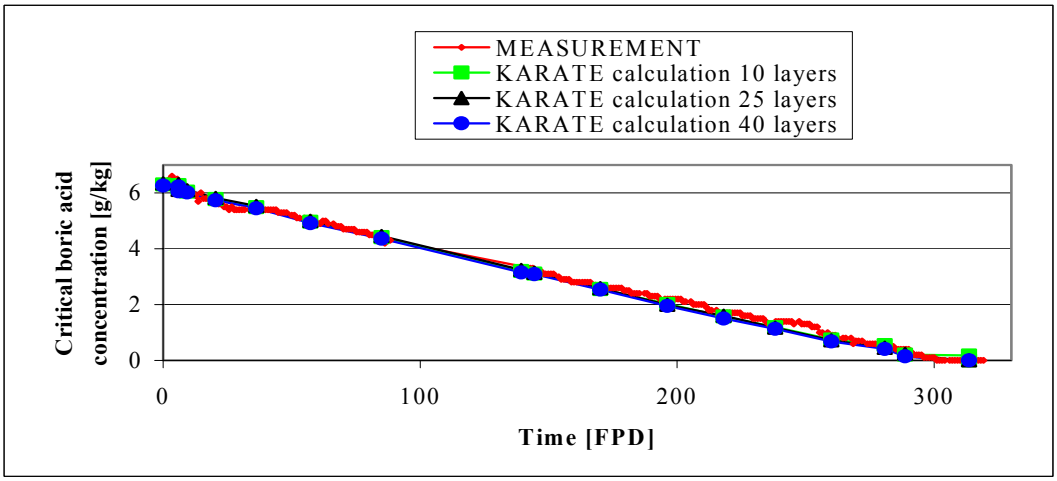

Figure 5

Measured and calculated critical boron concentration of Paks unit 3 cycle 10

The KARATE code system is validated by reproducing physical start-up measurements. The measured ionization chamber signals were converted to differential rod worth using 3 different evaluation methods. The measured and calculated differential control rod worth for cycle 15 of unit 3

shows good agreement in Figure 6.

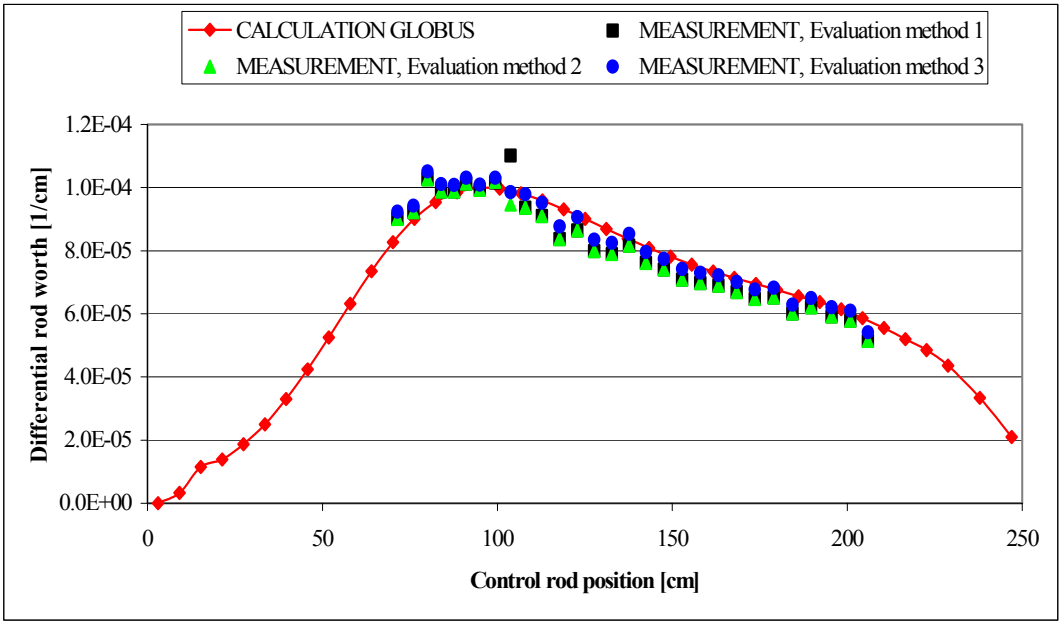

Figure 6

Measured and calculated differential control rod worth for Paks unit 3 cycle 15.

2. SPECIAL APPLICATIONS OF THE KARATE CODE SYSTEM

2.1 RESPONSE MATRIX WITH BURNUP TILT

In the original version of the KARATE nodal calculations the response matrices were supposed to be symmetric. To assess the effect of burnup tilts caused by non-symmetric environment during the irradiation of assemblies the fine mesh burnup calculations were carried out applying not only symmetric but also asymmetric boundary conditions. The asymmetry of boundary conditions was limited to the cos(α) and sin(α) Fourier components, where α is the angle from axis X (see Fig. 7).

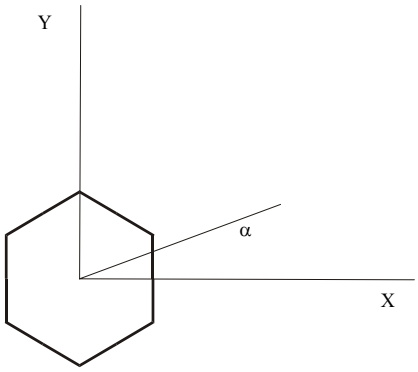

Figure 7
Orientation of the hexagon.

The response matrix taking into account the burnup tilt is approximated as

$$\Gamma = \Gamma_0 + \Delta\Gamma_X + \Delta\Gamma_Y$$

where Γ_0 corresponds to the symmetric burnup, $\Delta\Gamma_X$ and $\Delta\Gamma_Y$ are correction terms. $\Delta\Gamma_X$ is calculated from the burnup calculations where the cos(α) type boundary condition was used. In this case the burnup distribution is symmetric to axis X. For $\Delta\Gamma_Y$ sin(α) applies, which results in burnup distribution symmetric to axis Y. For the characterization of the burnup tilt the time integrals of the cos(α) and sin(α) flux components were applied which can be evaluated in the nodal calculations. As a result of comparative calculations the application of the usual response matrices leads to negligible error. The maximum error of the assembly power distribution was only 0.8 %.

2.1 DETERMINATION OF KINETIC PARAMETERS FROM EIGENVALUE CALCULATIONS

As β_{eff}, the effective delayed neutron fraction plays an important role in reactivity transients it is necessary to calculate it with sufficient accuracy. Some problems come up using the bilinear definition of β_{eff}. The calculation needs flux and adjoint flux detailed in space and energy [4], and special combination of cell level and global results is necessary. A possible method to avoid these problems is the calculation of the delayed neutron fraction as a special reactivity coefficient [5]. The eigenvalue equation for an unperturbed reactor:

$$L\,\Phi = \frac{1}{k} P\,\Phi$$

where L , the loss operator contains neutron streaming, absorption and scattering and P is the

neutron production operator. Let us consider a perturbed system, where

$$\delta P = \alpha \chi_{i,m}(E) \int dE' \upsilon \Sigma_{f,m}(r, E')$$

is added to the production operator. Here α is the amplitude of the special perturbation, $\chi_{i,m}(E)$ is the emission spectrum of delayed neutrons in delayed neutron group i emerging from fission of actinide m, $\upsilon\Sigma_{f,m}(r,E)$ is the macroscopic neutron production cross section of actinide m. Using the formulae of first order perturbation theory it can be shown that the ratio of the effective and nuclear delayed neutron fraction in delayed neutron group i emerging from fission of actinide m is the following expression:

$$\lim_{\alpha \to 0} \frac{1}{\alpha} \left(\frac{k(\alpha) - k(0)}{k(0)} \right) = \beta_{i,m}^{eff} / \beta_{i,m}$$

Here k(0) is the k eigenvalue of the unperturbed system, k(α) belongs to the specially perturbed case with amplitude α. Similarly, for the calculation of Λ , the prompt neutron lifetime the

$$\delta L = \alpha \frac{1}{V}$$

perturbation is applied, where $\frac{1}{V}$ is the reciprocal neutron velocity. Regarding k as a function of the α amplitude and approaching 0. with the α amplitude, the result is the definition of the prompt neutron lifetime:

$$\lim_{\alpha \to 0} \frac{1}{\alpha} \left(\frac{1}{k(\alpha)} - \frac{1}{k(0)} \right) = \Lambda$$

The method was applied consistently in the KARATE program system in order to check the time saving β_{eff} calculation scheme. On the level of the assembly calculations the MULTICELL code was applied. At fixed technological parameters the unperturbed state and the perturbed states necessary for the determination of all the delayed neutron fractions and the prompt neutron lifetime were calculated. The corresponding diffusion type 2-group XS sets were parametrized and subsequently used in the global calculations. Sensitivity analysis had to be carried out because at low α values loss of digits occur and at high α values the perturbation method fails. Intermediate α values give appropriate perturbation as can be seen in Figures 8-9.

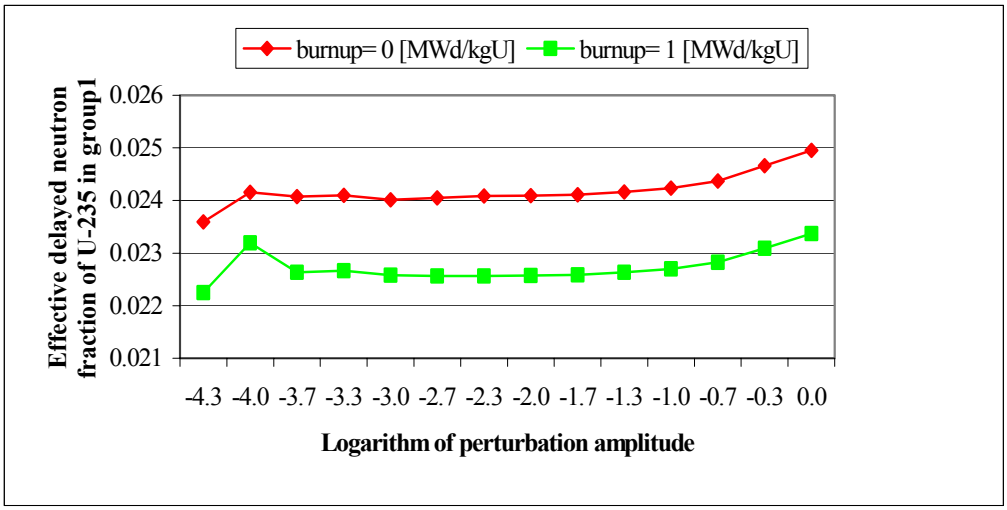

Figure 8

Calculated β_{eff} as a function of perturbation amplitude in the production term of the transport operator.

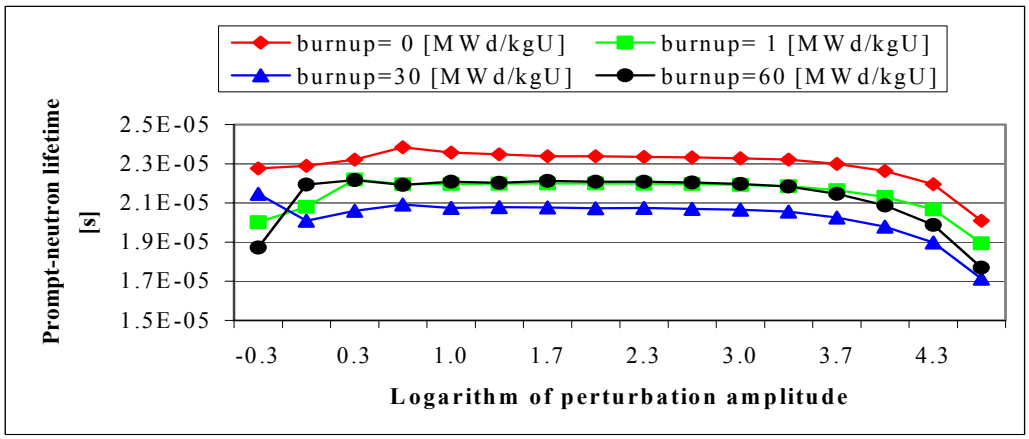

Figure 9

Calculated prompt neutron lifetime as a function of perturbation amplitude in the absorption term of the transport operator.

As a result of the detailed analysis using ^{235}U delayed neutron spectra instead of the isotope wise spectra practically did not affect the value of β_{eff}. Using the high precision method less than 1. % difference in β_{eff} was observed compared to the original time saving KARATE method, where the 2-group adjoint flux is not available.

2.2 HIGH PERFORMANCE LIGHT WATER REACTOR (HPLWR) CALCULATIONS

The overall objective of the HPLWR project is to assess the feasibility of a high efficiency Light Water Reactor operating at thermodynamically supercritical region [6]. In a once-through concept, the water enters the reactor as water and exits as high-pressure steam without change of phase. An efficiency of over 40% is expected. The evaluation and improvement of the Japanese concept is carried out by eight institutions within the 5th Framework Programme of the EC. Our role is to perform neutronic transport and core diffusion calculations. The water density at the top of the HPLWR assemblies is very low, so to reach sufficient moderation in the tight-lattice hexagonal assemblies the application of water rods extending to seven hexagons is necessary (Figure 10)

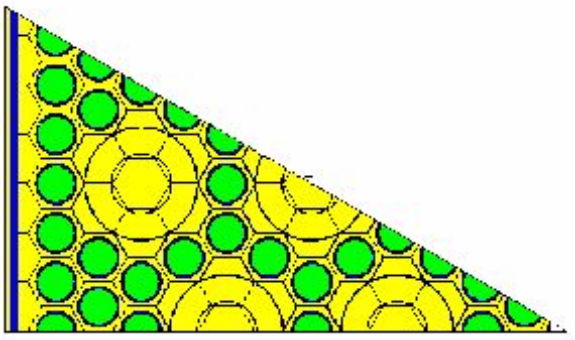

Figure 10

30^o sector of the HPLWR assembly.

Preliminary calculations have shown that the maximum k_{eff} deviation of our deterministic transport code from Monte Carlo results does not exceed 0.9% even in case of 0.096 g/cm^3 water density. As the coolant density along the axial direction shows remarkable change, coupled neutronic-thermohydraulic calculations are essential which take into account the heating of moderator in the special water rods of the assemblies. We have prepared a parametrized diffusion cross-section library of the HPLWR assembly at low burnup. The parameter range covers the cold zero power and hot full power states. The feedback parameters are calculated with the SPROD code of the Tokyo University, which was coupled with the KARATE core calculation module. Preliminary core calculations have been done to demonstrate the applicability of the code. The axial density distribution of the coolant, the water rod and the water insulator between them can be seen in Fig. 11.

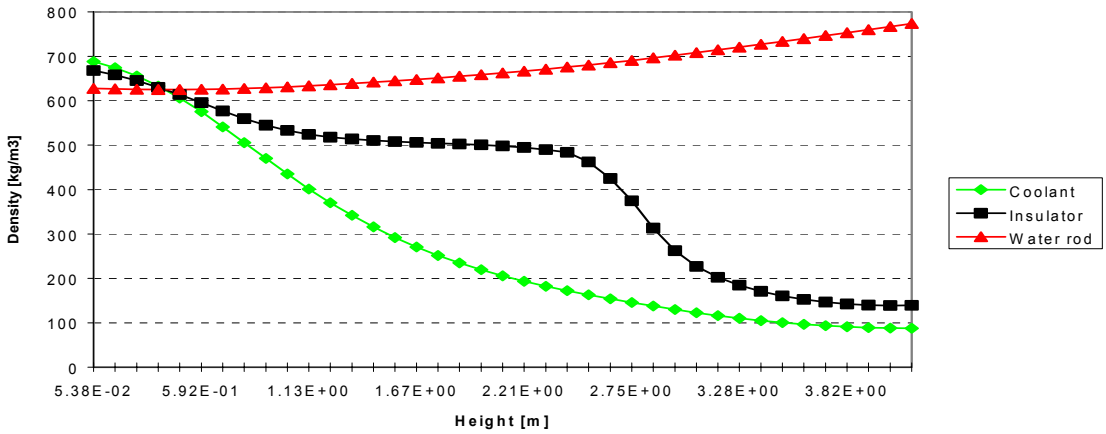

Figure 11

Axial water density distributions of the HPLWR assembly under nominal conditions.

CONCLUSIONS

The complicated follower and absorber joint which has an important role in calculating the maximum linear pin power in VVER-440 cores was successfully modeled according to MCNP benchmark calculations. Taking into account the burnup tilt inside the VVER-440 assemblies does not affect the calculated assembly wise power distribution significantly. The verification and validation results of the code system show that KARATE can be applied reliably for the calculation of VVER type reactors.

ACKNOWLEDGEMENTS

The HPLWR Monte Carlo calculations were carried out by Frej Wasastjerna and Aapo Tanskanen, VTT Finland.

REFERENCES

1. E. Temesvári, Gy. Hegyi, G. Hordósy, A. Keresztúri, Cs. Maráczy: “Neutron Fluence Calculations for the Reactor Vessel”, *International Meeting on Mathematical Methods for Nuclear Applications,* Salt Lake City, USA, September 9-13, 2001
2. M. Makai, "Response Matrix of Symmetric Nodes", *Nuclear Science and Engineering*, **86**, p. 302 (1984)
3. G. Hordósy, A. Keresztúri, Cs. Maráczy, "Methodological Studies on the VVER-440 Control Assembly Calculations“, *Internal Conference on Mathematics and Computations, Reactor Physics and Environmental Analyses*, Portland, Oregon, USA, April 30-May 4,1995 p.1049

4. G. I. Bell, S. Glasstone, *Nuclear Reactor Theory*, Van Nostrand Reinhold Company, City & Country (1970)
5. Ch. Decurnex et al., "Direct Calculation of Kinetic Parameters", *Proceedings of the International Conference on Reactor Physics and Reactor Computations*, Tel-Aviv, January 23-26, 1994 pp. 504-512 (1994).
6. D. Squarer, Y. Oka, D. Bittermann, N. Aksan, C. Maraczy, R. Kyrki-Rajamaki, A. Souyri, P. Dumaz, "High Performance Light Water Reactor (HPLWR)", *FISA-2001 EU Research in Reactor Safety, Mid-Term Symposium on Shared-Cost and Concerted Actions*, European Commission, Luxembourg 12-14 November 2001