ABSTRACT

In the first step the Serpent-PARCS coupling codes, applying Monte Carlo based group constants in a 2-D reactor core nodal diffusion calculation is demonstrated. The results were compared to detailed full scope Monte Carlo results, where the average difference of power distribution was 1.7% for specific core loading pattern.

Based on a lesson learned from the Serpent homogenization procedure a 3D model of nuclear reactor core and quasi 3D model of reactor pressure vessel was developed in computer program Apros. The goal is to perform transient reactor kinetic calculations with reactor coolant response. First the 2D homogenization of cross-sections was calculated in Monte Carlo code Serpent. The calculation is done for different types of fuel in terms of enrichment, burnable poisons and burnup steps. The corrections (reactivity coefficients) are calculated for changes in boron, fuel temperature and moderator temperature (density) change. Core reflector is modelled using albedo coefficients. The steady state simulation at full power was run in Apros. Two analyses are performed: symmetrical control rod bank insertion and asymmetrical insertion of one control rod.
1 INTRODUCTION

To accurately predict the behaviour of light water reactors the determination of power distributions, control rods' worth, shutdown margins, and isotopic depletion rates must be known throughout the reactor cycle. The ability to perform such core-follow calculations depends critically on models employed to predict the free neutron density in space, direction and energy. If thermal-hydraulic properties of the reactor and fundamental nuclear data are assumed to be known, the reactor physicist is faced with the straightforward task of solving the three-dimensional neutron transport equation. Unfortunately the complexity inherent in explicit modeling of every fuel pin, control rod, burnable poison rod, and water channel limits the direct methods of solving the three-dimensional transport equation. Although tools such as three-dimensional continuous energy Monte Carlo are available, the magnitude of the computational problem posed by explicit modeling is such that even the most sophisticated digital computers are incapable of determining reactor parameters, with the possible exception of multiplication factor for steady state model. Additional, the deterministic neutron transport methods are similarly overwhelmed by the complexity of the computational problem of explicit geometrical core modeling.

Many reactor analysis methods circumvent the computational burden of explicit geometrical modeling by coupling geometrically-simple, energy-intensive calculations with few-group, geometrically-complicated calculations using homogenization and group condensation. In previous research the Serpent-GNOMER code sequence application for design calculations of PWR's [1], [2], [3] was conducted. The Serpent-GNOMER system has been validated for nuclear design calculations of Krško NPP. This approach was developed from the existent CORD-2 [5] simulator, where the existing WIMSD-5B code [4] for lattice cell calculations was replaced with Monte Carlo code Serpent 2 [6]. The homogenization procedure was developed using the same methodology used in the CORD-2 system, namely the effective diffusion homogenization method (EDH), to obtain the cell homogenized cross sections using Monte Carlo method and to use them in fuel assembly calculations and finally the full core calculations were performed using GNOMER code. The positive outcome of this study was the satisfactory performance of the EDH method using Monte Carlo method at hot zero-power conditions. One aspect that was not satisfactory was the long computational time. Usually one fuel cycle analysis using WIMSD-GNOMER takes 3 hours using personal computer with 1 CPU while Serpent-GNOMER calculation takes 14 days using one computer cluster node with 20 CPU’s.

In this paper we present the development of an improved methodology that could be used for 3-D full core calculations of recent (generation II) and future (generation III and IV) power reactors. In the first part, the results of development of a two-step procedure are presented. The results are focused on using Serpent 2 and PARCS to generate group constants using fuel assembly model and to obtain a full core solution for zero power conditions, where no reactivity feedbacks and burnup calculations are considered. Therefore the motivation and the goal was to obtain the fuel assembly model that is capable to model any kind of fuel configuration using existent and future fuel designs.

In the second part the Serpent-Apros simulation is presented using knowledge that we have gained in the first part. Serpent 2 is utilized for the generation of burnup-dependent cross section data for the 3D nodal core simulator code APROS [10] and perform the static and dynamic simulation using fresh core configuration on Krško NPP.
2 SERPENT-PARCS COUPLING

Serpent use a Monte Carlo method which can be characterized as a brute-force calculation technique, which is particularly well-suited for complicated problems. The simplicity and the potential to produce very accurate results are indeed amongst the most attractive features of the Monte Carlo method. Therefore the model construction is straightforward since the geometry is modelled explicitly.

The main purpose was to study the group constant generation for deterministic reactor simulator codes, which require a certain level of specialization and most of the widely-used general purpose transport codes are not capable of performing this task without extensive modifications. The main difficulty in the generation of homogenized group constants using Monte Carlo code is the calculation of the neutron diffusion coefficient. This constant is based on an approximation and it is essentially the only parameter without any continuous-energy counterpart in Monte Carlo calculation. Homogenization and infinite lattice calculations are closely related to neutron leakage models, which are artificial corrections needed to account for the radial and axial streaming of neutrons across the geometry boundaries. We have tested several methods for the treatment of neutron leakage. The Serpent-PARCS coupled model gives the best results. The difference of the assembly power distribution between the Serpent Monte Carlo results and coupled code Serpent-PARCS are shown on Figure 1. The results were tested on BEAVRS benchmark [7].

![Figure 1: Case 1 comparison between MC Serpent and Serpent-PARCS: R(max)=5.06% in R(avg)=1.86%](image)

3 APROS CALCULATION

3.1 Model description

The nuclear power plant Krško (NEK) was modelled. NEK is a 2 loop Westinghouse PWR. The complete reactor coolant system including: reactor coolant pump, steam generators, pressurizer, reactor pressure vessel and reactor core was developed in Apros. The details are not described in this paper [8].

For this study a 3D Apros model was developed, with fresh cycle with 121 fuel assemblies consisting of a 16×16 rod arrangement was analysed. Each assembly is composed of 235 fuel rods and 21 positions filled with water. The fuel rods are constructed of Zircaloy-4 cylindrical
tubes containing slightly enriched UO$_2$ fuel pellets. This loading pattern contains three regions, each with different enrichment: 2.1, 2.6 and 3.1 wt.\% of U-235. In addition, the burnable poison rods (BPR) in the form of Pyrex glass containing 12.5 \% B$_2$O$_3$ were used for partial control of excess reactivity. There were a total of 512 burnable poison rods inserted in the core in 3 configurations: 8, 12 and 16 absorber rods per assembly. Each fuel assembly having 235 fuel rods has been subdivided in 10 axial volumes, thus also the blanket of each one can be taken into consideration. There are 33 rod cluster control assemblies, or control rods. The absorber material in the control rods is an alloy of 80\% silver (weight percentage), 15\% indium and 5\% cadmium. The control rod cladding is made out of type-304 stainless steel. The control rods are arranged into A, B, C, D, as for the shutdown rods, those are arranged into Banks SA and SB.

Heat transfer modules are connected to six-equation thermal hydraulic models with their own heat conduction solutions. Calculation of fuel rod temperatures is performed in the thermal hydraulic part of Apros. The fuel rod is described as a solid heat structure consisting of three materials: fuel, fuel-cladding gap and cladding. One-dimensional heat conduction solution in fuel rod is calculated using ten radial nodes. Specific material properties are given for the fuel pellet, gap and cladding without the need of re-compiling the code.

The three-dimensional nodal neutronics model of Apros is based on the two-group neutron diffusion theory in homogenized nodes. Solution of the neutron diffusion equations relies on an advanced nodal method. Albedos in radial direction are calculated using 2-D full core model, where different albedos are defined in the outer region of the core model. The following parameters is calculated in Serpent and is input to Apros: fast diffusion coefficient, fast absorption cross section, fast fission production cross section, fast fission cross section, slowing down cross section into thermal group, thermal diffusion coefficient, thermal absorption cross section, thermal fission production cross section, thermal fission cross section, average energy per fission, fast inverse velocity of prompt neutrons, thermal inverse velocity of prompt neutrons, fraction of delayed neutron group 1-6, slowing-down length, diffusion length, and migration area.

### 3.2 Steady state calculation

The steady state calculations were performed to check the stability of 3D Apros model. The relative thermal power is presented at Figure 2. The relative power distribution shows reasonable agreement with reference CORD-2 results (Figure 3). The relative power of fuel elements on the blue line on right part Figure 3 is plotted. Computer program CORD-2 is taken as reference result, since it is used for verification of core design in NEK. The power at the edge of the core is too high in Apros calculation for about 40 \%. This error is due to inadequate reflector modelling. The power in the middle of the core is 9 \% lower in Apros then in CORD-2, and elsewhere the difference is less then 5 \%. However the Apros is a simulator, where the basic assumption has been that only dynamic calculation is used, which is presented in chapter 3.3.
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Figure 2: Radial (middle height of core) and axial (middle cross section) relative thermal power at full power.

Figure 3: Relative power of fuel elements in Apros and CORD (reference value) at the hot full power. The fuel elements at the middle radial cross-section of the core are shown.

3.3 Symmetrical rod insertion

The 10% step load decrease (BOC) is analysed and compared with point kinetic results obtained with LOFTRAN, Relap5 and Apros 3D and point kinetic model [5]. The position of control rods (Figure 4) is compared with RELAP, Apros and point kinetic model results. The control rod position in 3D Apros simulation agrees well with Relap5. The relative nuclear power in Apros 3D shows no undershot at the beginning compared to Relap5. The final power in 3D Apros is 91.76% and is only slightly higher compared to power in Relap, which is 91.46%. The results are better than in Apros point kinetic model. The LOFTRAN results show even larger undershot, however the model differ significantly: the moderator and fuel reactivity coefficient are not taken into account.
Figure 4: Fission power vs. time in simulation with control rod bank insertion.

3.4 Asymmetrical rod insertion

The asymmetrical situation with one control rod fully inserted was also calculated for demonstration purposes. From Figure 5, we can see the effect of one control rod insertion to entire core. The relative power of fuel assembly where the control rod is inserted is significantly reduced. The relative power in surrounding fuel assemblies is also reduced.

Figure 5: Radial (middle height of core) relative thermal power in simulation with one control rod insertion.

4 CONCLUSION

The Monte Carlo method is widely used for solving various neutron transport problems encountered in nuclear reactor physics. Monte Carlo codes are also used for providing reference results for the validation of deterministic transport codes. The common factor in all applications is the need to model the geometry and the interaction physics to within maximum accuracy and detail.
The potential to produce accurate results without major approximations is clearly the most significant advantage of the Monte Carlo method. Neutron interaction data collected in the evaluated nuclear data files can be used in a continuous-energy format, without the micro-group condensation necessary for all deterministic transport methods. Resonance self-shielding and other collective phenomena are consistently modelled without additional effort. The development of computer capacities and the growing interest in parallel calculation suggest that the importance of Monte Carlo calculation will only increase in the future, especially since the method is particularly well-suited for parallelization.

New applications will probably arise along with code development, but it may take decades before Monte Carlo codes start to compete with deterministic transport methods in coupled full-core analyses. The presently-used reactor simulator codes are typically based on few-group nodal diffusion methods. Traditionally this task was handled by second-generation lattice codes, based on advanced deterministic transport methods. However we have shown in this report that it is becoming increasingly challenging, since more advanced fuel like MOX fuels and next-generation reactor concepts may require the use of more elaborate transport methods, and Monte Carlo calculation seems like a viable choice.

Since the homogenization problem is basically reduced to the calculation of the neutron diffusion coefficient, which is the only parameter without any analogy in the continuous-energy Monte Carlo calculation and a very fundamental problem in the methodology is that the group constant data is produced in a two-dimensional infinite-lattice calculation, which is inconsistent with physical reality suggested that many studies had to be performed to validate the results with deterministic codes. Based on the results presented in this report it was decided that the use of stochastic method is being used for group section generation for future nodal simulator.

We have used the Monte Carlo code Serpent 2 to calculate burnup-dependent cross section data for the Apros nodal reactor core simulator of Krško NPP. The steady state results are overall in good agreement with the reference data, except for core edge of the core. In the future the reflector model will have to be adjusted to have more realistic neutron flux at the edge of the core. We have successfully demonstrated dynamic simulation using control rod insertion into core, where the 3D Apros results are in good agreement with the RELAP5 results.

In the symmetrical rod insertion we have demonstrated that the overall response of reactor coolant system is very similar to the well validated RELAP5 code results. The calculation speed is about 3-times slower than physical time, which means that analyses can be done in timely manner and with some adjustment, also used for real time simulator. With the 3D Apros model the reactor kinetics can be simulated in almost real time on single PC.

The asymmetrical case with on control element insertion was used for demonstration of capabilities of developed 3D Apros model. Due to lack of reliable measurements or reference calculations no comparison was shown.

The 3D reactor kinetic model is coupled with thermal-hydraulic model of reactor coolant system, which include reactor coolant pump, steam generator and pressurizer. Such model can be used to simulate system transients and design basis accidents. The anticipated transients without scram are of special interest, since the reactor trip does not occur and the reactivity feedbacks can be analysed.

REFERENCES


