20th International Conference
NUCLEAR ENERGY FOR NEW EUROPE 2011
Bovec /Slovenia /September 12-15

Implementation of the Neutronics Model of HEXTRAN&TRAB-3D into APROS

Jukka Rintala
VTT, Technical Research Centre of Finland
Tietotie 3, FI-02150 Espoo, Finland
Tel: +358 40 521 9567, Email: jukka.rintala@vtt.fi
http://www.vtt.fi

ABSTRACT

APROS - Advanced PROcess Simulation environment – is a widely used simulation tool by VTT and FORTUM for nuclear power plant modelling. The three-dimensional neutronics calculation in APROS has been based on a model using the finite difference method. The model was originally intended also for power plant simulator applications, where it fits well because of its speed. For safety analysis purposes in challenging cases from neutronics point of view, however, a new model was considered to be an important improvement to conform to the increasing accuracy requirements. A sophisticated nodal model used already in HEXTRAN and TRAB-3D safety analysis codes by VTT, was decided to be implemented into APROS. The hexagonal part of the model has now been implemented and tested. For practical reasons, the model, with several improvements was completely re-programmed into APROS, and thus, an extensive validation program was necessary to prove the correct behaviour of the model. The paper presents the physical base of the method and outlines the key points of the implementation. At the end, some results of the latest validation calculation are shown.

1 INTRODUCTION

APROS (Advanced PROcess Simulation environment) [1] is a multifunctional software for simulation of industrial processes, developed by VTT, Technical Research Centre of Finland and Fortum power and heat Ltd. A model in APROS can be constructed freely, using extensive selection of process components for the processes with enormous variability in type and size. With sophisticated physical models in a calculation level, this allows an accurate simulation of entire production plants, including the process and automation. APROS is a widely used analysis tool for nuclear power plants in Finland and abroad. The possibility to consider all the main components of the nuclear power plant under a detailed simulation at a same time, gives more freedom and reliability to all the analyses.

The three-dimensional neutronics calculation with a good thermal-hydraulics model gives a tool to analyze all type of transient scenarios of nuclear reactor in a realistic way. The original three-dimensional neutronics model in APROS, based on the finite difference method, is known to have difficulties to perform accurately enough in the most demanding safety analyses from the core neutronics point of view. In developing the original model, the calculation time was a key parameter. Thus, a new sophisticated model was decided to be implemented for safety analysis purposes, where the accuracy was the main target.

Since the reactor dynamics codes HEXTRAN [2] and TRAB-3D [3], had been used successfully for demanding safety analysis of nuclear reactors with hexagonal and square-lattice geometry successively, a similar model was decided to be added into APROS as well. This also helps in validation work of the model. Since programming languages and computers have developed enormously since HEXTRAN and TRAB-3D were originally programmed and the code structure of APROS is fully different to them, the model was completely re-programmed. This allows also better code maintenance and further development. By combining the accurate neutronics nodal solver with the extensive process and automation simulation tools of APROS provides a tool to analyze the whole nuclear power plant with detailed and accurate submodels at one time, which is not possible by HEXTRAN/TRAB-3D. Fully reprogrammed solver with modern programming techniques also improves significantly the possibilities for further development and maintenance.

At the beginning, the hexagonal part of the model was implemented and tested and the square lattice part was taken under work after that. This paper concentrates on the physical base of the model. The improvements to the original codes are outlined and at the end, some validation results are presented shortly to show the capability of the model.

2 PHYSICAL BASIS OF THE NEUTRONICS MODEL

The basis of the method for the hexagonal (HEXTRAN) and square-lattice (TRAB-3D) geometries is the same. Small variation, however, exists in addition to the natural differences arising from different geometries. Since the improvements in the new implementation are mainly related to the control rod follower of VVER-440 reactors in hexagonal geometry, this paper approaches the method also in hexagonal geometry.

The model is a typical sophisticated neutronics model based on two-group diffusion theory in homogenized nodes. Instead of solving the group fluxes directly, the model uses the well-known theory of differential equations of constant coefficients, which shows that the group fluxes are linear combination of two characteristic solutions *f*, also referred as spatial modes, of Helmholtz equation

$$\left(\nabla^2 + B^2\right) f(\bar{r}) = 0. \tag{1}$$

Diffusion equations with trial solution with spatial modes give the coefficients for this matrix equation and setting determinant to zero, one obtains a second order equation for buckling $B_{I,II}^2$ and further for the coefficients of group flux equation

$$\Phi_{1,2}(\bar{r}) = \sum_{i=1}^{2} \varphi_i f_i(\bar{r}). \tag{2}$$

Iteration is divided into two parts. During an outer iteration, spatial mode shapes inside singular nodes are improved. The key idea of the mode decomposition is the practical fact that fundamental mode is slowly varying function, since realistic values in Eq. (1) give a relatively small fundamental buckling. On the other hand, a transient mode buckling becomes larger and negative implying that the transient mode is non-zero only near the boundaries in practice. This gives a possibility to approximate the fundamental mode with a simple function and already the third degree polynomial used in the model gives good results. On the other hand, the transient mode can be handled separately in each boundary approximating it with a function of exponential shape. Approximating directly the group fluxes would be more complicated.

It requires relatively much algebra to reach the final equations, but in practice, the coefficients of the functions approximating the spatial modes are never calculated directly. Instead of that, all the information of the node internal flux shapes are transformed into the form of coupling coefficients, which give a simple relation between the boundary currents and fluxes to the average value of the fundamental mode. The fundamental mode average is chosen, because it is chosen to be iterated later in inner iteration. The boundary value for a neutron current from node A to node B for example is

$$\bar{J}_{1,2}^{AB} = K_{J1,2}^{A} f_I^{A} - K_{J1,2}^{B} f_I^{B}, \tag{3}$$

where $K_{J1,2}^{A}$ is a coupling coefficient from node A to B in fast (1) or thermal (2) energy and f_I^{A} is a fundamental mode average value in node A.

In case of neutron flux, the discontinuity factors bring an additional coefficient. The solution of the coupling coefficient requires enough conditions to fix the function coefficients of the mode shape approximations. The natural continuity conditions of neutron current and group fluxes (with discontinuity factors) bring two conditions at each interface. This is not enough compared to the total degree of freedom. In the model, an additional so called Galerkin's condition is used, which implies that the node integrated residual of Eq. (1) vanishes. With a further approximation of separable fundamental mode function in transverse and axial direction, this gives 2 two more conditions. The rest of the conditions are obtained by setting the first degree terms in the residuals to zero.

After these assumptions are made, the rest of the calculation of the coupling coefficients is basically just algebra. Since the internal mode shapes are relatively slowly varying functions, it is computationally favourable to perform several inner iterations during one outer iteration. In the inner iteration, only the average value of the fundamental mode is recalculated with fixed internal mode shapes, that is, fixed coupling coefficients. The relations in Eq. (2) are used for transforming the summed node integrated diffusion equations to one equation with only fundamental mode average as an unknown.

A neutron dynamics code requires an efficient time discretization and a description of delayed effects. Inside each node, typical approximations of separable flux shape in respect to time and spatial variables during one time step as well as a similar shape of prompt and delayed neutron source are used. These approximations do not lead to a large error compared to the computational advantage. Delayed neutrons are divided into six groups.

For the actual time discretization, a spectral matching method W_{11} is used [4]. It offers an efficient and accurate but also a numerically stable method. With appropriate parameter selection, it takes very accurately into account the time variation of the total power. The discretization method also allows a flexible choice of the time step.

3 IMPLEMENTATION AND IMPROVEMENTS

As mentioned earlier, the model was fully reprogrammed. The totally different code structure between APROS and HEXTRAN/TRAB-3D was alone a reason to do this. Another advantage is a modern programming style, which gives understandable source code. It is easy to understand and develop further.

Since main parts of APROS are programmed in FORTRAN, it was used also for the new model. The FORTRAN 90/95 standard gives enough tools to create a modern modular code, even though it is not a full object oriented language.

During the programming, several small targets for improvements were noticed. Mainly the handling of the VVER-440 control rod follower was implemented partly in a new way. In some cases, the tip of the follower can be near a calculational node boundary and thus be too short to be a node of its own. In such cases, this tip must be added to the next fuel node and in APROS, the neutron current from and to this tip area is calculated in a more physical manner than in HEXTRAN. A much bigger improvement is a dynamically changing fuel node number in the follower. This is important especially to small insertions. In HEXTRAN, relatively big part of the follower below the reactor core must be added to the fuel node inside the core, which can cause relatively big local errors. This problem can be avoided, if the number of fuel nodes can be changed after control rod movement. Furthermore, the model in APROS was developed to calculate correctly, when the core contains fuel assemblies with different height, which must be handled slightly unphysically in HEXTRAN.

4 EXAMPLES OF THE VALIDATION RESULTS

The hexagonal part of the model was implemented first and thus, a validation process for that has been already completed. Validation includes comparison of the results to HEXTRAN in several benchmarks as well as comparison to the measurements data in test reactor. In addition, test calculations with real power plant data from Loviisa NPP have been made.

In Fig. 1, some results of comparison to space-time kinetics experiments carried out in the zero-power test reactor LR-0 in former Czechoslovakia [5] are shown. As a total, five different experiments with slow and fast control rod movement, with the neutron flux measured in several different locations around the core and, with shortened VVER-1000 fuel assemblies, were done.

The results in Fig. 1 are from the first experiment with a slow eccentric trapezoidal control rod movement. On the left, some of the detector rates (C3, C10, C13) are compared to the results calculated by the original model of APROS based on the finite difference method. Results of the same calculation by the new nodal model are presented on the right. The

detectors in experiment were situated around the core and details can be found from references.

The results show the very good accuracy of the new model, but also the level of improvement of the original one. The results for all five tests show the same good agreement of the calculation results to the measurements.

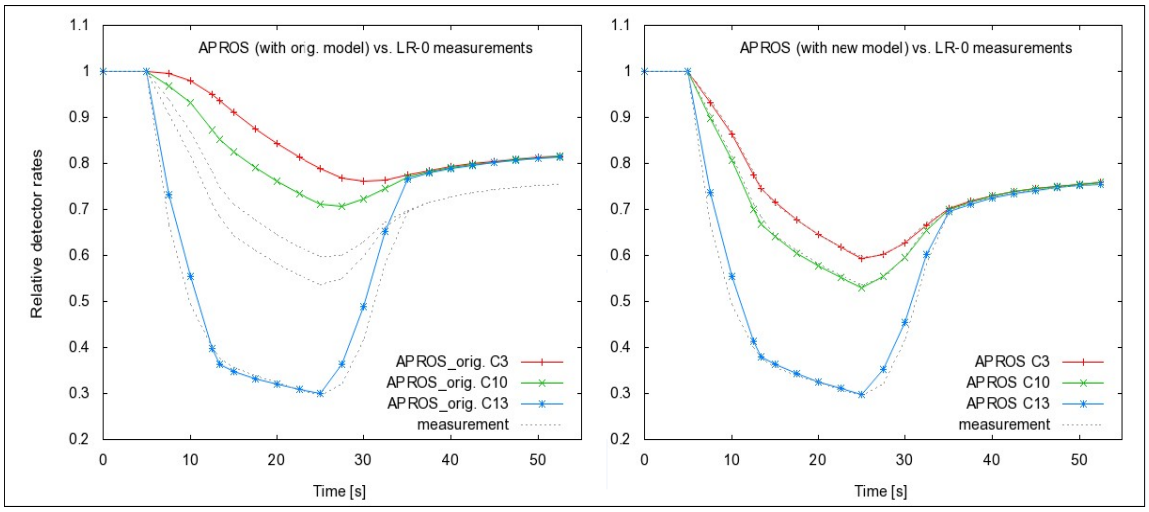

Figure 1: Relative detector rates at detector positions C3, C10 and C13 calculated by APROS with the original and new model are compared to measurements in first LR-0 experiment.

The first validation calculations for the square-lattice are NEACRP 3-D LWR core transient benchmarks [6]. The first PWR transient is a rapid control rod ejection at hot zero power in critical core. Since the neutronics model is similar to the one in TRAB-3D, it is natural to compare APROS to it. In Fig. 2, total fission power versus time has been compared. It can be seen that differences are small, but APROS seems to estimate the power peak slightly earlier and higher. However, small differences in the power peak timing are not alarming in such extreme fast transient. Despite those differences, relative difference in total released energy is below 0.3 % at the end of transient (5 s.).

During the validation of hexagonal model, it was noted that differences in the water property tables between APROS and HEXTRAN/TRAB-3D cause systematically slightly smaller coolant density for APROS. Thus, the calculation was repeated with slightly modified APROS, where the water density was increased by 2.02 kg/m^3. This calculation showed that the difference in the power peak position was mainly related to this difference. In Fig. 3, a comparison of a radial power distribution between the codes at a radial layer 13 of 16 is presented at the moment of power maximum. The results by APROS are calculated with the modified density, so that the time of the power maximum is the same. One can see only small differences especially with the radial variation of the relative differences.

More detailed study shows also only small differences between APROS and TRAB-3D taken into account the different thermal-hydraulics models like the examples above. The results of the first square-lattice validation calculation thus give a good signal of the functionality of the new model also in this geometry. It is also important to see that the thermal-hydraulics models in different codes give relatively similar results. However, more calculations with different type of inputs and scenarios are needed to guarantee the functionality of the square-lattice model and that is currently in progress.

303.6

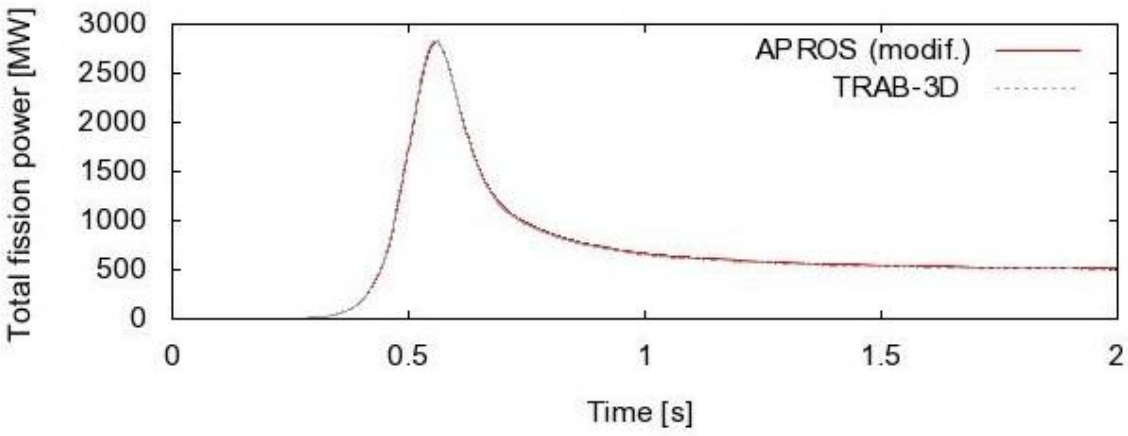

Figure 2: Comparison of total fission power between APROS and TRAB in neacrp prw-transient A1.

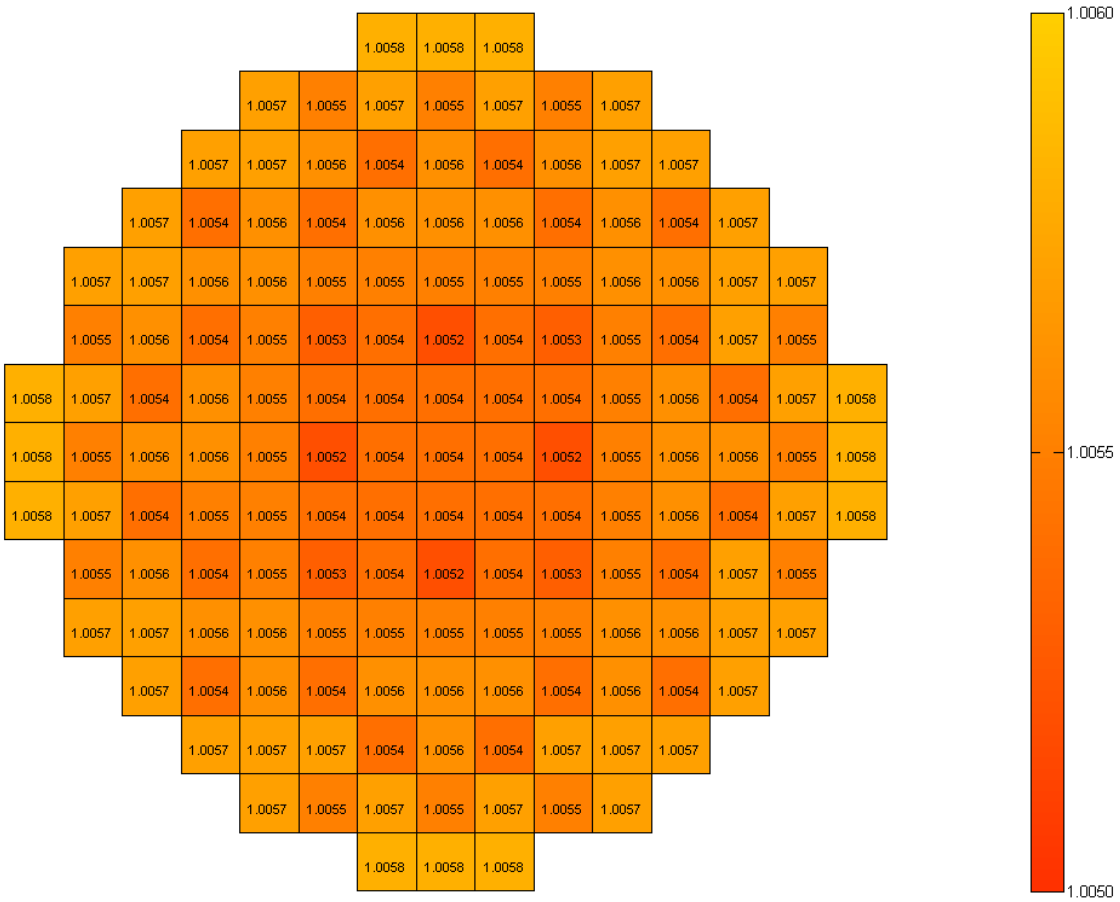

Figure 3: Comparison of radial power at a level 13 of 16 between APROS and TRAB-3D in neacrp prw-transient A1 at the time of maximum power [APROS/TRAB-3D].

5 CONCLUSIONS

The original three-dimensional neutronics model in APROS, based on the finite difference method, was developed also for simulator purposes and constricted by requirements for calculation speed. It is known to have difficulties to perform accurately in the most demanding safety analyses from neutronics point of view. The desire to carry out such safety analyses with APROS, in which the three-dimensional power distribution is important, caused a decision to implement a more accurate model in APROS. The new model uses the same method that is used in HEXTRAN and TRAB-3D by VTT for hexagonal and square-lattice, respectively. The codes with a sophisticated nodal neutronics model have been well validated and widely used in demanding safety analyses. The model was completely reprogrammed into APROS for practical reasons. Also some improvements were added.

The hexagonal part of the model has been now validated already widely and the results show a good agreement with HEXTRAN and measurement data. The validation of square-lattice part has started lately and the first results are also promising.

6 REFERENCES

[1] E. Silvennoinen et al., *The APROS software for process simulation and model developement*, Technical Research Centre of Finland, research report 618, ISBN 951-38-3463-8, 1989.

[2] R. Kyrki-Rajamäki, *Three-dimensional reactor dynamics code for VVER type nuclear reactors,* dissertation for the degree of Doctor of technology, Technical Research Centre of Finland, ISBN 951-38-4784-5, 1995.

[3] Kaloinen, E., Kyrki-Rajamäki, R., 1997. TRAB-3D a New Code for Three-dimensional Reactor Dynamics, CD-ROM Proceedings of ICONE-5, 5th International Conference on Nuclear Engineering, Nice, France, paper ICONE5-2197.

[4] J. Devooght, E. Mund, *Numerical solution of neutronkinetics equations using A-stable algorithms,* Progr. Nucl. Energy, Vol 16, No 2, pp. 97-126, 1985.

[5] U. Grundmann, J. Hádek, *Calculation of neutron kinetic experiments with the code DYN3D/M1,* Kernenergie 34 1/1991: 12-20

[6] H. Finnemann, A. Galati *NEACRP 3-D LWR CORE TRANSIENT BENCHMARK final specifications,* OECD Nuclear Energy Agency, 1991

6.1 ACKNOWLEDGEMENTS

The project to implement the neutronics model of HEXTRAN into APROS is performed in concert with Fortum power and heat Ltd.