VVER REACTOR PHYSICS CODE APPLICATIONS

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ABSTRACT

The coupled steady-state reactor physics and thermohydraulic code system KARATE has been developed and applied for VVER-1000 and VVER-440 operational calculations. The 3D coupled kinetic code KIKO3D has been developed and validated for VVER-440 accident analysis applications. The coupled kinetic code SMATRA developed by VTT Helsinki has been applied for VVER-440 accident analysis. The paper gives a summary of the experience in code development and application.

Introduction

KFKI Atomic Energy Research Institute has a long tradition in developing, validating and applying reactor physics code systems. The present paper deals with recent developments and applications.

The KARATE code system [1] was developed for VVER-1000 and VVER-440 operational calculations. The code system has been validated against critical facility experimental results, mathematical benchmarks and operational data.

The KIKO3D code [2] has been developed for 3D kinetic calculations. The validation of the code is still going on.

The SMATRA code [3] developed by the Technical Research Centre of Finland, has been adopted for performing RIA and ATWS analysis in the framework of the safety re-assessment of the Paks NPP.

In each code mentioned above the reactor physical properties of VVER cores are modelled in a straightforward manner. The thermohydralic feedbacks are taken into account in a proper way. The results presented below illustrate the applicability of the codes.

KARATE

The KARATE code system [1] was developed originally for VVER-1000 applications and it has been adopted for VVER-440 calculations. The purpose of code development was the proper calculation of steady-state neutron physical and thermohydraulical processes in the reactor core. The code system is applicable for normal operational, slow transient and start-up conditions. The system consists of three calculational levels:

- multigroup calculation of fine-mesh diffusion-type constants
- fuel assembly homogenization in fine-mesh diffusion theory
- coarse mesh global calculation of the reactor.

Few-group fine-mesh diffusion-type constants for each type of lattice cells are generated by multigroup transport calculations. They are performed by using 35 epithermal and 35 thermal energy groups on the basis of the ENDF/B-IV nuclear data library. Effective multigroup cross-sections in the resonance region are calculated by using refined equivalence theory. The effective group constants are parametrized as a function of all relevant parameters, including a special spectral index introduced to account for the non-asymptotic character of the neutron spectrum.

Fuel assembly calculations are performed by solving the two-group hexagonal 2D fine mesh diffusion equation. The equations are solved for the actual assembly and the six one-thirds of the neighbouring assemblies. The irregularity between assemblies is accounted for. The boundary condition permits both for the traditional white-boundary-condition homogenization and also for heterogeneous-boundary-condition response matrix calculations. The fuel assembly calculations provides for the global calculations with homogenized constants or response matrices, but they can be also used for microdistribution calculations prescribing the actual boundary conditions resulting from global calculations.

Reflector regions and also the VVER-440 bulk absorbers are represented by albedo matrices in the global calculations. These albedo matrices are calculated by using fine mesh diffusion equation (reflectors) and transport theory (bulk absorbers).

The assembly group constants and reflector albedo matrices are parametrized for each assembly type (reflector segment type) according to the relevant parameters.

The global calculations are performed in two energy groups by means of a 3D nodal code in hexagonal geometry. The nodes are the assemblies subdivided into axial layers. The global calculations can be applied for the modeling of

- given reactor state
- fuel cycle
- xenon transient
- refuelling

and also for the determination of reactivity coefficients and kinetic parameters.

The thermohydraulical models of KARATE make it possible to calculate the thermal and hydraulical properties of the core on the subchannel basis, preserving the mass, energy and momentum. KARATE is supplied with two-phase flow option and the parametrization of reactor physical quantites also allows for such calculations.

From among the very many comparisons made for validation purposes three results are presented here. Table 1 shows some results for well-known critical facility benchmarks. Table 2 gives the relevant results for various 2D and 3D mathematical benchmarks set up for VVER reactor coarse mesh reactor calculations. Finally the comparison of the variation of boron concentration with time for the VVER-1000 type Kalinin NPP Unit 1, Cycle 1, is shown on Fig. 1 (both KARATE and the Russian BIPR-7 results are given).

Table 1. Calculation vs. Measurements for BAPL and TRX Lattices

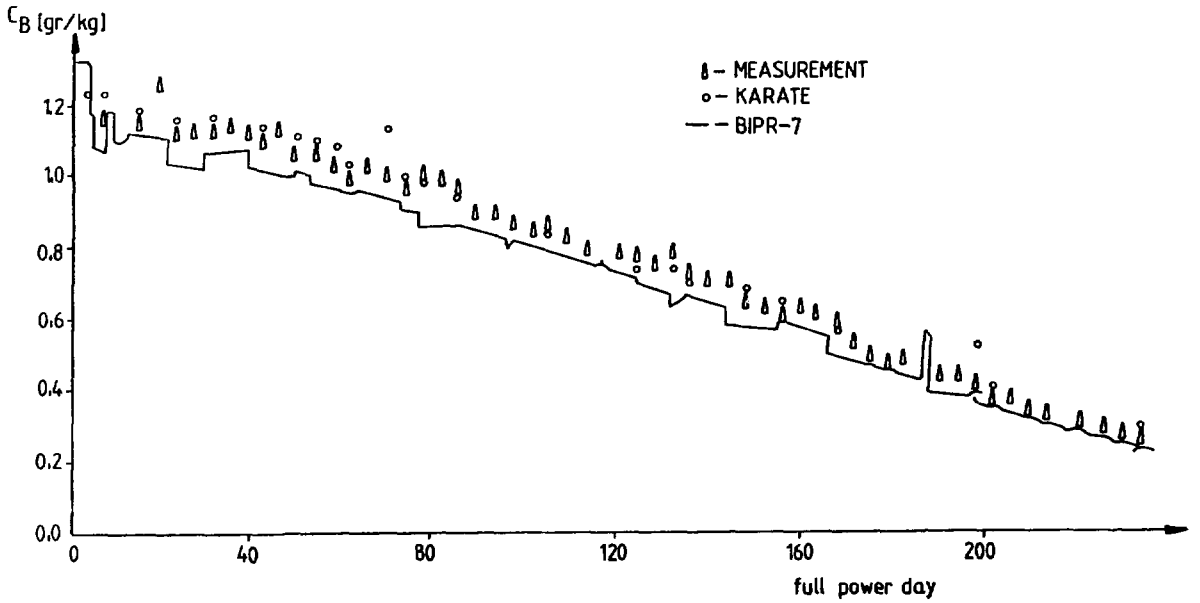
Parameter Lattice		δ^{25}	δ^{28}	ρ^{28}	k_{eff}
BAPL	measured	0.084 ±0.002	0.078 ±0.004	1.39 ±0.01	1.0
	KARATE	0.082	0.073	1.44	0.9986
TRX-1	measured	0.0987 ±0.0010	0.0946 ±0.0041	1.320 ±0.021	1.0
	KARATE	0.0990	0.0941	1.392	0.9949
TRX-2	measured	0.0614 ±0.0008	0.0693 ±0.0035	0.837 ±0.016	1.0
	KARATE	0.0606	0.0679	0.865	1.0019

Table 2. Results of VVER Mathematical Benchmarks

Case	k_{eff} reference	k_{eff} KARATE	ΔP(%) average	ΔP(%) maximum
VVER-440 2D	0.9776	0.9779	0.6	1.8
VVER-440 3D	1.0113	1.0119	1.0	5.0*
VVER-1000 2D	1.0116	1.0170	2.0	2.0

* Boundary problems exist in the reference solution

Figure 1. Comparison of boron concentration time dependence



KIKO3D

KIKO3D is a 3D kinetic code [2] developed in the KFKI Atomic Energy Research Institute under the auspices of the Hungarian State Commission for Technical Development. The code is capable for solving the kinetic problem in 3D hexagonal and 1D slab geometry. The time dependent nodal equations were derived under the following approximations:

- the unknowns are the scalar flux surface integrals on the boundaries of the nodes
- the flux inside the nodes is a linear combination of static solutions, asymptotic and transient plane-waves are perpendicular to the node boundary surfaces
- the flux distribution inside a node must be close to the solution of the time-dependent equation
- the flux and current integrals are continuous on the node boundaries.

The Improved Quasi Static factorization method is applied for solving the equations.

The code is supplemented with solvers for the thermohydraulic equations (only for VVER-440 studies, i.e. for 3D hexagonal geometry with closed fuel assemblies). Coupling the code with a primary circuit model has not been made yet.

Various benchamrk problems have been solved in validating the KIKO3D code. First the kinetic problem proposed by Langenbuch, Maurer and Werner [4] and a heavy water reactor kinetic problem [5] were solved with very good accuracy [6]. The results of these methodical benchmarks are illustrated on Figs. 2 and 3. Later a VVER kinetic benchmark problem set up in the framework of the AER cooperation [7] was solved also with very satisfactory results [8]. This rod ejection problem does not assume thermohydraulic feedback. Fig. 4 shows the results obtained by KIKO3D and the code DYN3D developed at the Research Centre Rossendorf, Germany [9].

Figure 2. Average Power in Benchmark Problem [4] as Calculated by KIKO3D

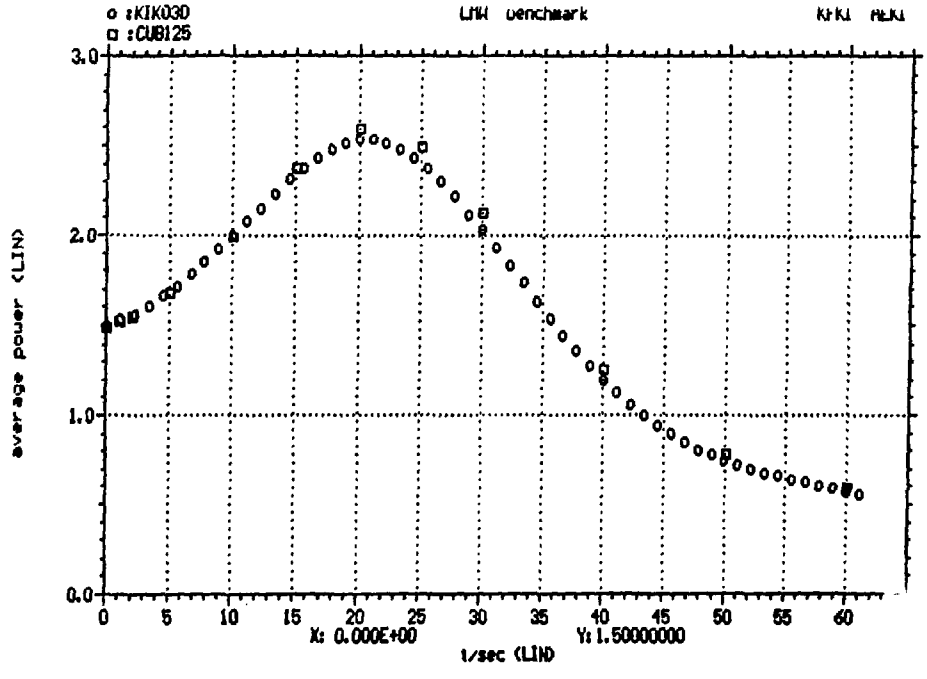


Figure 3. Average Power in Benchmark Problem [5] as Calculated by KIKO3D

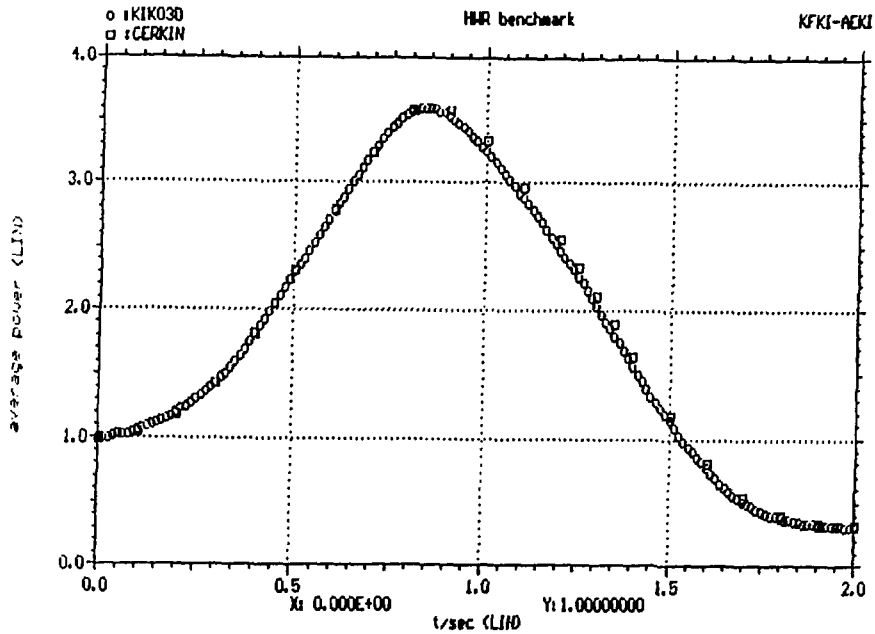
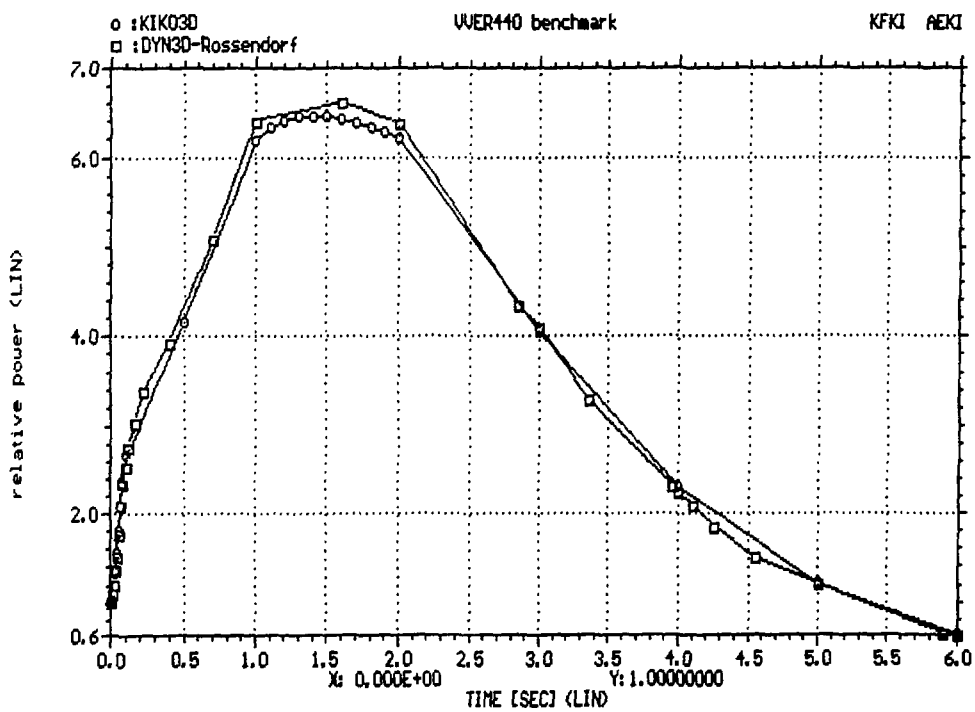


Figure 4. Relative Power in Benchmark Problem [7] as Calculated by KIKO3D and DYN3D



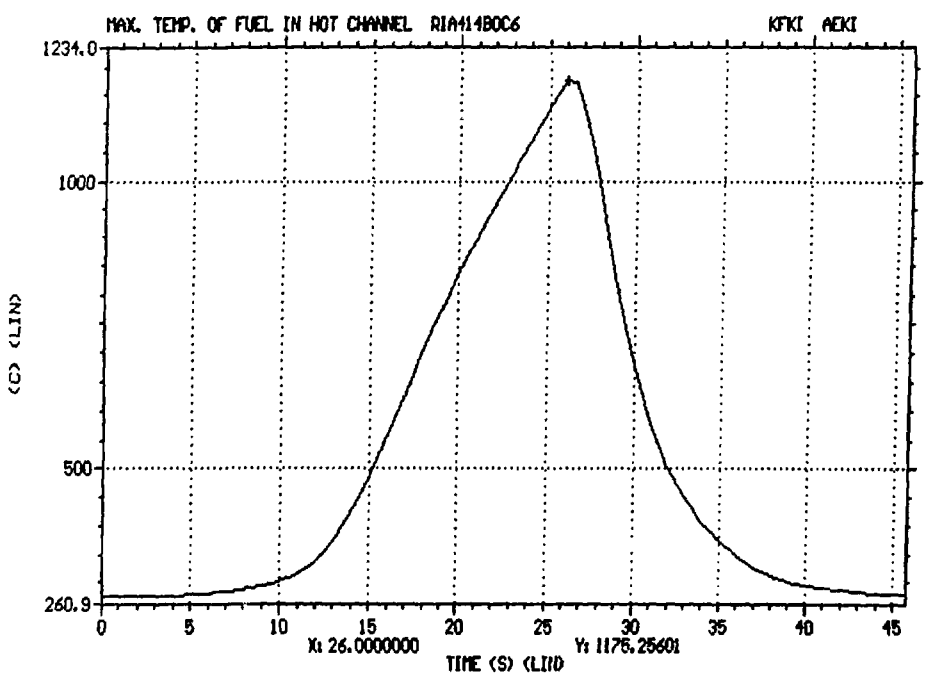
SMATRA applications

The SMATRA 1D kinetic code [3] was developed and validated at the Technical Research Centre of Finland. This code was chosen to perform RIA and ATWS analysis in the framework of the safety re-assessment of the Paks NPP (AGNES Project).

The SMATRA code was adapted in the KFKI Atomic Energy Research Institute and a full Paks NPP specific data base was provided by the AGNES project. The data base includes the modeling of the reactor protection system. The results of validation calculations demonstrate the applicability of the code for Paks NPP specific RIA analysis purposes.

RIA analysis is being performed in a close co-operation with IVO International, Finland. Initiating events analysed were selected in accordance with the onternational practice, taking into account the peculiarities of VVER-440 reactors. In certain RIA cases 3D analysis is required and either IVO's HEXTRAN code [10] or KIKO3D is applied. In other RIA cases the 1D calculations are preceeded by a careful selection of input data assuring the necessary conservatism of the calculation, while in ATWS analyses mainly best estimate data are used. The results of analyses are compared with adequate acceptance criteria in order to show whether the consequences of the anticipated operational occurrences and postulated accidents remain within the limits permitted for these categories of events.

Figure 5. Maximum Fuel Temperature in a VVER-440 RIA - SMATRA Calculation



Calculational results can be illustrated with an inadvertent rod withdrawal case. The rod withdrawal occurs at 2% power and it is assumed that the setpoints correspond to the 100% case. Maximum fuel temperature is shown on Fig. 5. It is worth to mention that the actual temperature dependence does not lead to cladding failure.

References

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