

**NUMERICAL MODELING OF REACTIVITY  
EXCURSION ACCIDENTS IN SMALL  
LIGHT WATER REACTORS**

**Ph.D. Dissertation**

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*“Everything should be made as simple as possible, but not simpler.”*

Albert Einstein

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## ***Introduction***

For safety reasons, nuclear power plants need fast and accurate plant calculation models. Plant calculation models mainly consist of two different modules which account for the basic physical phenomena taking place in the plant: a neutronic module which simulates the neutron balance in the reactor core, and a thermal-hydraulic module which simulates the heat transfer from the fuel to the water used as a coolant, and the various evaporation and condensation processes which take place in the reactor core and in the condenser systems. These kinds of nuclear plant calculation methods are too complicated for being applied for research reactors, consequently, the later require their own models.

One of the accidents to be analyzed using such calculation models is the reactivity accident or power excursion. The subject of the present thesis is to describe a model developed for this purpose for the Training and Research Reactor, (BME-Reactor) of Budapest University of Technology and Economics. The BME-Reactor became critical in 1971. It is a tank type reactor, which is located on the site of the University. It is designed to be compact and safe and it is used mainly for neutron activation analysis, production of short-lived radioisotopes and for education and training. Its maximal thermal power is 100 kW

The training reactor, being located in the downtown of Budapest, has to satisfy special safety requirements, namely, it has to be inherently safe. It is required by the safety authority that the safety report of the reactor proved that such an accident could not lead to core damage and release of radioactive material. The former safety report written in 1996 used a combination of point reactor kinetic code REMEG and module HEATING of the SCALE 4.1 program system. This study was accepted at that time but the excess reactivity may not be increased until the study is repeated by a more detailed and mathematically better founded computer model.

The actual excess reactivity of the reactor is 0.82 \$ which means that no power excursion is possible in the strict sense of the word. In 1996, this value was estimated to be between 1.04 \$ and 1.10 \$. Such values would not allow serious reactivity accidents either. At the same time, certain irradiation experiments regularly performed at a power of 100 kW during 8 hours use up practically all this excess reactivity: 0,6 to 0,7 \$ for the xenon and the temperature effects. Therefore, the control of the reactor becomes rather difficult during the last hours of the irradiations. Consequently, the excess reactivity needs to be increased to a value what the reactor had in 1980, i.e. near 1.2 \$ in order to restore the original flexibility of the reactor operation.

The reactor can get permission for achieving this only if we are able to prove that even a prompt supercritical state of 1.2 \$ can not result in a reactivity accident leading to core damage. It is trivial that it is not possible to check this statement experimentally at this reactor. Thus, we need a computer model simulating this accident. The first question that naturally arises is whether such a model is available in international computer libraries. The answer is simply no because there are very few reactors operating with EK-10 type fuel elements (either research or training reactors). Such reactors were built in the former Soviet Union and socialist countries but most of them have been upgraded to higher powers by replacing the fuel rods with other types. For

example, the 2 MW Budapest Research Reactor was upgraded to 5 MW in 1965 when the EK-10 fuel elements were replaced by tubular fuel clusters containing three coaxial tubes. This explains why no computer model is actually available for this particular fuel type, consequently, we had to develop our own model. This explains why this study has been done.

The second question is what kind of a computer model we need. A power excursion is stopped only by the feedback through the temperature coefficients (both Doppler and moderator). It follows from this that only a time dependent neutronic code with thermal-hydraulic feedback could be acceptable. We need a 3D space-time dependent model with thermal-hydraulic feedback. Similar codes at other reactors are based on the diffusion equation. This approximation should be sufficient for our case, too. There are codes which use finite elements while others use finite differences. The latter seemed to be simpler and more accurate because we could afford to put a mesh point in every elementary cell. We write the neutronic equation to be solved in the following form.

Many models for coupled kinetics and thermal-hydraulics consider a point reactor. REMEG Code is one of these models. It uses the point kinetic equation for the neutron balance, fuel temperature sub-model to calculate the heat transfer from the fuel through the cladding into the moderator, and simplified coolant temperature sub-model. Its results for the power excursions are compared with the current work. Also, PARET and Lumped-Parameter Model are other examples of these models.

TWIGL model is one of such models that use direct solutions techniques, which solve the equations by finite differencing them in space and time. In the core dynamics of TWIGL code the fluid dynamics is modeled for only single-phase liquid water. It is used to do conservation safety calculations for those accidents like the fuel element ejection accident.

There is a variety of models that using the nodal methods in treating the neutron dynamics such NLSANMT/COBRA-IV, KIKO3D/ ATHLET, and TRAC-PF1/NEM. They are all developed for power plants.

Most of the models mentioned are strongly dependent on the special features of the reactors the behavior of which they simulate. Therefore, I have used only methodic ideas applied in them for developing our own model for the BME-Reactor.

### ***Structure of the Dissertation***

Chapter 1 describes the problem to be solved and why it is important from reactor safety point of view. Chapter 2 gives a simple survey about the available reactivity excursion models. Chapter 3 deals with the iteration methods for solving the space-time dependent few group neutron diffusion equations. Chapter 4 contains the thermal-hydraulics feedback calculations, cross-section generation and the coupling of the 3D-neutronic and thermal-hydraulic parts. Chapter 5 discusses the nuclear safety analysis of the BME-Reactor during transient conditions, final results and conclusions.

### *Methods Used in the thesis*

Numerical solutions of the coupled time-dependent transport and precursor equations for reactor kinetics problems of practical interest are prohibitively difficult, so approximate methods are employed. This thesis is concerned with the most common approximation to the time-dependent transport equation i.e., the time dependent few group diffusion equation that have been developed for the purpose of analyzing nuclear reactor transients that occur on time scales of milliseconds to ten's of seconds.

There are various methods for calculating the spatial neutron flux distribution such finite difference, nodal, finite element, and synthesis ...etc, direct finite difference methods are the most straightforward approach to the accurate solution of the space time kinetics problem. It represents the basic tool of this work.

The BME-Reactor to be described consists of several homogeneous regions: core, graphite reflector, water reflector. For the description of this reactor, the x-y-z geometry is the most adequate. In the finite difference method, this system is covered by a 3D mesh so that every region boundary should be a mesh plane.

The standard methods of the numerical solution of the multigroup neutron diffusion equation are based on the outer-inner iteration strategy. In the current work a 3D space-time-dependent neutron diffusion POWEX-K code has been developed. It solves the few group neutron diffusion equations by using the backward finite difference approximation for the time derivatives. The spatial derivatives are approximated by usual finite difference scheme XYZ-geometry. We have developed a new iterative scheme which simplify the treatment of the delayed neutron precursors concentrations and omits the usual source iteration. It is designed as a part of integrated neutronic/thermal-hydraulic code for simulating power excursion accidents for the BME-Reactor. WIMS-D4 code and GRACE and THERMOS codes are used to calculate the group constants and delayed neutron parameters, respectively.

The method has been found unconditionally stable. Due to the backward scheme, the time step  $\Delta t$  is limited only by the speed of change of the neutron fluxes. Thus, relatively large time steps may be used for slow processes. However, they should be of the order of milliseconds for power excursion accidents.

Fuel temperature, moderator density, fission product poisons, and fuel burn-up are examples of feedback mechanisms that influence reactor dynamics. Feedback mechanisms can be distinguished by the response time to the changes occurring in the system. Fuel temperature and moderator density have time constants of order of seconds. They are hours for fission product poisons and weeks or months for fuel burn-up. Since the power excursion transient is very fast, we may assume that only the fuel and moderator temperatures (consequently the moderator density as well) are the significant feedback mechanisms. These feedbacks are taken into account by recalculating the few-group constants at each time step for each node.

Thermal-hydraulic code MI based on the simple two phase momentum integral model (originally developed for boiling water reactors) has been developed and coupled with the neutronic code POWEX-K to form the integrated neutronic/thermalhydraulic code (POWEX-K/MI). The heat transfer model in MI inside the fuel region is based upon one-dimensional radial heat conduction. The conservation equations are written for one-dimensional axial homogeneous upward flow through the channel. On the other hand, constitutive equations as heat transfer coefficient, friction factor are used taking into account the geometry as well as the convection regime (forced and natural).

In MI code module I solve the one-dimensional conductive heat transfer from fuel to the clad and the convective heat transfer from the clad to the coolant for nuclear fuel rods to compute the rod internal temperature distribution and the rod surface heat fluxes, in which the effects of radial conduction and temperature dependence of thermal conductivity are considered. The fully implicit finite difference method is used again for treating the spatial, time variables of the hydrodynamic equations and the radial, time variables for the heat transfer process.

When, for a given power distribution inside the core, the thermal-hydraulic module computes the average fuel, clad and coolant temperatures for every mesh point of the finite difference scheme, we are able to calculate the few-group constants of the 3D diffusion equation. (If necessary, we can recalculate the power distribution leading the new temperatures for which the few-group constants can be recalculated and so on until convergence is reached. But, this iteration has been found superfluous in practice). These few-group constants can be obtained by an asymptotic slowing-down and thermalization code (WIMS-D4 -4D in our case). The thermal-hydraulic feedback is taken into account via the recalculation of the few-group constants.

It is needless to say that the few-group constants are calculated once for ever for some selected values of the fuel, clad and coolant temperature. We thus obtain tables of the few-group constants with three entries. These tables allow interpolating the few-group constants for every particular set of these three temperatures. We reach now an important point. The range and the selected values of the temperatures should enclose those temperatures which will occur during the studied power excursions. Preliminary calculations have shown that the following temperature ranges would be sufficient:

- From room temperature to 100 °C for the coolant temperature;
- From room temperature to 1600 °C for the fuel temperature;
- From room temperature to 500 °C for the clad temperature.

If, in the actual calculations, anyone of these temperatures gets out of these ranges, we have to enlarge the range of the pre-calculated few-constants. Especially critical is the range of the coolant temperature.

## Results and Discussion

The code POWEX-K was tested by a start-up experiment during which the reactor power increases exponentially but slowly. Since such experiments are performed at very low power, no thermal-hydraulic feedbacks need be taken into account. It follows from this that the thermal-hydraulic module of the code was not used in these test calculations i.e., POWEX-K only is used. The group constants were obtained by WIMS-D4 assuming the same temperature for fuel, clad, and moderator, Figure 1.

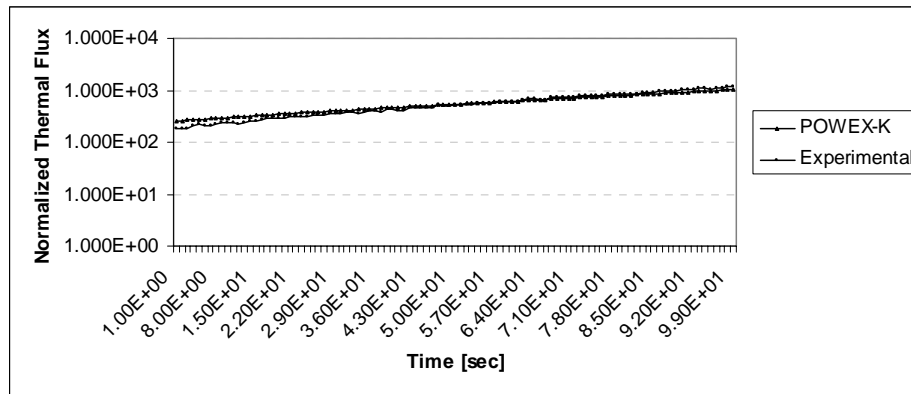


Figure 1. Time dependence of the measured count rate and the thermal flux in the BME-Reactor at 14 cents reactivity

Figure 1 presents the comparison between the calculations using the POWEX-K/MI code and the experimental data. From this plot, one notices the exponential trend of the thermal flux at the measuring channel of BME-Reactor and that the results of simulations performed by POWEX-K/MI are in good agreement with the corresponding experimental data. The small discrepancy at the beginning can be explained by the fact that the code did not simulate the presence of the neutron source which was in the reactor in the initial phase of the experiment.

The desirable excess reactivity of BME-Reactor is 1.2 \$, and the actual excess reactivity is 0.82, thus, the power excursion is analyzed for these ramp reactivity's.

Figure 2 shows the clad and coolant temperatures at a fuel rod located near the core center and in the axial position located at half length of the fuel pin. The clad temperature increased to 46 °C according to POWEX-K/MI while only to 28 °C according to REMEG in 3.5 sec following the ramp reactivity insertion of 1.2 \$. This figure shows that the clad temperature is much lower than the melting point of aluminum (which is 660.2 °C). The moderator temperature for both codes is still less than 21 °C. The calculations were done up to 3.5 seconds until all the heat generated due to the transient went into the moderator and fuel and clad temperatures are stabilized.



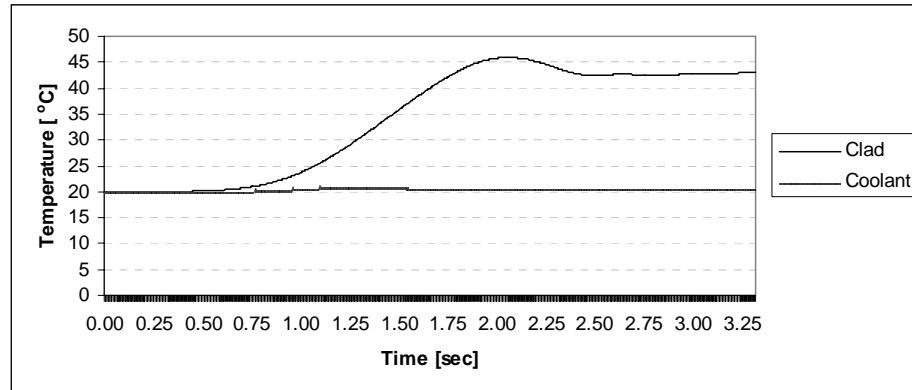


Figure 2. Time behavior of clad and coolant temperatures for ramp reactivity 1.2 \$

### ***Conclusion***

From the results presented in the previous section, we can draw the following conclusions:

- The comparison of the POWEX-K/MI model predictions with the data measured during the start up experiment at about 14 cents reactivity shows a very good agreement. This is a strong argument in favor of the neutronic part of the model.
- From the comparisons with the results of the point kinetic REMEG code, we may conclude that the 3D model predicts the reactivity accident more severe both as far as peak powers and temperatures are concerned. This is due to a better thermal-hydraulic description by POWEX-K/MI.
- Even for 1.2 \$ reactivity insertion, we need not reckon with the melting of the clad and boiling of the moderator.

## ***New Scientific Results***

### ***Thesis One***

I have written a time dependent 3D-neutron diffusion code adapted for the Training and Research Reactor of Budapest University of Technology and Economics. I have developed a new iteration scheme for solving the discretized neutron group diffusion equations which simplifies the treatment of the delayed neutron precursors and omits the source iteration. In addition, it accelerates the iteration. The method requires a relatively great number of iterations only for the first time steps but, later on, only very few iterations are needed for satisfying the convergent criteria.

### ***Thesis Two***

The new iterative scheme deviates from the usual source iteration schemes. Hence, a study of the stability and convergent was needed. Thus, the numerical stability of the new scheme and the convergence of the associated iterative process have been proved. I have proved that the iteration is always convergent for subcritical reactor states while, for supercritical states, it is convergent if the time step  $\Delta t$  sufficiently small:

$$\Delta t < \Delta t_{\max} = \frac{1}{\omega_0}$$

where  $\omega_0$  is the asymptotic time constant.

### ***Thesis Three***

I have written a code for radial heat conduction in the fuel and the cladding, furthermore a code of thermal-hydraulics of the moderator realizing the so called momentum integral model. I have coupled them to the neutronic part of the POWEX-K/MI code.

### ***Thesis four***

By the aid of WIMS-D4 program, for one hand, and GRACE and THERMOS programs, for the other hand, I have calculated the 4-group constants as functions of the fuel, clad, and moderator temperatures for describing the feedbacks necessary in power excursion accidents. For any set of values of this temperature, the actual 4-group constants can be calculated by an interpolation procedure (also included in the POWEX-K/MI code).

### ***Thesis five***

I have checked the neutronic part of the program by comparing the calculated results with experimental values measured during a start up experiment for about 14 cents of super-critical reactivity. The agreement was excellent. I have checked the thermal-hydraulic model by comparing the results with values taken from literature.

### ***Thesis six***

I have simulated the power excursion accidents for positive reactivity insertions 1.2 \$ and 0.82 \$ with reactivity feedback calculations. I could conclude that no fuel damage is to be expected for these reactivity insertions. The results were compared with those obtained from the point kinetic program REMEG.

### ***Future Work***

At this point of my research, it becomes visible to me that in this framework, two research programs would be interesting:

- Including the control rods effects inside the neutronic module.
- Changing the core geometry, coolant properties to be applied for power plants which have good benchmark problems.

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**AUTHOR LIST OF PUBLICATIONS RELATED TO THE THESIS**

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1. **S. M. Khaled**, Z. Szatmáry, “Numerical Method for Solving the Three Dimensional Time Dependent Neutron Diffusion Equation”, The International Journal of Nuclear Energy Science and Technology, Vol. 1, n° 4, 297-311, 2005. (Published)
2. **S. M. Khaled**, Z. Szatmáry, “Two-dimensional Time Dependent Calculations for the Training Reactor of Budapest University of Technology and Economics”, Romanian Reports in Physics Journal, Romania, pp.53-60, Volume 57, Number 1, 2005. (Published)
3. **S. M. Khaled**, Z. Szatmáry, “Time Dependent Calculations for the Training Reactor of Budapest University of Technology and Economics”, 4th International conference of PHD students, University of Miskolc, Hungary, pp. 69-74, 11-17 August 2003. (Published)
4. **S. M. Khaled**, “Analysis of A Coupled Neutron Kinetics and Thermal-hydraulics model for Small Light Water Reactors”, International Conference on Nonlinear Operators, Differential Equations and Applications, August 24-27, 2004, Cluj-Napoca, Romania. (short talk)
5. **S. M. Khaled**, “Stability and Convergent Analysis of 3D-Kinetic and Thermal-hydraulic Model for Training Reactor”, International Conference on Nonlinear Operators, Differential Equations and Applications, August 24-27, 2004, Cluj-Napoca, Romania. (Poster)
6. **S. M. Khaled**, “Major Advances in Neutronic-thermalhydraulics Transient Analysis”, A Century of Great Discoveries in Physics, organized by: Egyptian Nuclear Physics Association (ENPA), 19-20 Feb. 2005, Cairo, Egypt. (short talk)
7. **S. M. Khaled**, “Power Excursion of the Training and Research Reactor of Budapest University”, The International Journal of Nuclear Energy Science and Technology, (Submitted)

## CURRICULUM VITA

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- **Khaled Sayed Mahmoud** is Egyptian, Married. He was born in Helwan, Egypt 1968, December, 13. He was graduated at Helwan University in the Applied Mathematics Sciences with grade “Excellent with Honour grade“. He was granted the degree of M.Sc. thesis in 1997 entitled ““High Energy Elastic Scattering of Hadrons by Nuclei”. From September, 1998 to Dec.1999, He got his Pre-Ph.D. from Helwan University in the Quantum Chromodynamics Field. Since Dec. 2000, He is a regular Ph. D. student in the Dept. of Nuclear Techniques, Institute of Nuclear Techniques, Budapest University of Technology under supervision of Prof. Dr. Szátmary Zoltan and Ass. Prof. Dr. Aszódi Attila. He participated in 3 International Conferences and delivered 3 lectures of it in addition to the local conferences in Egypt. His interests of research are: Related reactor physics and thermal-hydraulics models, Computational fluid dynamics.

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- Computational fluid dynamics