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The studies of RBMK-1500 reactor core behavior during abnormal operation transients

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This paper describes series of RBMK-1500 reactor transient investigations, performed with CORETRAN code. Aspects of the reactor core neutronic and thermal hydraulic behavior during postulated core transients were analyzed. In particular, RBMK-1500 transients leading to changes in reactor power and core reactivity were considered. Three reactivity-initiated accident cases were addressed: a) spontaneous control rod bank withdrawal in the central part of the core; b) spontaneous control rod bank withdrawal in core periphery and c) release of one Shortened Absorber Rod from the reactor core. Reactor nominal power operation was used as the reference core state. Analysis was performed using data, obtained from the actual plant database recorded for Ignalina Nuclear Power Plant Unit 2 on 27th January 2001. The CORETRAN calculations were benchmarked against STEPAN code results.

KEYWORDS: RBMK-1500, reactor core analysis, abnormal operational transients, control rods, reactivity-initiated accidents.

1. Introduction

The two largest RBMK-type reactors in the world (RBMK-1500) operate at Ignalina NPP in Lithuania since 1983 (1987) [1]. The reactor design thermal power is 4800 MW_{th} (which corresponds to 1500 MW_{el}). However, due to the safety recommendations the nominal reactor thermal power was reduced to 4200 MW_{th}.

RBMK is a channel-type graphite-moderated reactor. The core heat transfer and neutron dynamics processes in the RBMK reactor are different from the conventional Western design Light Water Reactors (LWR): in particular, the RBMK has a positive void reactivity coefficient at operational power levels.

Originally, the RBMK-type reactor core analysis codes were developed and used in the former Soviet Union by the Russian reactor designers and code developers. Two codes were developed for the purpose: STEPAN at the Russian Research Center Kurchatov Institute and the SADCO at Research and Development Institute of Power Engineering.

Recently, several computer codes, developed for the Western-type LWR reactor analysis are also being applied for the RBMK core studies and plant safety evaluation. The validation and verification of these codes for the RBMK-type reactor applications is still being performed. Therefore, the modeling of complex neutronic and heat transfer processes in the RBMK using advanced Western computer codes and benchmarking the results against traditional codes (in this case – STEPAN) is an interesting and important scientific issue.

Soon after the Chernobyl accident in 1986 the scientific community in Western countries felt necessity for the independent investigations of RBMK reactors, however, they had no ac-

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cess to the RBMK data. Now the documentation and reactor operation data became available to the Western scientific community. The possibilities for the Western partners to participate in the scientific investigations and safety monitoring of RBMKs were opened and series of studies were performed during 1991-2003. The intensive scientific work since 1991 has resulted in adoption of several codes for the RBMK reactor thermal hydraulic and neutronic calculations.

One of the codes, currently used for the RBMK-1500 core analysis is CORETRAN, initially developed at the Electric Power Research Institute (EPRI) for the LWR analysis. Several modifications were introduced into the CORETRAN code ([2], [3], [4], [5]) in order to facilitate the RBMK type reactor calculations. Since 1998 the code is used for the RBMK-1500 core studies at the Nuclear Power Safety Division of Royal Institute of Technology (KTH) (Stockholm, Sweden). A large number of various RBMK-1500 core transients were investigated during the last 5 years. The specialists from KTH, together with there colleagues from Lithuania participated in several Ignalina NPP (INPP) RBMK-1500 reactor safety evaluation projects. This paper presents a part of this work.

Previous studies on reactivity-initiated accidents (RIA) for RBMK-1500, performed with CORETRAN included, among the other, analysis of the following transients:

- Erroneous reloading of a fuel assembly [6];
- Withdrawal of single control rod from RBMK-1500 core central part and core periphery [8,9];
- Drop of control rod into the core [8,9];
- Control and Protection System Loss of Coolant Accident (CPS LOCA) [8,9].

Recently, due to the international cooperation, Lithuanian specialists from Kaunas University of Technology (KTU) were also involved in the RBMK calculations using the CORETRAN code. This cooperation was elaborated during the review of the Safety Analysis Report (SAR-2) of Ignalina NPP Unit 2, where CORETRAN code was used to perform independent audit calculations on chosen RIA. The results of the calculations are the subject of this paper.

2. RBMK-1500 reactivity-initiated accidents modeling

2.1 Initial data for core and related systems models

For the study presented in this paper the actual Ignalina NPP core state, recorded by the reactor data acquisition system was used for the definition of initial core conditions. Reactor status data from the plant database for January 27, 2001 (INPP unit 2) were used as the initial data for the CORETRAN model. The database was produced by the INPP main computer information system “TITAN”. The database contained maps with core loading pattern, evaluated and measured power distribution, axial fuel burnup profiles, fuel channel coolant flow rates, insertion depths of the control rods, etc. This data were used to define the initial core parameters for the model.

The global core parameters at the reference state: reactor thermal power – 4164.1 MW_{th}; coolant core inlet temperature – 265 ^oC; total coolant inlet flow rate - 42776 m^3/h; pressure in main circulation circuit - 7 MPa.

RBMK Control and Protection System (CPS) [1] is designed to provide a compensation for possible reactivity deviations in the reactor core. The entire core, schematics of which are presented on Figure 1, is divided into 12 Local Automatic Control (LAC) zones and 12 Local Emergency Protection (LEP) zones. Each LAC zone contains 1 LAC rod, and each LEP zone contains 2 LEP rods. Each zone contains radial and axial neutron flux detectors (divided into several independent groups), which operate the relevant control rods, i.e. the rods in a certain local sub-region are moved automatically in response to signals due to disbalances between core operational set points and the local power level (neutron flux).

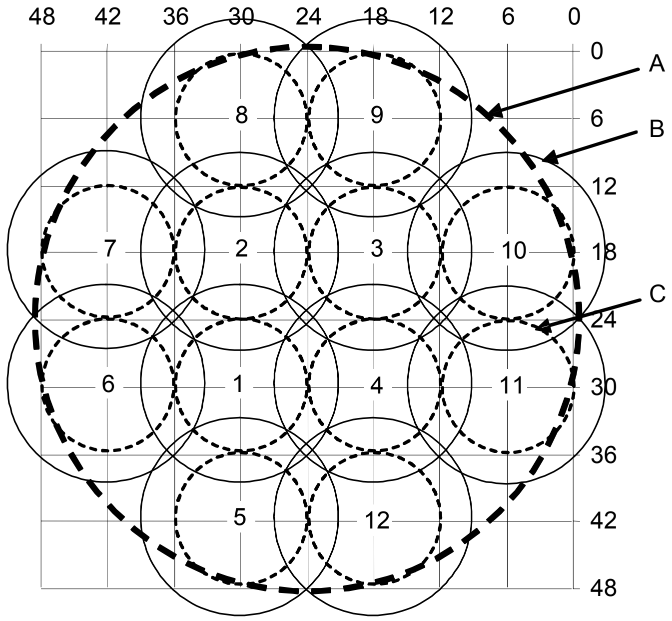

Fig.1 RBMK-1500 core and CPS zones in INPP coordinates:
A – core boundary; B – LEP zone boundary; C - LAC zone boundary

-In addition to the rods, which are operated automatically by the in-core sensors (LAC and LEP), RBMK CPS possesses additional means, which provide reactor operator with a possibility to smooth the radial power distortions manually. These are Manual Control Rods (MCR) (which are inserted from the top of the core) and shortened absorber rods (which are inserted from the bottom of the reactor). The both types of rods are distributed evenly within the core.

The servo-drive device of shortened absorber control rod is situated above the core, similar to the MCR design in RBMK. Shortened control rods are suspended on steel tapes via dampers, which absorb their movement (lifting and passing down) shocks and protect from twisting.

The CPS in-core monitoring sub-system consists of axial and radial detectors distributed evenly in the reactor core. The detectors in each local zone (LAC and LEP) are divided into separate groups and each group has an independent alternating current supply. This ensures continuity and credibility of monitoring. The signals from detectors are processed by the CPS: they are appropriately summarized and discrepancies against the current operational set points are estimated. In order to compensate the discrepancies, the appropriate LAC and LEP rod motions could be initiated to bring the monitored system to the balance.

According to the signals, generated by the in-core sensors, the following CPS modes could be actuated during a reactivity-initiated transient course:

- Compensation of inserted reactivity with LAC rods;
- Compensation with LEP rods;
- Emergency power reduction to 50 % of nominal power (i. e., to 2400 MW_{th});
- Emergency shutdown of the reactor.

Main set points and CPS signals are presented below:

- Power deviation of 0,5 % in the central (No.1-4) LAC zones and 1% in the peripheral (No. 5-12) LAC zones triggers the signal for LAC rod to compensate the power distortion in the certain LAC zone (core local power automatic regulation signal).

- Power deviation of 2 % in certain LEP zone triggers the signal for one of the LEP rods to be actuated for the compensation of reactivity in join with LAC rod;
- Power deviation of 10% triggers the signal for reactor power decrease to 50 % of nominal thermal power;
- If emergency power reduction signal generated in more then 2 LEP zones, the reactor scram signal is generated.

The representation accuracy of complex reactivity-initiated transient during the modeling depends on the chosen code design principles. The problem is, that only little part of computational results could be compared with the data from actual measurements. In a frequent case of absence of experimental data the only way to ensure the reliability of the analysis is to compare the relevant results obtained by the same case calculation with several independent codes. Benchmark study of CORETRAN results against STEPAN was performed. All STEPAN code calculations were carried out in KTH.

2.2. CORETRAN calculation model for reactivity-initiated accidents simulation

The CORETRAN model of RBMK-1500 incorporates the entire core (with both radial and axial neutron reflectors). The core is divided into 58x58x16 nodes (dimensions of each node are 25x25x50 cm in the X-Y-Z directions respectively). Therefore, each node corresponds to a separate graphite block in the reference reactor.

CORETRAN model for the reactivity-initiated events investigated includes accurate simulation of all above-mentioned CPS modes. Three different cases are available to be represented by the model: normal operation case, single failure constraint and Anticipated Transient Without Scram (ATWS). The code also simulates all CPS rod motions initiated by relevant in-core detector signals.

The calculation results are the basis for further analysis of reactor behavior. The methodology of the case analysis follows the conventional IAEA methodology for RBMK accidents analysis [10,11].

Scientific evaluation of RBMK behavior and safety during RIA accidents is based on acceptance criteria. All the accident cases for RBMK were classified qualitatively. According the present IAEA categorization both continuous withdrawal of MCR bank and shortened absorber rod dropout from the core, assuming the single failure condition, are classified as Design Basis Accidents (DBA). This means that DBA acceptance criteria should be taken into account.

The main criteria for reactivity initiated DBA consequences evaluation will be RBMK-1500 fuel element integrity criteria:

- Maximum fuel enthalpy is less then 712 kJ/kg;
- Maximum fuel temperature is less than UO_2 melting temperature (i.e. 2600^{o}C);
- Maximum fuel cladding temperature is less then 700°C.

Shortened absorber rod dropout was considered as ATWS case as well. According the classification of the IAEA [10,11] ATWS are beyond design basis accidents. RBMK-1500 beyond design basis accidents acceptance criteria:

- Maximum fuel enthalpy is less then 1000 kJ/kg;
- Maximum fuel temperature is less then UO_2 melting temperature (i.e., 2600 °C);
- Maximum fuel element cladding temperature is less then 1200 °C.

The safe reactivity limit must be ensured when core behavior during the course of any abnormal operation is evaluated. Hence, the reactivity monitoring is necessary to ensure, that the reactivity at a certain moment of time has a margin to the safety limit (1 β_{eff}).

2.3. Scenarios description

The electromechanical clutch limits the duration of each MCR withdrawal to 8 seconds (a timer is set to determine the time of withdrawal). After 8 seconds alternating current supply to the rod servo drive is blocked and the rod stops.

The continuous withdrawal of a manual control rod (during more than 8 seconds) would be possible only as a result of multiple failures of independent CPS subsystems. The continuous withdrawal of control rod bank could be initiated by reactor operator and must be must be a result of multiple failures of independent CPS subsystems.

The operator has a possibility to form a group of MCR rods (up to 4 rods per group) to perform some simultaneous operations.

During control rods bank withdrawal calculations it was postulated that all the rods of the manually formed group are being fully withdrawn from the core due to the multiple failures in the CPS sub-systems. The same scenario was valid both for rod bank withdrawal in the core center and rod bank withdrawal in the core periphery.

Even though operator can form the group of up to 4 rods, during the calculations of control rod bank withdrawal in core center the group of only 3 rods was chosen; this should impose later actuation of the relevant CPS subsystems (the fuel channels in the core center are of higher power and, respectively, the control rods are of higher reactivity worth). The case of the control rod bank withdrawal in core periphery was considered to compare the consequences. Peripheral rods are of lower reactivity worth, therefore, in this case the group was formed of the maximum possible number of rods.

The withdrawal of a control rod would induce local and global power fluctuations since two opposite processes would compete in the core: the reactivity insertion (by withdrawal of control rod) and the reactivity reduction due to CPS initiated rod movement. The single failure constraint is imposed on the CPS system: the first signal from the CPS in-core detectors to initiate reactor emergency power reduction or shutdown will be neglected and the whole group of detectors, which initiated the signal, will be excluded from the further consideration during the calculation.

Such important global parameters as total power, reactivity, maximum local temperatures should be precisely monitored. Since changes in enthalpy characterize the effects of fuel fragmentation during prompt processes (with duration of several milliseconds), this criterion was not significant in the context of the present study.

The initiating event of the third analyzed case is the rupture of the suspension tape of shortened absorber rod. According to the scenario the rod would drop out of the core, influenced by the gravitation force. The dropout of shortened absorber rod inevitably initiates reactivity insertion into the core: this would cause the disturbances in local power, monitored by the CPS detectors. Disturbances would possibly inflict the violation of the CPS automatic control and emergency protection sub-systems prescribed set-points. If automatic control subsystems will not be able to restore the balance and bring the reactor to a new safe steady state, CPS will actuate emergency protection modes in accordance with the prescribed algorithm.

The event was investigated i) by imposing single failure approach and ii) under ATWS conditions. In the first case single failure was treated similar as in the control rod bank withdrawal cases. The first emergency protection signal was not credited and the signal initiator was excluded from the further consideration in the case simulation. In ATWS approach the complete failure of reactor shutdown system was postulated (whereas all other CPS sub-systems remained available).

The preliminary study of the shortened control rod dropout with the single failure assumption demonstrated fast and reliable CPS systems operation. Thus, the authors decided to consider the same case as ATWS. This is in agreement with international practice and provides more complete details of the event.

Similar to the control rod bank withdrawal cases, the total power, reactivity, and maximal local parameters were monitored.

3. Calculation results

The results on three investigated reactivity-initiated cases: control rod bank withdrawal in the center and periphery of the core and dropout of shortened absorber rod are presented. Some benchmark calculations were carried out using STEPAN code.

At the beginning reference steady state was calculated by both – CORETRAN and STEPAN codes. The steady state axial power distribution is presented at Figure 2 (curves 1,2).

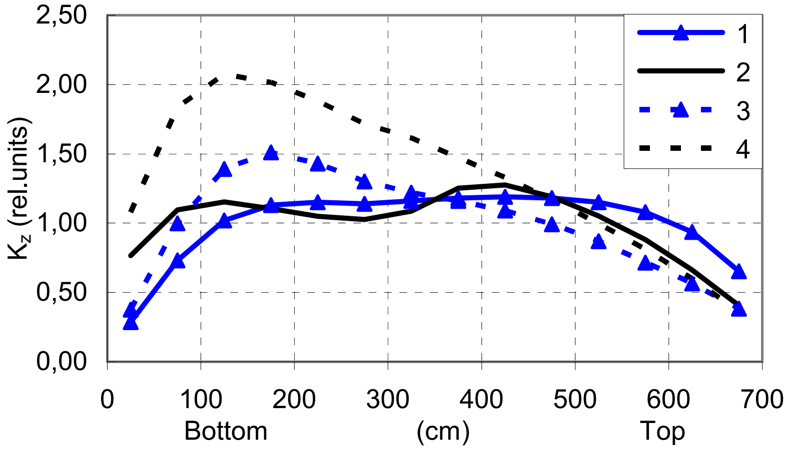

Fig.2 Axial distribution of local power at maximal power channel: 1,3 – STEPAN, 2,4 – CORETRAN; 1,2 – steady state, 3,4 – status after 2 s of transient.

There are significant discrepancies between CORETRAN and STEPAN steady state axial power profiles (Figure 2, curves 1 and 2). This can be explained by differences in approaches used in CORETRAN and STEPAN for the core initial state evaluation. Recovering procedure is being used by the STEPAN and some initial state data modifications (adjustment of the thermal group neutron cross section data for certain fuel assemblies) are introduced in order to reproduce the radial power distribution, similar to the distribution recorded in the INPP database. Modifications of initial state data are not performed with the CORETRAN code. Therefore, initial-state neutron field, evaluated with STEPAN, is “smoother” than the one, evaluated with CORETRAN and radial peaking factor is lower in STEPAN case.

An average axial power distribution simulated by CORETRAN is quite realistic and close to measured one [8].

All simulated cases were initiated from the steady state according described scenarios.

The behavior of the reactor total power in the case of MCR bank withdrawal out of the core is illustrated on Figure 3. The withdrawal of chosen MCR rods leads the consequent increase of the reactor power. The local power variations produce the appropriate response of the CPS. First of all, LAC rods are inserted into the core to compensate the inserted reactivity and restore the balance. Then some LEP are actuated in the compensation regime. But the CPS monitoring system eventually will detect the local power distortions in several zones of the core and signal for reactor shut down was generated. Some delay of the signal was imposed, due to the use of the single failure assumption. Consequently, the reactor power is running down. Obviously, that STEPAN results demonstrate the similar tendency of the transient evolution.

The behavior of total reactivity reflects the same tendencies in CPS responses. Reactor emergency shutdown initiated fast decline in reactivity. Due to proper CPS operation during the transient the reactivity did not reach the limiting value.

More detailed analysis of the RBMK-1500 reactor behavior was performed for the fuel channel of the highest power. The maximal values of chosen local parameters (fuel and cladding temperature, heat density) were monitored. The results, obtained with CORETRAN, demonstrate, that the largest values of these parameters are observed just before the actuation of the reactor emergency protection system. Afterwards the parameters are going down rapidly. It is necessary to note, that obtained maximum values of monitored parameters do not violate the established acceptance criteria. So the maximum fuel temperature reached 1325 ^{o}C and maximum cladding temperature - 347 ^{o}C during the calculations of the withdrawal of control rod bank in core central part and 1290 ^{o}C and 346 ^{o}C respectively for the case, when the control rod bank withdrawal in core periphery was calculated.

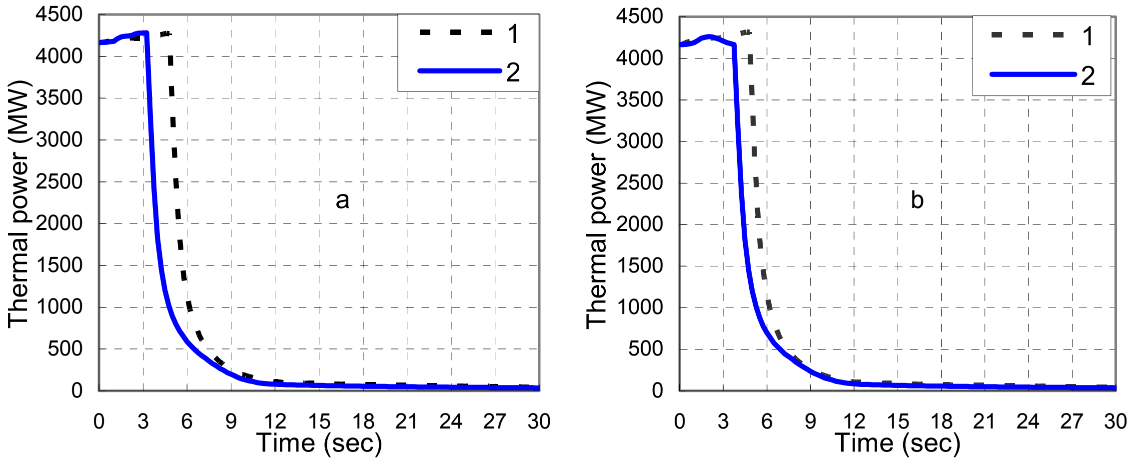

Fig.3 Total reactor power behavior during MCR bank withdrawal:
a - in the center of core; b – in the periphery of core; 1 – STEPAN; 2 –CORETRAN

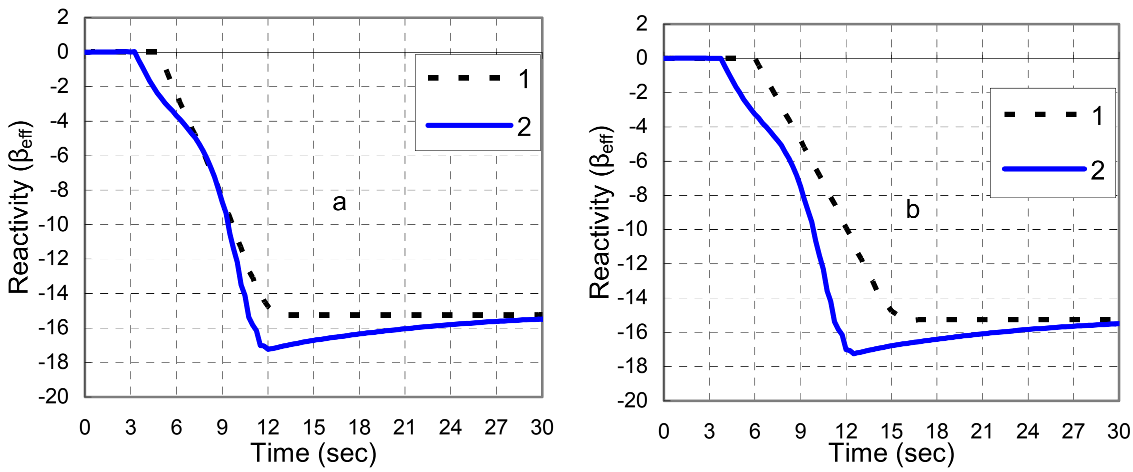

Fig.4 Total reactivity behavior during MCR bank withdrawal:
a - in the center of core; b – in the periphery of core; 1 – STEPAN; 2 –CORETRAN

According to STEPAN calculation the reactor emergency shutdown system will be actuated later compared to the CORETRAN calculation results. Actuation of emergency protection system in the cases investigated above is a result of local neutron flux distortions. Since the local power distortions in CORETRAN simulation are more significant as compared with

STEPAN, CPS initiated the shutdown signal earlier during the CORETRAN calculations.

During the present study the case of shortened absorber rod dropout was investigated in more detail, by benchmarking the CORETRAN results with STEPAN code and with sub channel analysis of the fuel channels, performed using VIPRE-02 code under single failure and ATWS assumptions. Results of the investigation are presented on Figures 5-13.

Neutron flux behavior can be analyzed by evaluating reactivity. Reactivity, in turn depends on variations of core parameters. Such analysis was provided for CORETRAN calculations results (see Figure 5). The reactivity change depends mostly on the amount of reactivity, inserted with the CPS rods, but influence of other components should be taken into account because of their contribution to total reactivity dynamics.

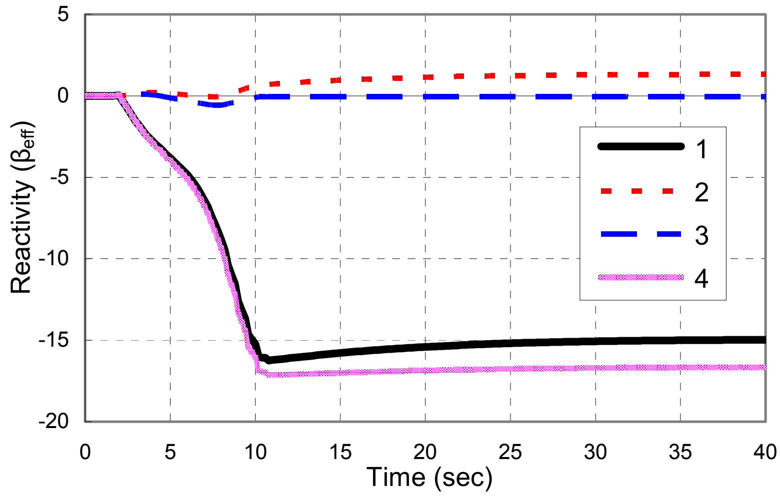

Fig.5 Total reactivity dependence on separate factors:
1- total reactivity, 2- fuel temperature (Doppler), 3 – coolant density, 4 – control rods.

CORETRAN calculations using the single failure approach demonstrated, that the local power distortions, initiated during this event, lead to the emergency shutdown mode actuation and reactor scram (see Figure 6).

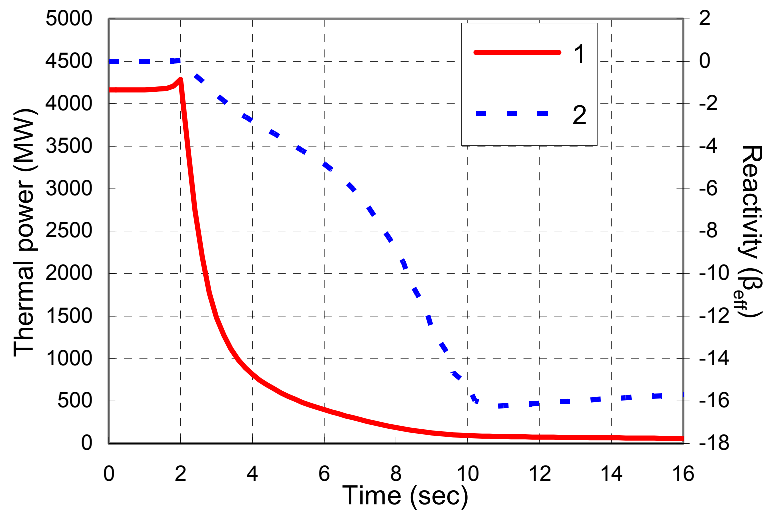

Fig.6 Simulation of drop out of the single short control rod:
1 – total power, 2 – reactivity.

More controversial results were obtained, while considering the ATWS (Figures 7 and 8).

In this case, according the STEPAN results, CPS maintains reactor in stable operation without the actuation of the emergency power reduction (Figures 7,8 curve 1). According CORE-TRAN results the same accident would initiate the emergency power reduction (Figure 7 curve 3 and Figure 8 curve 2). This could be explained as being the result of the above-described differences between the codes. The dropout of the shortened absorber rod perturbs the neutron flux in the lower part of the core. Local axial power distribution during the steady state, calculated by CORETRAN has much higher power level in comparison with STEPAN (see Figure 2 curves 2,1). Insertion of reactivity by shortened absorber rod dropout induces the rise of reactivity and power. The rise is much higher in CORETRAN calculations compared to STEPAN (see Figure 2 curves 4,3).

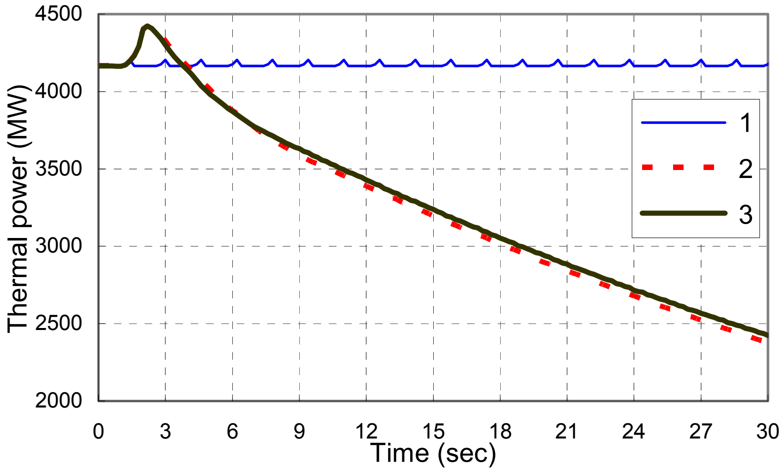

Fig.7 Total reactor power behavior during shortened absorber rod dropout:
1 – STEPAN, 2 – CORETRAN without feedback, 3 - CORETRAN with feedback.

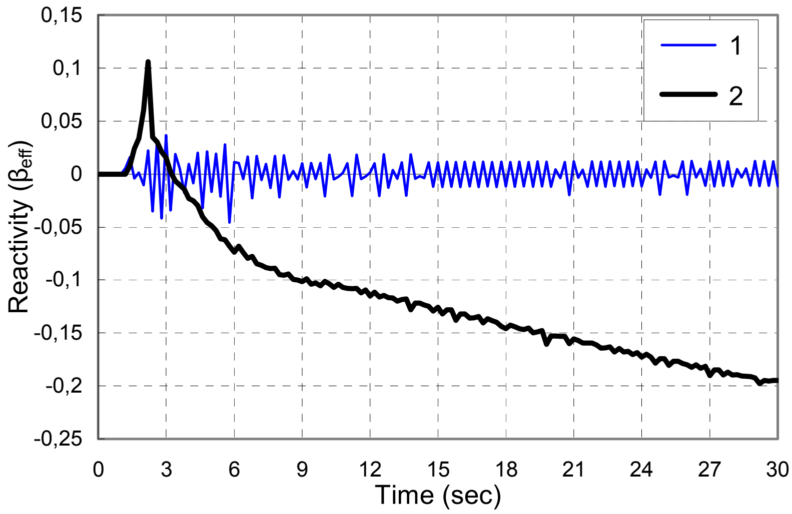

Fig.8 Total reactivity behavior during shortened absorber rod dropout:
1 – STEPAN, 2 – CORETRAN.

During the transient calculations the effect of the feedbacks on the results was investigated with CORETRAN. Thermal-hydraulics calculations indicate some influence of feedbacks on the total power level during the transient course (see Figure 7 curves 2 and 3).

It is necessary to get as accurate result representation as possible in each calculated RIA case. The neutronic module of CORETRAN (ARROTTA) has ability to represent the thermal-hydraulic parameters to some extent of accuracy, however it could not compete with any detailed thermal-hydraulic calculations code, since the thermal hydraulics homogeneous equilibrium model implemented in ARROTTA is used mostly to provide the appropriate reactivity feedbacks for the neutron dynamics calculations. Due to this reason in our study another CORETRAN package module – subchannel analysis code VIPRE-02 was employed to carry out more detailed investigations of the acceptance criteria in the RBMK-1500 fuel bundles [12].

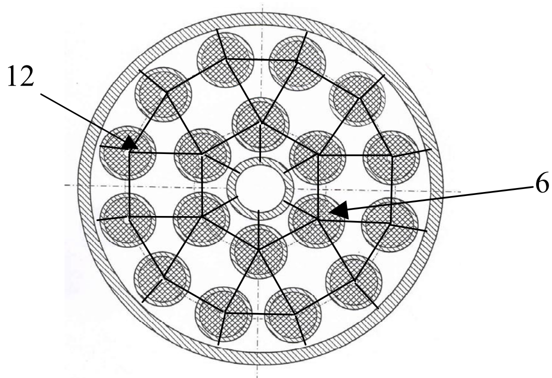

Fig.9 Cross section of the RBMK-1500 fuel channel model with VIPRE-02.

The solution method, used for the VIPRE-02 calculations was the two-fluid solution (six equation model) with iterative solution of continuity equations. For the VIPRE-02 calculations, the change of the axial power profiles during the transient was supplied as a boundary condition to the thermal hydraulic model from the CORETRAN neutronic calculations. The changes in the rod surface temperature during the transient were analyzed with VIPRE-02.

Figure 10 presents the axial power profiles of the four fuel assemblies under investigation prior to the transient.

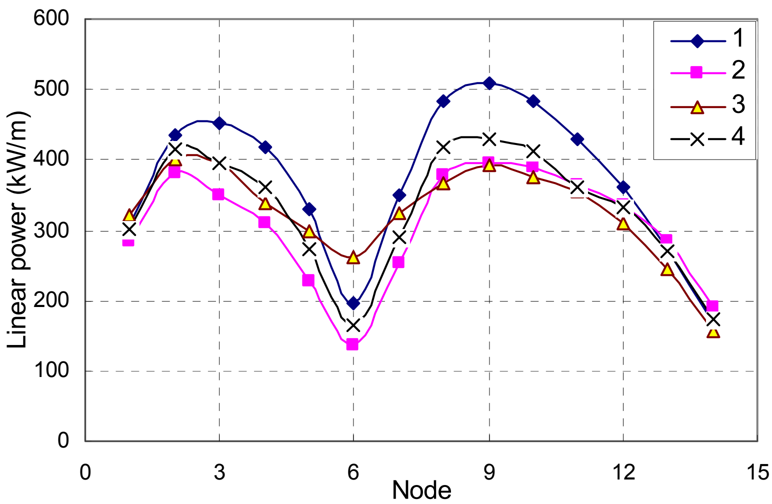

Fig.10 Initial axial power profiles at fuel assemblies related in INPP coordinates:
1 –41 – 32; 2 –41 – 34; 3 –39 – 33. 4 –42 – 33.

Fuel assembly with coordinates 41-32 was chosen to be investigated in more detail, since the power production at the beginning of the transient (as well as during the transient) for this fuel channel was highest of the four fuel channels, which surround the dropped shortened absorber rod. The change of the axial power profile for the assembly 41-32 is presented in Figure 11 for different times during the transient. As it is seen from this Figure, the peak power production was reached at transient time t=1.6 sec, i.e. as soon as the rod was dropped out from the core. Later on, the power level was reduced, due to the movement of CPS rods within the core.

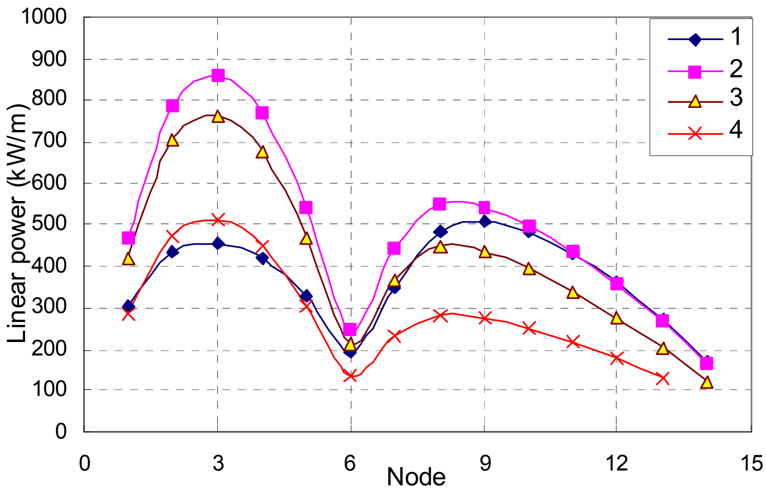

Fig.11 Linear power in fuel channel 41-32. Transient time: 1 – 0 sec, 2 – 1.6 sec, 3 – 7 sec, 4 – 40 sec.

The coolant flow rate in the fuel channel was not changing during the transient and for the channel 41-32 the coolant mass flux was 2812.8 kg/m^2·s, i.e. close to a nominal coolant flow through an average power fuel channel.

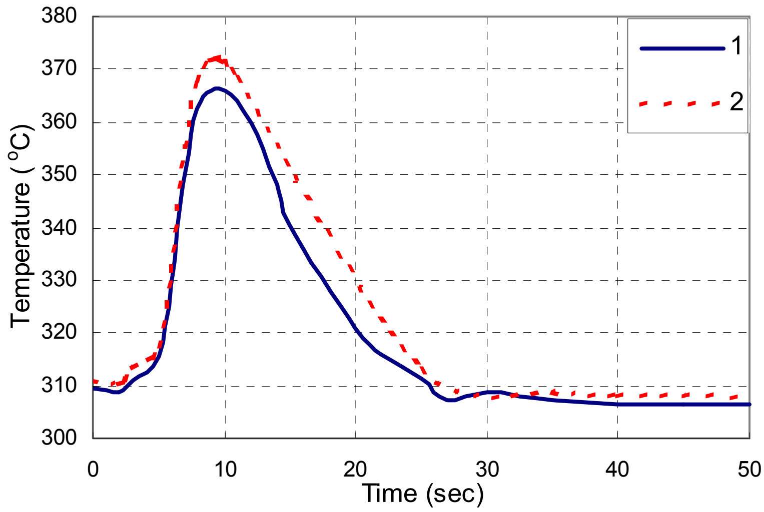

Fig.12 Peak cladding temperature in fuel rod. Number of rod: 1 – 12; 2 – 6 (see Figure 9).

The results of the VIPRE-02 calculations for the peak cladding temperature in the inner and outer rings of the fuel rods (i.e. rods 6 and 12 in Figure 9) are presented on the Figure 12. As it can be seen, the higher cladding temperatures were predicted for the fuel rods in the inner ring of the fuel assembly. The peak cladding temperatures during this transient did not reach the safety limit (700 ºC) during this transient. The peak temperature for the cladding was reached at transient time t=10 sec (Figure 12) and was equal to 371°C for the rod No. 6.

The axial temperature profiles for various times during the transient for the fuel rod No.6 (at the inner fuel rod ring of the RBMK-1500 fuel assembly) are presented on Figure 13. After the initial temperature increase at the beginning of the transient, following the drop-out of the control rod, the cladding temperatures started to reduce and reached the initial values (equal to the temperature before the transient started) at t= 50 sec for the lower part of the fuel assembly. For the upper fuel assembly part, the cladding temperatures started to reduce soon after the beginning of the transient, due to the reduction of thermal flux in the upper part of the core.

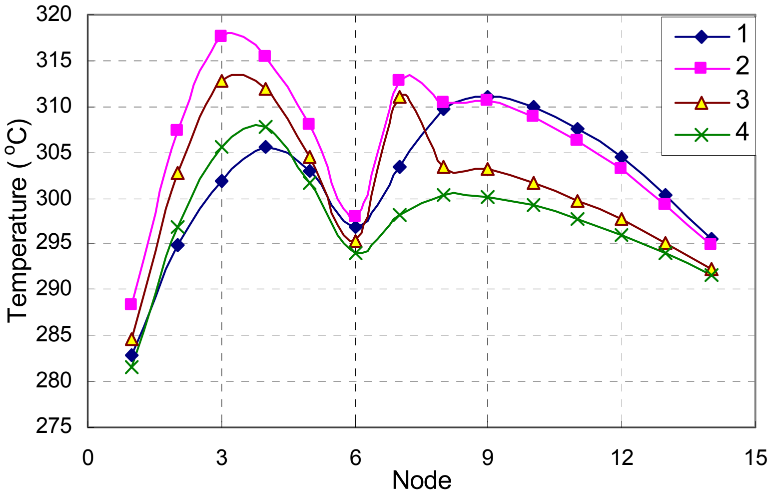

Fig.13 Axial cladding temperature profile for the fuel rod No.6. Transient time: 1 – 0 sec, 2 – 5 sec, 3 – 25 sec, 4 – 50 sec

4. Conclusions

This paper presents the results of RBMK-1500 reactor neutron dynamics calculations and analysis, carried out at the Nuclear Power Safety Division of the Royal Institute of Technology, Stockholm together with Department of Thermal and Nuclear Energy of Kaunas University of Technology. Studies on three reactivity-initiated transients were performed.

Spontaneous withdrawals of control rod bank in the central and peripheral part of the core and release of one shortened absorber rod from the reactor core were simulated. The calculations were carried out using WIMS-D4 code generated neutron cross-section library.

The results of the study demonstrated the capability of the Control and Protection System to mitigate the investigated reactivity-initiated transients, and to maintain the relevant core parameters within the safety margins.

As for thermal hydraulic subchannel analysis, the CORETRAN calculation results for the channel with most power during shortened absorber rod dropout transient were supplied as an input to the thermal hydraulic module of CORETRAN – VIPRE02 code. It was demonstrated, that safety margins for fuel cladding were not violated during the transient.

The partial benchmark of the CORETRAN calculations with the STEPAN code was provided. Some differences in initial steady state results and related transients courses were indicated. The further benchmark studies would provide additional confidence in the calculation results.

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