CORETRAN CODE APPLICATION FOR RBMK REACTOR SAFETY ANALYSIS

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ABSTRACT

The subject of this paper is RBMK-1500 reactor neutron kinetics analysis. The independent methodology and analysis tools were developed at the Nuclear Power Safety Division at Stockholm Royal Institute of Technology (RIT). The methodology employs Western codes for the RBMK reactor calculations. The two group neutron cross sections are calculated using the HELIOS code, where exact geometry of the various assemblies is employed. The core neutron dynamics calculations are performed using CORETRAN code, which employs neutron cross sections, generated with the HELIOS code. CORETRAN code also includes a thermal hydraulic module VIPRE02, which is capable of carrying out rod bundle thermal hydraulic analysis employing various thermal hydraulic models (from homogeneous equilibrium model to 6-equation model).

This paper describes the methodology of the analysis and its validation against the measured data. The results of the CORETRAN code calculations employing the HELIOS code generated neutron cross section data are compared to the measurements at the Ignalina RBMK-1500 nuclear power plant (NPP). The data include the series of measurements that are normally carried out at cold and hot reactor states. The former include subcriticality and control rod worth measurements and the latter include the power and void reactivity coefficients. Comparisons are also made when WIMS code calculated cross sections are employed, instead of those generated using the HELIOS code. In addition to this, reactor transient calculations, which are usually included in the scope of the Safety Analysis Report (SAR), were carried out and the results obtained are discussed in the paper.

1. INTRODUCTION

1.1 RBMK-1500 REACTOR

The RBMK reactors are channel type, water-cooled and graphite moderated reactors. The first RBMK type electricity production reactor was put on-line in 1973. Currently there are 13 operating reactors of this type. Two of the RBMK-1500 reactors are at the Ignalina NPP in Lithuania.

The core of RBMK-1500 reactor is a large graphite stack (11.8 m in diameter and 7 m high). The stack is penetrated by 2052 channels. Most of these channels (the total number is 1661) contain fuel assemblies. The fuel assembly consists of two parts: upper and lower, which are placed one above another in the reactor fuel channel. There are 18 fuel pins in each fuel assembly. Besides the fuel channels there are certain number of so-called 'special purpose channels' in the core. These channels contain control rods, various detectors, etc [1]. The reactor fuel is a slightly enriched uranium oxide (2.4% enrichment, containing 0.41% of the burnable poison Erbium). The fuel enrichment was increased from initial 2% level due to the safety modifications after the Chernobyl accident.

RBMK reactor power is controlled by the Control and Protection System (CPS). The system consists of two sub-systems: the standard reactor control system and Fast Acting Scram System. The CPS is cooled by an independent water circuit. In total there are 211 channels with control rods. There are 3 types of control rods: Manual Control Rods (MCR), Shortened Control Rods (SCR) and Fast Acting Scram Rods (FASR). The MCRs and FASRs are inserted from the top of the core while the SCRs (which are employed to control axial power distribution along the height of the core) are inserted from the bottom. The absorber used in the control rods is B_4C and Dy_2TiO_5.

The core of the RBMK-1500 reactor is divided into 12 Local Automatic Control (LAC) and Local Emergency Protection (LEP) sectors [1]. The twelve LEP sectors overlap and fully cover the entire core. The signals from the sensors in these sectors are processed by the Power Density Distribution Monitoring System (PDDMS). This system generates the signals that are used by the CPS to control the reactor power. PDDMS collects data from flux sensors, distributed discretely in the core. The completely processed information shows the radial and axial power distribution in the reactor, deviation from the power setpoints and other relevant reactor information.

1.2 AN OVERVIEW OF RBMK NEUTRON KINETICS CALCULATIONS

Two computer codes were developed in Russia for RBMK type reactor neutron kinetics calculations. The Kurchatov Institute (KI) in Moscow has developed and uses STEPAN code [2]. The STEPAN code is coupled with KOBRA thermal hydraulic code to obtain the transient thermal hydraulic feedback or with KONTUR code for the steady state thermal hydraulic feedback. The Russian Research and Development Institute of Power Engineering (RDIPE) has developed the SADCO code [3], which is similar to the STEPAN code. Both codes employ the 2 group cross section libraries generated with the WIMS code.

The combinations of both the STEPAN - WIMS and the SADCO - WIMS codes in general are not able to predict accurately the measured radial power distribution in the INPP. Modifications are made in the cross sections to obtain better agreement with the measured power distributions. The STEPAN code changes the thermal cross sections for some assemblies while the SADCO code changes the assembly burnups, which change the cross sections for both groups, and also the axial positions of control rods. The correction procedures employed by these two codes, are not documented and lack transparency

An independent methodology for RBMK reactor physics calculations is being developed at the Royal Institute of Technology (RIT), Division of Nuclear Power Safety in Stockholm, Sweden. Western computer codes are applied for the RBMK reactor calculations. The core neutron dynamics calculations are performed using CORETRAN code, which uses neutron cross sections generated with the HELIOS code. No corrections are made to the two group cross sections generated with the HELIOS code. The aim of this paper is to present the methodology and to describe the CORETRAN code calculation results for various RBMK-1500 reactor conditions.

Some other Western codes, e.g. German QUABOX/CUBBOX 3D neutronics code coupled with ATHLET thermal – hydraulic code [4] are also used for the RBMK reactor analysis. The code also uses some recovery procedures. WIMS code generated cross section data are employed for the QUABOX/CUBBOX calculations [5]. Recently, RELAP-3D code is also employed for RBMK reactor calculations [6]. RC Kurchatov Institute in Moscow also employs Monte Carlo code MCNP and MCU [7].

2. RBMK-1500 NEUTRON DYNAMICS CALCULATION METHODOLOGY DEVELOPED AT RIT

The neutron cross-section variation model for the RBMK applications is based on the logic shown in the Figure 1. The CORETRAN code reads a file with HELIOS code calculated 2 group neutron cross-section data, recorded in the form of tables and performs computation of 2D polynomial coefficients. The coefficients are allocated in the dynamic computer memory during the time of CORETRAN calculations and are directly accessed each time when a recalculation of each cross section is performed during the transient computations [3].

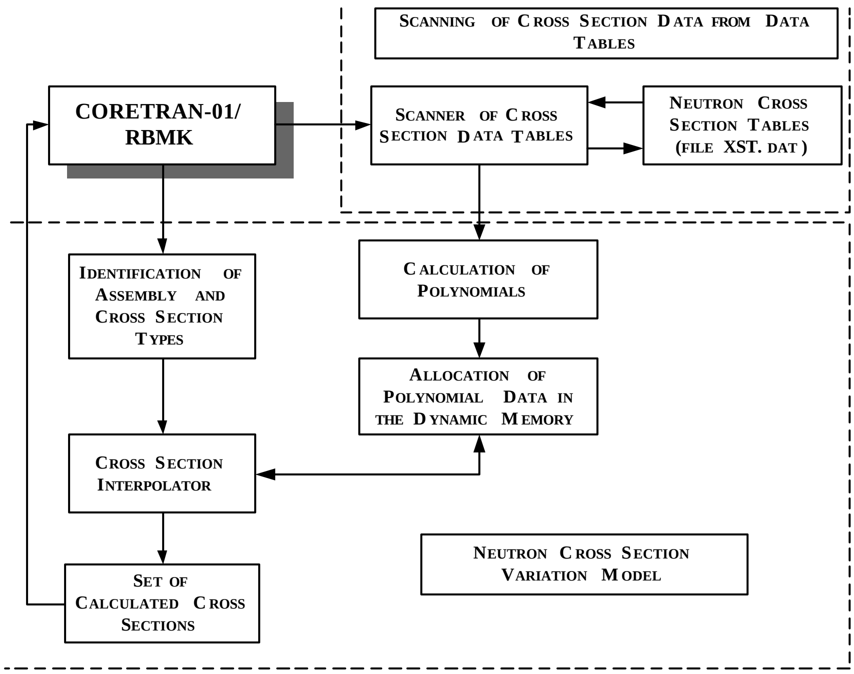

Figure 1. RIT cross section variation model for RBMK assemblies

The polynomial coefficients define 2D functions for each cross - section (XS) of each assembly type. These 2D functions are used to calculate basic components in the XS variation model. The basic components are calculated by running special interpolation procedure, which performs interpolation by using polynomial coefficients. The thermal-hydraulic parameters computed in a respective calculation node by the CORETRAN code are considered as the reference points for the calculation of the time dependent cross sections. The basic components are summed up and the final change in cross sections is obtained.

3. CORETRAN CALCULATIONS FOR RBMK-1500

The RBMK-1500 reactor calculations were carried out for cold and hot reactor conditions. Both series of steady state and transient calculations were carried out in order to assess the CORETRAN model for the wide range of the RBMK-1500 conditions. Two sets of neutron cross section data were used: neutron cross sections, calculated using HELIOS code and WIMS-D4 code generated cross sections. The HELIOS code generated data for RBMK-1500 were previously compared with WIMS-D4 generated data and validated against experimental results [3,8].

2.1 COLD REACTOR STATES

Some of the measurements of the RBMK reactor safety parameters are carried out during the reactor maintenance period, under the cold reactor conditions. One of the most important measurements in this phase is the reactor subcriticality under cold reactor conditions, after Xenon transient (i.e. about 120 hours after the reactor shutdown). These measurements are carried out on all RBMK type reactors at least once a year. The main safety requirement for these conditions is that the minimal reactor subcriticality remains under 1% (or under 2% during the reactor outage). During these measurements, the control rod efficiency is measured also.

This case was calculated with CORETRAN code using both HELIOS and WIMS-D4 code generated cross section data in order to compare the results of the calculation with the actual plant data.

The transient conditions were modeled according to the Report [9]: the reactor core state of 15 February 2001 was used as the initial state. The reactor was shut down. After 72 hours the reactor was made critical and the reactor subcriticality was evaluated. Then the reactor was shut down again and after 120 hours the second criticality measurement was carried out. The experimental data, as well as CORETRAN calculation results are presented in Table 1.

Table 1 Ignalina NPP-2 reactor subcriticality measurements for cold reactor conditions

	Time, hours	Experimental value	CORETRAN (HELIOS cross section data)	CORETRAN (WIMS-D4 cross section data)
FASS rod efficiency, β_{eff} Reactor subcriticality, β_{eff} (%) (when β_{eff}=0.0057)	72	1.98 5.6(3.2)	2.7 5.59(3.187)	- 5.03(2.87)
Reactor subcriticality, β_{eff} (%) (when β_{eff}=0.0057)	120	6.0(3.4)	5.54(3.16)	4.99(2.84)

In this case, a closer agreement with the measurements was obtained for CORETRAN calculations with HELIOS code generated neutron cross section data.

2.2 HOT REACTOR STATES

A series of validation calculations were carried out in order to assess CORETRAN code calculation results for the hot reactor states (i.e. under parameters, similar to reactor operational conditions). The results of the calculations were compared to the data, recorded in the Ignalina NPP reactor database (recorded data for the reactor state for each day is available, where the sensor readings allow calculating actual radial and axial power distribution in the operating reactor. This data allows testing the existing model under operating reactor conditions). The comparison of the radial power distribution is presented on the Figure 2.

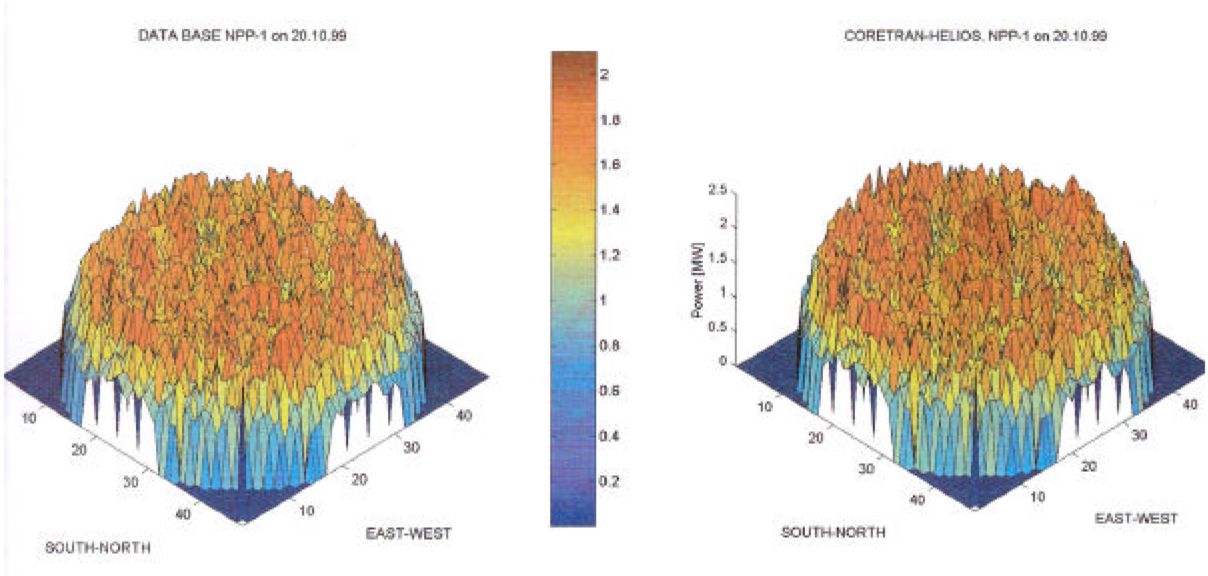

Figure 2. Radial power distribution in the INPP calculated with the CORETRAN code employing the HELIOS neutron cross-section library

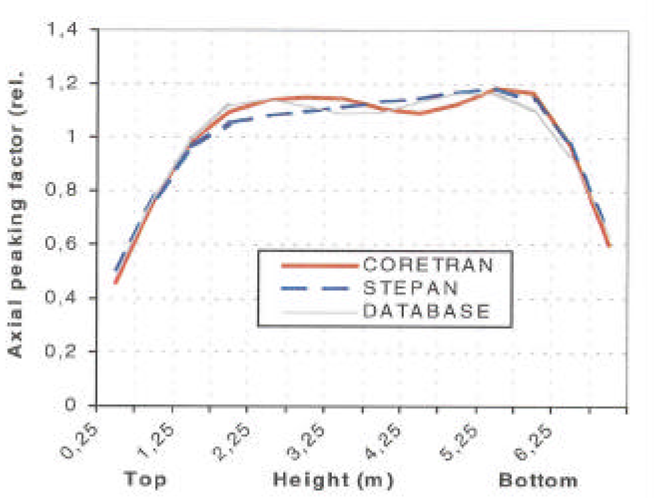

Figure 3 Averaged axial neutron flux distribution

Figure 3 presents a comparison of average axial neutron flux distribution. Here the CORETRAN calculation results are also compared to the STEPAN code calculated values. For this case, CORETRAN code provided more realistic representation of the axial flux distribution (the double-humped shape of the curve).

Table 2 Comparison of steady state calculation results (CORETRAN-HELIOS) with the Ignalina NPP database records (in brackets CORETRAN calculation results, using WIMS cross section data are presented)

Database	K_{eff} – *eigenvalue*	K_r – *radial peaking factor*	Average deviation of radial neutron field from measured, (%)	K_z - *axial peaking factor*
NPP-1 on 05.05.98	0.9947 (1.001)	1.46 (1.58)	9.1 (10.5)	1.27 (1.15)
NPP-1 on 17.03.99	0.9993 (1.002)	1.46 (1.52)	7.2 (11.9)	1.29 (1.28)
NPP-1 on 20.10.99	0.9976 (1.002)	1.46 (1.47)	5.9 (7.3)	1.19 (1.19)
NPP-2 on 17.03.98	0.9913 (0.9958)	1.49 (1.51)	9.5 (11.6)	1.28 (1.28)
NPP-2 on 01.10.98	0.9924 (0.9968)	1.53 (1.51)	10.3 (12.0)	1.29 (1.23)
NPP-2 on 26.11.98	0.9923 (1.001)	1.52 (1.47)	8.6 (10.5)	1.21 (1.22)
NPP-2 on 29.03.99	0.9954 (0.9985)	1.87 (1.78)	16.0 (14.4)	1.24 (1.17)
NPP-2 on 21.12.99	0.9953 (0.9997)	1.79 (1.68)	15.5 (13.0)	1.18 (1.19)

Table 2 provides comparison of the results of CORETRAN calculations carried out for various reactor states. Here, radial and axial peaking factors, as well as average deviations of the calculated results from the database data are provided.

Also, during the operation of the Ignalina NPP, at periodic intervals measurements of some important operational parameters, e.g. steam void coefficients are performed. A series of such calculations were performed at RIT, comparing results obtained with the CORETRAN code against data from both reactors at the Ignalina NPP (INPP-1 and INPP-2).

Table 3 presents the comparison of the measured data at the Ignalina NPP (for Unit 1 and Unit 2) with CORETRAN code results. The calculations using new HELIOS neutron cross section library were also compared against the same 3-D neutron kinetics calculations, but using cross section library, generated using WIMS code, which is already widely used in RBMK reactor applications, and against STEPAN code results. In most cases, the calculations, performed with new HELIOS cross-section library, give a better agreement with the experiment data. Here $\alpha_j, \alpha_w, \alpha_t, \alpha_C, \Delta\rho_{MCC}, \Delta\rho_{CPS}$ are respectively the void coefficient, the power coefficient, the Doppler coefficient, the graphite temperature coefficient, the MMC voiding effect and the CPS voiding effect.

Table 3 Comparison CORETRAN and STEPAN results for RBMK-1500 reactor

Unit, Data	***Code***	α_φ, %	$\alpha_w \times 10^6$, 1/MWt	$\alpha_t \times 10^5$, 1/°C	$\alpha_c \times 10^5$, 1/°C	$\Delta\rho_{MCC}$, %	$\Delta\rho_{CPS}$, %
2-29.03.99	STEPAN	0.30	-2.30	-1.10	4.10	0.40	1.40
____ "_____	CORETRAN-WIMS	0.49	-1.40	-1.75	4.12	0.75	1.34
____ "_____	CORETRAN-HELIOS	0.52	-1.38	-	-	-	-
____ "_____	Experiment	**0.54**±0.12	**-1.26**±0.12	-	-	-	-
2-21.12.99	STEPAN	0.20	-2.50	-1.00	4.00	0.20	1.20
____ "_____	CORETRAN-WIMS	0.52	-1.32	-1.75	4.19	0.58	1.17
____ "_____	CORETRAN-HELIOS	0.34	-1.51	-1.76	3.60	0.25	0.76
____ "_____	Experiment	**0.36**±0.12	**-1.56**±0.12	-	-	-	

2.3 TRANSIENT CALCULATIONS

Two transient calculations are presented in this paper : spontaneous withdrawal of a control rod and Control and Protection System LOCA (loss of coolant accident). These transients represent two types of hypothetical accidents, which have to be analyzed also in the scope of the Safety Analysis Report (SAR) for the RBMK-1500 type reactor.

2.3.1 Spontaneous withdrawal of a control rod

The RBMK-1500 Control and Protection System (CPS) is divided into 12 local zones. In each local zone, there are one automatic regulator (LAR) and two Emergency Shutdown Rods (LAZ) rods. About 350 groups of detectors are in the reactor. Each group comprises of the 2 to 7 detectors.

Besides these detector groups, there are 3 groups of the out-of-core detectors, which monitor total reactor power. These detectors generate full reactor scram signal when the total change of the reactor power exceeds ~10 %. The control logic for the RBMK-1500 reactor control and protection system operation was coded into the CORETRAN. The spontaneous single control rod withdrawal transient allows us to evaluate the performance of the coded logic.

We have analyzed a spontaneous withdrawal of one control rod at the periphery of the core. For the reactor database, the core state of January 29, 2001 of the Ignalina NPP Unit-1 was chosen. The reactor was operating at 4156 MWth. The description of the calculated transient:

According to Figure 4, the control rod was fully withdrawn after about 15 seconds from the beginning of the transient. At approximately 7th second of the transient, AZ-3 control and protection signal was generated. This signal enables the 50% reduction of the reactor power. After about 37 seconds, new power level at 50% of the initial power was reached, and the further movement of the control rods stabilized the reactor at this level.

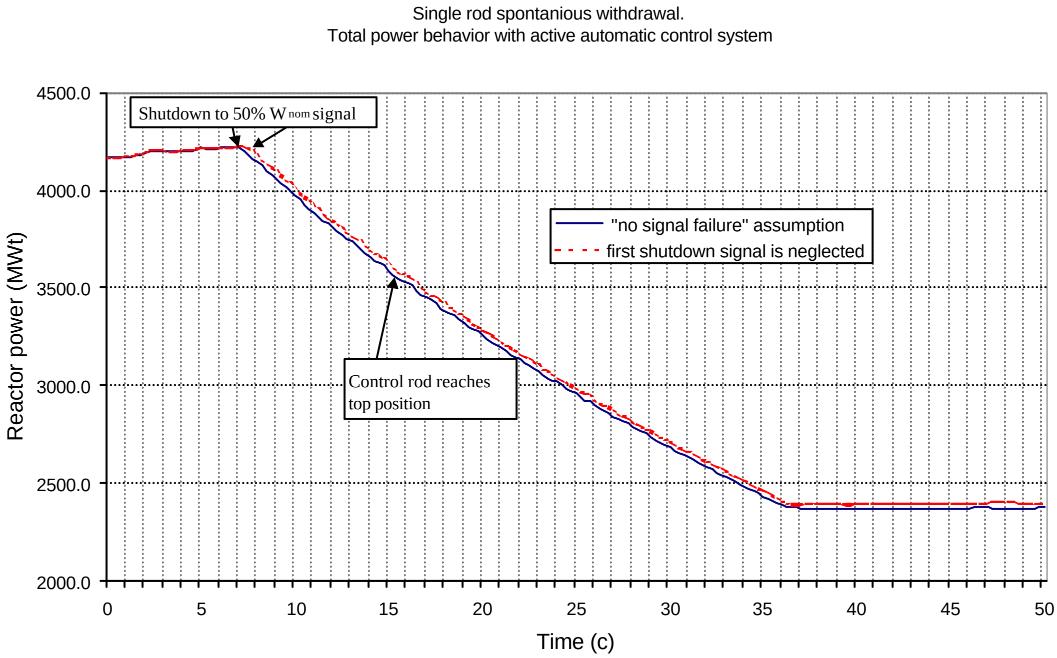

Figure 4 Reactor power behavior during CORETRAN calculation of spontaneous single control rod withdrawal.

All the process was calculated automatically, i.e. according the CPS logic, implemented into CORETRAN. CPS system ensured the stabilization of the reactor power at the new level of 50% (set-point values for the automatic regulation were automatically adjusted by the code for the new power level).

The calculations were repeated assuming ATWS (anticipated transient without scram) conditions. For this case, first shutdown signal, generated by the control and protection system was neglected. This led to the generation of a second CPS signal 0.5 sec later (Figure 4, red line). After the delay, the transient proceeded in a similar way as during the first case: the reactor power was reduced by 50% and stabilized at the new power level.

2.3.2. Control and Protection System Loss of Coolant Accident (CPS LOCA)

This transient is considered to have the most positive reactivity input in the RBMK type reactors. Although the exact time, during which all the coolant is removed from the CPS system, is not defined, it is usually assumed that the CPS system water level reaches the bottom of the RBMK reactor core in about 45 seconds (assuming that at the time zero the CPS coolant was at the core top level in the control and protection system). However, no precise data about the exact duration of the transient and the conditions (e.g. the distribution of the flow rates in various parts of the core) are available.

During the CORETRAN calculations, the following assumptions were made :

- All CPS channels lose coolant at the same rate (i.e. the coolant level reduces with the same rate in all regions of the core)
- Technological set points, according to which the reactor scram signal is generated are neglected. The transient starts at time zero, i.e. the moment when coolant level in the CPS system reaches the top of the reactor core.

- LAC - LEP system is operating.

The CORETRAN calculation results are presented in the Figure 5.

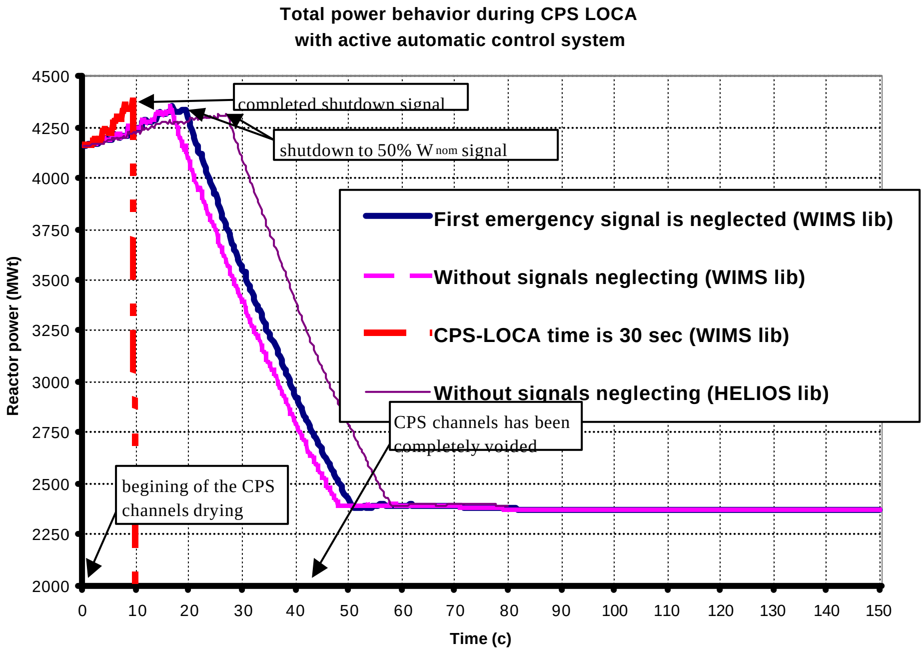

Figure 5 CORETRAN calculated reactor power during CPS LOCA transient

The CORETRAN calculations were conducted using two sets of neutron cross section libraries: generated with HELIOS and generated with WIMS-D4 codes. During the transient, at about 17 second AZ-3 control and protection signal is generated. This signal empowers the reduction of the reactor power by 50% of the initial value. At about 50 seconds of the transient, the reactor power is stabilized at the new level. Further movement of control rods keeps the new reactor power level constant.

The calculated case implies, that the control and protection system of the RBMK-1500 reactor is capable of handling even the transient with the most positive reactivity addition without having to scram the reactor. I.e. LAR and LAZ systems ensure the operation of the reactor at the reduced power level even after the coolant is no longer present in the control and protection system circuit.

The CORETRAN calculations were repeated employing both HELIOS and WIMS-D4 cross section data, assuming shorter loss of coolant time for the CPS circuit. It was assumed that the water level reaches the bottom of the reactor core at 30 seconds after the beginning of the transient (Figure 5). For this case, CORETARN calculations with WIMS cross section data predicted the occurrence of AZ-1 (full reactor scram signal) at about 10 seconds after the beginning of the transient, due to the CPS signal, initiated by the increase of the reactor power by more than 10%. The use of the HELIOS cross section date led to the delay in the generation of the emergency protection signal. This could be explained by the difference in the cross section data, generated with HELIOS and WIMS-D4 codes: HELIOS code predicts higher absorption cross sections for the control rods, which leads to higher 'weights' of the rods.

Analogous to the first transient, analyzed in this paper, the ATWS case for the CPS LOCA accident was also investigated. First signal, generated by the control and protection system was neglected during the CORETRAN calculations (Figure 5). This led to the delay of the system response by 2 seconds. Otherwise, the reactor power was stabilized at the new level of 50% initial power.

2.3.3. Thermal hydraulics analysis with VIPRE02 code

The results of CORETRAN calculations for the CPS LOCA transient were employed in the subchannel thermal hydraulic analysis of the fuel channel with the highest power. The thermal hydraulics calculations were carried out using the thermal hydraulic module of the CORETRAN code – VIPRE02 code. During the routine RBMK-1500 calculations with CORETRAN code, in order to reduce the calculation time, one fuel channel is represented as one thermal hydraulic channel. However, subchannel analysis is also possible with VIPRE code. For this purpose, channel with coordinates 14-25 was chosen to be investigated, as during the CPS LOCA calculations, the highest radial power was detected in this channel. The purpose of the calculations was to determine whether the limits of the protection set points were not exceeded, i.e. to evaluate the integrity of the fuel assembly.

In order to assess the fuel cladding integrity, one needs to investigate the maximum fuel cladding temperatures during the transient. In order to do that, the VIPRE code input was developed for a single RBMK-1500 fuel channel (Figure 6). The fuel channel was divided into 30 square and triangle subchannels. Results obtained during the CPS LOCA transient calculations were used for the thermal hydraulic analysis. From the previous studies [10] it was shown, that Bowring CHF correlation gives the closest predictions for the critical heat flux occurrence in long tubes (i.e. conditions, similar to RBMK-1500).

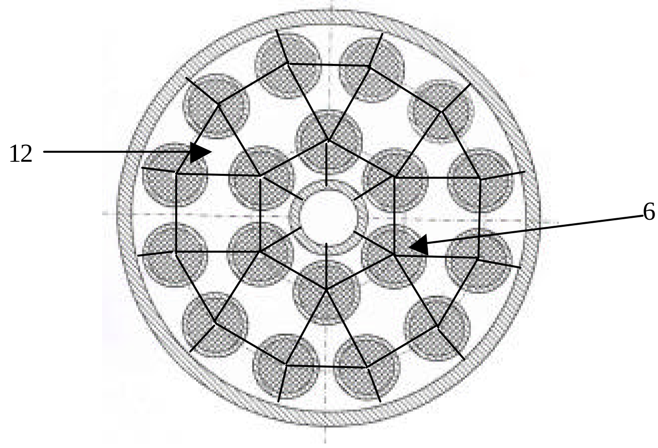

Figure 6 Cross section of the RBMK-1500 fuel channel

According to the CORETRAN results, the maximum local power in the fuel channel occurred at 27.5 seconds after the beginning of the transient. The maximum power value was 14 (W/cm^3)/node, which corresponds to 37789 W/cm^3 per fuel volume.

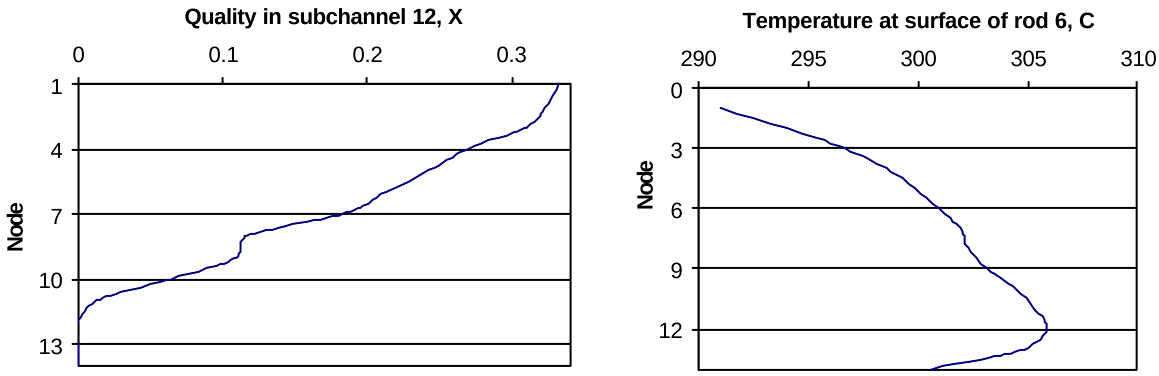

Figure 7 Quality and rod surface temperature profiles in the fuel channel of highest power

Figure 7 presents the VIPRE calculation results for the axial quality distribution for the hottest subchannel (which in this case was triangular subchannel between two rows of fuel rods, e.g. subchannel 12 in Figure 6). The highest temperatures during this transient were detected to occur at the inner part of the fuel assembly. As it is seen from the Figure 7, the maximum allowed temperature for the cladding (which is 700°C according to the RBMK-1500 regulations) was not reached during this transient.

CONCLUSIONS

This paper presents results of RBMK-1500 reactor neutron dynamics calculations, carried out at the Nuclear Power Safety Division of the Royal Institute of Technology, Stockholm. Three types of calculations were performed: cold reactor state calculations, hot reactor steady state calculations and calculations of two transients, considered under the scope of Safety Analysis Report (SAR) for RBMK-1500 reactors.

As for the transient calculations, spontaneous withdrawal of one control rod and Control and Protection System Loss of Coolant Accident were simulated. The calculations were carried out using HELIOS and WIMS-D4 code generated neutron cross section libraries.

The cold and hot steady state calculations allowed carrying out the validation of the CORETRAN model for the RBMK-1500 reactor against the experimental reactor data. In general, CORETRAN code provides good agreement with the experimental results (and with STEPAN code calculations). CORETRAN code provides closer to experimental, values for the power coefficient, compared to the STEPAN code results.

Transient calculations showed, higher, than expected, level of performance of the RBMK-1500 reactor Control and Protection System: during both spontaneous withdrawal of one control rod and the CPS LOCA transient, the LAC and LEP control rods of the reactor proved to be capable of controlling these reactivity transients and reducing the power level to 50% in an orderly fashion. No full reactor shutdown was predicted.

As for thermal hydraulic subchannel analysis, the CORETRAN calculated results of the channel with most power during CPS LOCA transient were supplied as an input to the thermal hydraulic module of CORETRAN – VIPRE02 code. It was shown, that safety margins for fuel cladding were not violated during the CPS LOCA transient. However, as a future work, wider spectrum of thermal hydraulic transients should be investigated.

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