APPLICATION OF COUPLED RELAP5-3D CODE FOR ANALYSIS OF RBMK-1500 REACTOR NEUTRON KINETIC - THERMAL-HYDRAULICS TRANSIENTS

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ABSTRACT

This paper presents the application results of coupled RELAP5-3D code for the analysis of RBMK-1500 reactor neutron kinetic – thermal-hydraulic transients and discusses its suitability to model specific transients that take place during RBMK-1500 reactor operation, where the neutronic response of the core is important. A successful best estimate RELAP5-3D model of Ignalina NPP RBMK-1500 reactor has been developed and validated performing a set of transient calculations, where real plant data was available for comparison. Certain RELAP5-3D transient calculation results were benchmarked against calculation results obtained using the Russian code STEPAN, specially designed for RBMK reactor analysis. Comparison of the results, obtained using RELAP5-3D and STEPAN codes, showed reasonable mutual coincidence of the calculation results and their reasonable agreement with the real plant data.

1. INTRODUCTION

RELAP5 code originally was designed for PWR reactors to provide the US Government and industry with an analytical tool for the independent evaluation of reactor safety through mathematical simulation of transients and accidents. This paper presents the application results of coupled RELAP5-3D code for the analysis of RBMK-1500 reactor neutron kinetic – thermal-hydraulic transients and evaluation of RELAP5-3D code's suitability to model specific transients that take place during RBMK-1500 reactor operation, where the neutronic response of the core is important. Two benchmark problem analyses, that were performed during the validation of the best estimate RELAP5-3D model of the Ignalina NPP RBMK-1500 reactor and reported here are: feedwater flow perturbation and reactor power reduction transients. Both benchmarks were modeled using RELAP5-3D code and the calculation results compared to the calculation results obtained using STEPAN code, specially designed for RBMK reactor analysis, as well as to the real plant data registered by the TITAN information computer system at Ignalina NPP.

2. DESCRIPTION OF RELAP5-3D MODEL

The main purpose for using RELAP5-3D code was that RELAP5 MOD3.2 code was not capable to predict local effects taking place in such a big reactor core as that of RBMK-1500 reactor. RELAP5 MOD3.2 code uses point kinetics, but that was not sufficient for the modeling of the selected transients. The main advantage of RELAP5-3D code - suitability of the code to model specific transients that occur during reactor operation, where the detailed neutronic response of the core and the local power effects are important (in case of spontaneous control rod withdrawals, reactor power variations, feedwater perturbations, etc.).

2.1 THERMAL-HYDRAULIC PART OF IGNALINA NPP RELAP5-3D MODEL

The RBMK-1500 is graphite moderated, boiling water, multi-channel reactor. Several important design features of RBMK-1500 are unique and extremely complex with respect to Western reactors [1]. The general thermal-hydraulic nodalization scheme of the model is presented in Fig. 1. The model of the MCC consists of two loops, each of which corresponds to one loop of the actual circuit. The left half in the model is simplified. This half has one generalized MCP, GDH, and generalized steam DS (1). All downcomers are represented by a single equivalent pipe (2), further subdivided into a number of control volumes. The pump suction header (3) and the pump pressure header (8) are represented as RELAP5 “branch” [2] elements. Three operating MCPs are represented by one equivalent element (5) with check and throttling-regulating valves. The pumps are characterized by pump impeller angular speed and coolant flowrate through the pump. In the RELAP5 pump model the four-quadrant characteristics are expressed by so-called homologous curves [3]. The throttling-regulating valves are used for coolant flowrate regulation through the core. These valves are modeled by employing “servo valve” [2] elements. The normalized flow area versus normalized stem position is described in the RELAP5 model. The stand-by MCP is not modeled. The bypass line (7) between the pump suction header and the pump pressure header is modeled with the manual valves closed. This is in agreement with a modification recently performed at the Ignalina NPP. All fuel channels of this left core pas are represented by seven equivalent channels (12) operating at specific power and coolant flow. The group of 20 distribution headers (9) with connecting pipelines is modeled by RELAP5 “branch” component. The pipelines of the water communications (10) are connected to each GDH. Each of these components represents the quantity of pipes appropriate to the number of elements in the corresponding FC in the core. The vertical parts of the FC (13) above the reactor core are represented by RELAP5 components “pipes”. The pipelines of the steam-water communications (14) are connecting the fuel channels with DS. Compared to the model for the left loop, in the right loop, the MCP system is modeled with three equivalent pumps. The right loop model consists of seven equivalent core passes also. The CPS channels (16) and radial graphite reflector cooling channels (18) are modeled too. These channels are cooled by a separate water circuit (17).

The steam separated in the separators is directed to turbines via steam lines (15). Two Turbine Control Valves organize steam supply to the turbines. The control of these valves was modeled by “servo valve” [2] elements based on algorithm of steam pressure regulators used at Ignalina NPP, when one turbine operates in a power maintenance regime, and other – in pressure maintenance in DS regime. There are four Steam Discharge Valves in each loop of the MCC to direct the steam to the condensers of the turbines. The pressure of the steam is also controlled, and peaks of pressure are eliminated by two high pressure steam loops (one for each MCC loop). One Steam Discharge Valve to Accident Confinement System and six Main Safety Valves, which are connected to high pressure steam loop, discharge the steam to pressure suppression pool of the Accident Confinement System tower. The model also takes into consideration steam mass flowrate through the Steam Discharge Valve to the deaerator for in-house needs. All models of steam discharge valves are connected to the “time dependent” elements, which define boundary conditions in turbine condensers or ACS pressure suppression pool.

The feedwater injection into the DS is simulated explicitly using RELAP5 “pipe”, “junction”, “volume” and “pump” elements. The nodalization scheme of the feedwater system is not presented in this paper. The feedwater from the deaerators is supplied to the MCC by Main Feedwater Pumps.

There are seven MFWPs. During normal conditions one pump is in stand-by and one pump can be out of service due to maintenance. The capacity of one MFWP is about 400 kg/s.

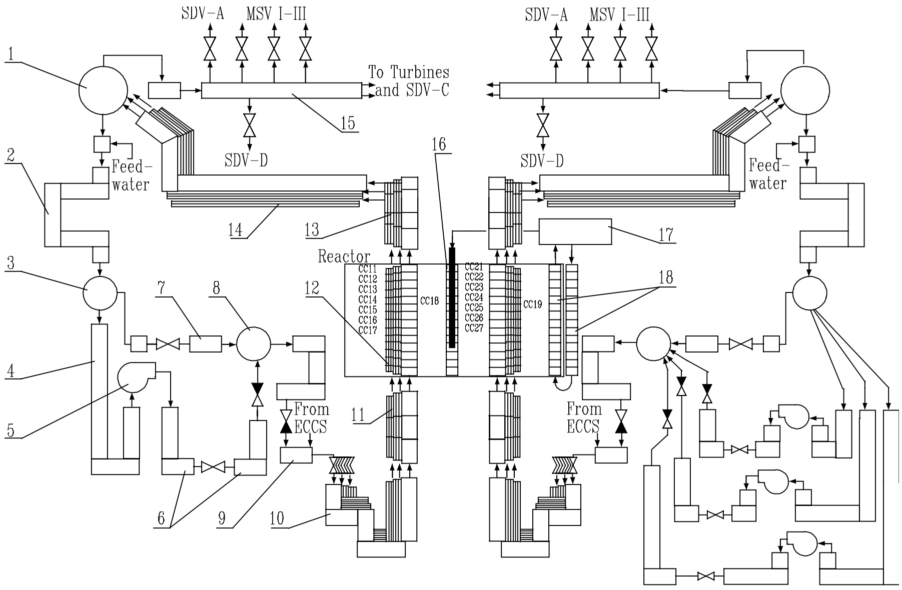


Figure 1. Ignalina NPP thermal-hydraulic model nodalization diagram: 1 - DS, 2 - downcomers, 3 - MCP Suction Header, 4 - MCP suction piping, 5 - MCPs, 6 - MCP discharge piping, 7 - bypass line, 8 - MCP Pressure Header, 9 - GDHs, 10 - lower water communication line, 11 - reactor core inlet piping, 12 - reactor core piping, 13 - reactor core outlet piping, 14 - Steam-Water Communication line, 15 - steam line, 16 - CPS channel, 17 - CPS channels cooling circuit, 18 - radial graphite reflector cooling channels

The reactor core is modeled by 14 RELAP5 pipe components, each of which represents a separate group of FC. Seven RELAP5 “pipe” components represent the 835 FC in the left loop and seven RELAP5 “pipe” components represent the 826 FC in the right loop. The distribution of FC in both MCC loops is shown in Table 1 (for INPP Unit 2 reactor core states on November 26, 1998 and March 29, 1999). Square profile 0.25 x 0.25 m graphite blocks are modeled by cylindrical elements with the equivalent cross-section area. The heat structure of the equivalent fuel channel simulates not only active region in the reactor core, but the top and bottom reflectors are modeled as well. Each equivalent channel is modeled using 16 axial nodes of 0.5 m length each. The fuel element is modeled using eight radial nodes, five to represent the fuel pellet, one for the gap region and two for the cladding. The fuel channels and graphite columns are modeled using eight radial nodes. Two of these radial nodes are for the fuel channel wall, two for the gap and graphite rings region and four for the graphite column.

2.2 NODAL KINETICS PART OF IGNALINA NPP RELAP5-3D MODEL

The RBMK-1500 reactor core has a 7.0 m fuel region and a 0.5 m reflector region above and below the fuel region. The overall height of the core region is 8.0 m. The neutronics mesh represents each rectangular graphite column as one individual stack in the radial plane. The reactor core region in the RBMK-1500 RELAP5-3D model has 32 axial nodes (0.25 m each) and 56x56 nodes (0.25 m each) in the radial plane. This mesh results in 28 axial nodes in the fuel region and 2 axial nodes in each of the top and bottom reflector region. In thermal-hydraulic model of the reactor core we have 16 thermal-hydraulic meshes: 14 nodes (0.5 m each) in the fuel region and 1 node in each of the top and bottom reflector region. In this way the height of the two neutronics nodes are equal to the height of one thermal-hydraulic node.

The two developed models of nodal reactor kinetics are based on the two real states of the reactor of Ignalina NPP Unit 2, registered by ICS “TITAN” on November 26, 1998 and on March 29, 1999. Reactor core loading information was obtained from the plant as a part of the database from the main information computer system “TITAN”. Besides the reactor core loading information, the database provided the following information that was used in RBMK-1500 RELAP5-3D model: insertion depth of the CPS control rods, burnup of each of the fuel assemblies, axial fuel burnup profile, coolant flowrate maps of the MCC and the CPS cooling circuit. Radial fuel assemblies burnup profile and axial relative fuel burnup profile were input into the model as user input variable.

Table 1. Summary specification of the thermal-hydraulic channel groups as being modeled in the RBMK-1500 reactor RELAP5-3D model

INPP Unit 2; November 26, 1998					INPP Unit 2; March 29, 1999				
Ch. group specific.	Reactor side	No. of ch.	Av. power in ch., MW	Av. flowrate in ch., m^3/h	Ch. group specific.	Reactor side	No. of ch.	Av. power in ch., MW	Av. flowrate. in ch., m^3/h
CC11	Left	355	2.95	28.2	CC11	Left	304	1.5	28.2
CC21	Right	378	2.95	28.2	CC21	Right	308	1.5	28.2
CC12	Left	249	2.5	26.2	CC12	Left	301	1.3	25.6
CC22	Right	234	2.5	26.2	CC22	Right	305	1.3	25.6
CC13	Left	60	2.4	25.1	CC13	Left	55	1.1	24.6
CC23	Right	59	2.4	25.1	CC23	Right	51	1.1	24.6
CC14	Left	59	1.8	21.1	CC14	Left	65	0.8	19.7
CC24	Right	55	1.8	21.1	CC24	Right	63	0.8	19.7
CC15	Left	39	1.6	17.5	CC15	Left	38	0.8	16.1
CC25	Right	37	1.6	17.5	CC25	Right	35	0.8	16.1
CC16	Left	61	1.2	15.6	CC16	Left	60	0.6	14.3
CC26	Right	70	1.2	15.6	CC26	Right	71	0.6	14.3
CC17	Left	3	1.8	33.5	CC17	Left	3	0.8	34.0
CC27	Right	2	1.8	33.5	CC27	Right	2	0.8	34.0
CC18		235			CC18		235		
CC19		592*			CC19		592*		

* 436 channels are radial reflector channels

Cross sections for the different compositions of the RBMK-1500 reactor core were obtained from two-group macro x-section library of the STEPAN code that was provided by Russian Research Center “Kurchatov Institute”. X-section library includes subroutines for fuel cells, non-fuel cells and the CPS control rods. An external user subroutine interface was written that accesses the coding of the RRC “KI” x-section library subroutines at each time step of the calculation. The interface receives thermal-hydraulic and control rod position information from the RELAP5-3D code and provides input to the RRC “KI” x-section library subroutines. X-section library subroutines return the diffusion, absorption, fission and scattering x-sections for the two neutron groups. The interface then transfers the obtained x-sections to the NESTLE code kinetics solver that is part of the RELAP5-3D code.

As previously described, the reactor core is divided into two halves with 7 thermal-hydraulic channels per core half. There are 2 additional thermal hydraulic channels that model 1) radial reflector and radial reflector cooling channels lumped together, and 2) the CPS cooling circuit channels lumped together. Therefore, the reactor core has 14 thermal-hydraulic channels for the fuel channels and 2 thermal-hydraulic channels for the non-fuel channels. The fuel channels were divided into 7 groups according to power and coolant flowrate values. Table 1 show the assignment of thermal-hydraulic channels to each group for both reactor core states mentioned above. The kinetics part of the model models each fuel and non-fuel channel individually. The number of channels in each group varies from 2 to 378 and from 2 to 308, respectively. 'CC' represents the thermal-hydraulic axial mesh number. Channel groups CC11, CC12, CC21 and CC22 are located in the center of the reactor core, while the remaining groups are on the periphery. The CC18 CPS channel group is distributed evenly all through the reactor core. The CC19 channel group represent radial reflector and radial reflector cooling channels group.

Another complicated part of RBMK-1500 reactor RELAP5-3D model is the CPS control rods and the CPS operation logic. All CPS 211 control rods are modeled individually, because all of them have different insertion depths into the reactor core. Four types of control rods are modeled: 2091 mod. manual control rods, 2477 mod. manual control rods, fast acting control rods and short absorber control rods. The first three types of control rods are inserted from the top of the reactor core, while the fourth type of control rods is inserted from the bottom. RELAP5-3D control variable system is used for CPS logic and CPS control rod movement modeling. Movements of the CPS control rods are controlled by the CPS logic, based on the power deviation signals coming from 127 radial detectors of the DKER-1 radial detector system. The DKER-1 detectors are modeled as having 7 sensitive elements (0.25 m each) distributed evenly over the height of the fuel region of the reactor core. Power deviation signal is based on the steady-state thermal neutron flux value in each detector location. All the detectors of the DKER-1 detector system are located in 12 local automatic control / local emergency protection (LAR/LEP) zones. In each LAR/LEP zone there is one LAR control rod and 2 LEP control rods. LAR and LEP rods move based on a certain percent deviation of the transient thermal neutron flux value from its initial value at the beginning of transient calculation.

2. VALIDATION OF RELAP5-3D MODEL

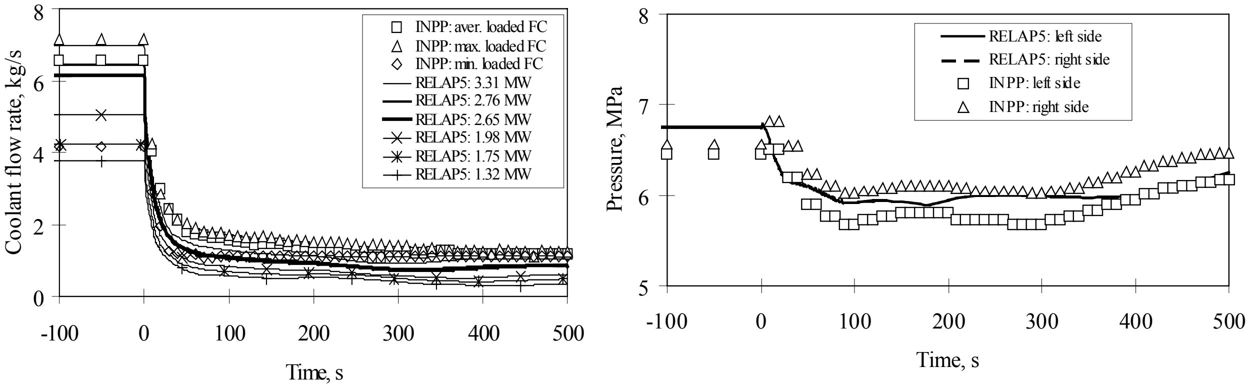
The state-of-the-art code RELAP5-3D was originally designed for Pressurized Water Reactors. Because of the unique RBMK design, the application of this code to RBMK-1500 encountered several problems. Comparison of the calculation results with the real plant data allowed to verify the suitability of the developed model for the future modeling of the processes, taking place in RBMK-1500 reactor during DBA and other different transient modes of the reactor operation.

In the process of development of the Ignalina NPP RELAP5-3D model for RBMK-1500 type reactors, comparative analyses of actual operational events are essential because this allows to establish realistic hydraulic resistances of different MCC components and realistic behavior of the controllers of the reactor systems. For this purpose the following RELAP5-3D benchmark analysis were performed:

- one and all operating MCPs trip events,
- three Main Safety Relief Valves LOCA event,
- inadvertent actuation of ECCS.

Thermal hydraulic part of Ignalina NPP RELAP5-3D model was validated against real plant transients and the validation results presented in [4]. The calculation results obtained using RELAP5-3D model on the Ignalina NPP specific base compare reasonably with the real plant data.

Some examples of the comparison of calculated results using RELAP5-3D RBMK-1500 model against actual transients data are presented below. In Fig. 2 calculated and measured coolant flowrate through individual FC and pressure in DS behavior for loss-of-all-MCPs event are presented.



a) b)

Figure 2. Loss-of-all-MCPs transient. a) Coolant flowrate through individual FC; b) Pressure in DS

In Fig. 3 the calculated ECCS water flowrate is compared against real plant data for inadvertent actuation of ECCS event.

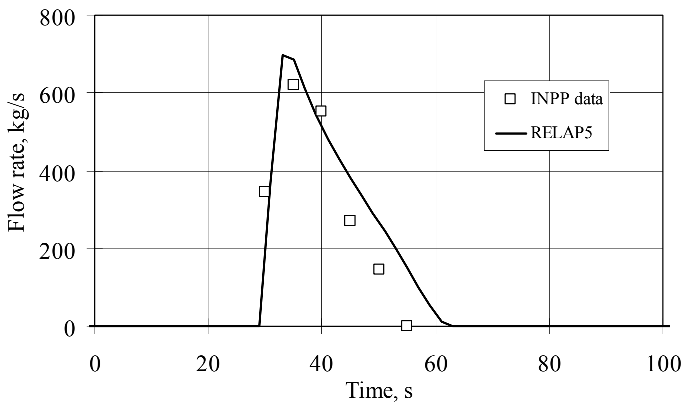
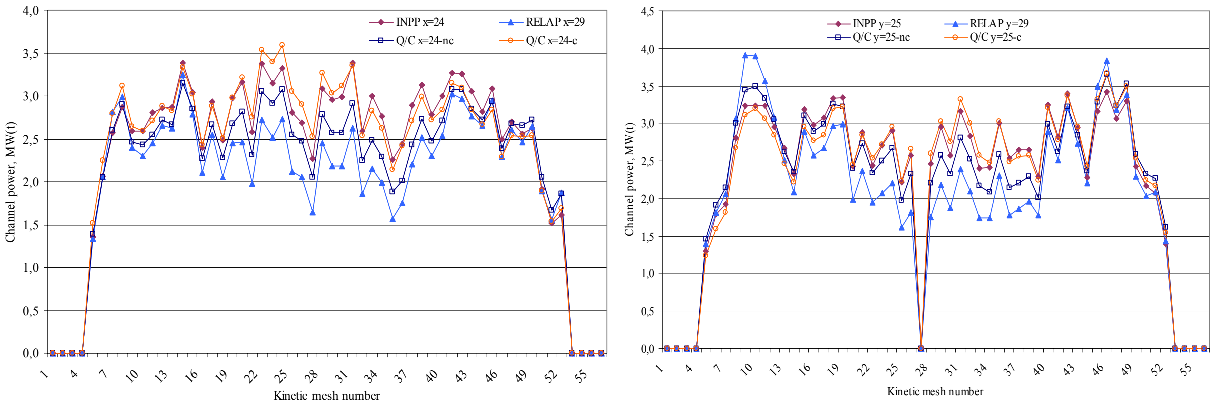


Figure 3. Inadvertent actuation of ECCS event. Mass flow from the ECCS accumulators

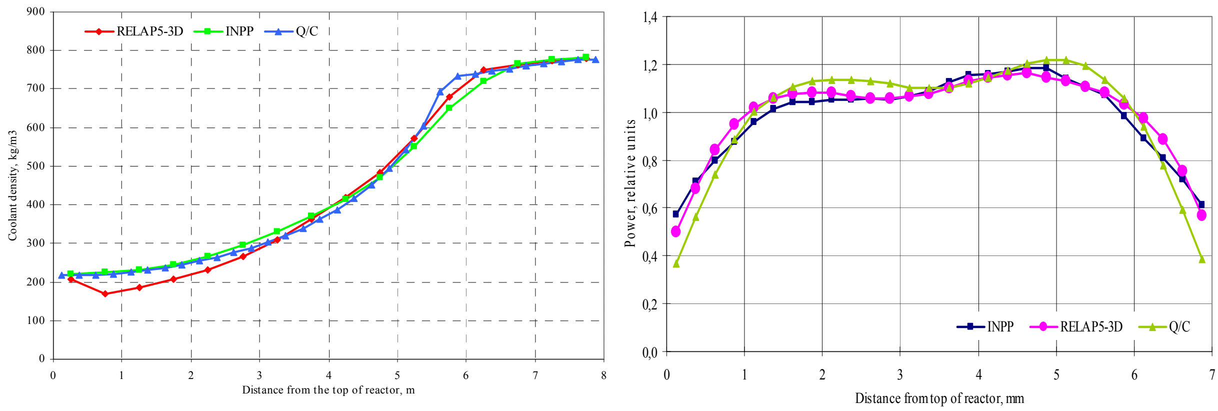
Steady-state calculations of the Ignalina NPP RBMK-1500 reactor (Unit 2, reactor core state for November 26, 1998) core state were made and calculation results obtained for comparison with the real plant data and the calculation results of the same reactor core state obtained using German neutron-dynamic code QUABOX/CUBBOX. Parameters that were compared are: radial and axial power distributions, eigenvalue and coolant density profile in fuel channels in the core region.

Fig. 4 shows the comparison of the radial power distribution as calculated by codes RELAP5-3D and QUABOX/CUBBOX with the real plant data. In these pictures compared are RELAP5-3D code calculation results (indexes: RELAP x=29 and RELAP y=29 – reactor core coordinates used by RELAP5-3D code), QUABOX/CUBBOX code calculation results, obtained using plant data without any fuel assembly burnup corrections (indexes: Q/C x=24-nc and Q/C y=25-nc – reactor core coordinates used by QUABOX/CUBBOX code), QUABOX/CUBBOX code calculation results, obtained using special fuel assembly burnup correction by 5% to fit the power distribution profile in the core measured by in-core detectors (indexes: Q/C x=24-c and Q/C y=25-c) and the real plant data (indexes: INPP x=24 and INPP y=25 – reactor core coordinates used at Ignalina NPP). The compared radial power distributions in fuel channels in X and Y directions are located on the center line of the reactor core in X and Y directions, correspondingly. As one can see radial power distribution calculated by RELAP5-3D code is very similar to the one calculated by QUABOX/CUBBOX code, obtained using plant data without any fuel assembly burnup corrections, although RELAP5-3D code gives slightly lower power values. But both these radial power distributions are still lower than the radial power distribution values obtained from the database.



a) b)

Figure 4. Comparison of the radial power distribution as calculated by codes RELAP5-3D and QUABOX/CUBBOX with the real plant data in X (a) and Y (b) directions in the center of the core



a) b)

Figure 5. Comparison of the axial coolant density (a) and power (b) distribution profiles as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data

In Fig. 4(b) one can notice also two places at the periphery of the core, where RELAP5-3D and QUABOX/CUBBOX values of radial power distribution are higher than the values obtained from the database. Still the biggest deviations from the plant data are in the reactor core center, but going to the periphery of the core the calculated power values agree quite well with the real plant data, except for several points. Now lets compare the radial power distribution values, obtained by QUABOX/CUBBOX code using special fuel assembly burnup correction by 5% to fit the power distribution profile in the core measured by in-core detectors with the plant data and the calculation results, obtained by RELAP5-3D and QUABOX/CUBBOX codes, using plant data without any fuel assembly burnup corrections. Calculation results, obtained by QUABOX/CUBBOX code using fuel assembly burnup correction fits reasonably the plant data obtained from the database, and are closer to the reality than the calculation results, obtained by RELAP5-3D and QUABOX/CUBBOX codes using plant data without any fuel assembly burnup corrections. Here one can see the difference in the calculation results which is due to the availability or unavailability of the fuel assembly burnup correction procedure, used during calculations.

Fig. 5(a) shows the comparison of the coolant density axial distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data. As one can see, the calculation results and plant data are in reasonable agreement. Some disagreement with the plant data could be seen in QUABOX/CUBBOX results near the bottom of the core and in RELAP5-3D results near the top of the core, but the differences are rather small. Fig. 5(b) shows the comparison of the axial power distribution profile as calculated by RELAP5-3D and QUABOX/CUBBOX codes with the real plant data. The calculated axial power distribution profile and the measured one at the Ignalina NPP are in reasonable agreement. Slightly bigger difference is noticed in QUABOX/CUBBOX calculation results, where two power peaks could be seen, one located ~2 m from the top of the core, and another located ~5 m from the top of the core. These are the places where a center of the two fuel bundles in a single fuel assembly are located. On the whole, the axial power non-uniformity value is acceptable being ~1.22. Eigenvalue obtained by RELAP5-3D code is ~1.0013, while eigenvalue obtained by QUABOX/CUBBOX code is ~0.997.

The calculation results of RBMK-1500 reactor core state obtained using RELAP5-3D code are in reasonable agreement with the real plant data and the RELAP5-3D nodal kinetics model represents the Ignalina NPP Unit 2 reactor power and coolant density profiles reasonably well. Eigenvalue close to unity indicates reasonable values are calculated for neutron fluxes.

3. FEEDWATER FLOWRATE PERTURBATION BENCHMARK

Dynamic calculations to repeat the experimental results for void reactivity coefficient measuring were performed for Unit 2 (on November 26, 1998) core conditions. During this experiment feedwater flowrate increases by 200 t/h. It inserts the negative reactivity into the reactor core. This reactivity is compensated by 4 automatic control rods located in cells (32-33; 16-33; 16-17; 32-17). Each automatic control rod operates according to a signal, coming from one lateral ionization chamber, located in annular water tank around the reactor core, serving as a biological protection shield. Positions of all other control rods are not changed during this experiment.

Increasing of feedwater flowrate by 200 t/h causes decreasing of coolant temperature at the inlet of the reactor core by 1 ^{0}C. Such coolant temperature change was modeled in the experiment dynamic calculation. Initial automatic regulator positions in the reactor core condition files were corrected for the calculation according to the reactor core condition before the experiment and comes to 300 cm for each automatic regulator.

Void reactivity coefficient measuring is one of the regular procedures, used at a nuclear power plant of RBMK type. During this measurement, feedwater flowrate changes. It causes reactivity perturbation. Automatic rods change their insertion depths to compensate this reactivity change. The experimental procedure for void reactivity coefficient measuring consists of two stages. At the first stage the perturbation of feedwater flowrate is performed and reactivity change due to this perturbation is compensated by automatic control rods movement. At the second stage the worth of these control rods is being calculated.

The following scenario of the experiment was taken as the basis for the modeling: at the first stage of the void reactivity coefficient measurement, feedwater flowrate was increased by ~(205÷210) t/h per reactor core side to decrease the void fraction in the reactor core. This led to the reactor core neutron field distortion. Four ionization chambers located in lateral water tank (No. 3, 9, 15, 21) measured neutron field change. Four automatic control rods were changing their positions to compensate the reactivity change. Table 2 presents the initial/final calculated and measured automatic control rod positions. In parenthesis differences between experimental (measured) and RELAP5-3D final calculation results are presented as well.

Table 2. Initial/final calculated and measured automatic control rod positions

ΔGfw , t/h 210/ 205	Location of automatic control rods, cm.				
	16-33	32-33	16-17	32-17	Av. value
Initial	300	300	300	300	300
Calculation (Final)	285(5)	226(64)	280(10)	274(6)	266.25 (21.25)
Measurement (In./Fin.)	300/290	300/290	300/290	300/280	300/ 287.5

According to the RELAP5-3D calculation results, automatic regulators average shift is more than obtained during the experiment (33.75 cm and 12.5 cm respectively). The difference can be explained by the fact that the reactor core condition files were obtained not quite before the start of the experiment. It seems, that during this time the core condition was changed, i.e. automatic regulators locations were changed, etc.. To obtain better coincidence of the calculation and experimental results it is needed to get the reactor core condition just before the start of the experiment.

Since the real transient data is not available for comparison of each parameter presented below, the calculation results obtained using RELAP5-3D code were compared only with the calculation results obtained by RRC "KI" using STEPAN code. The STEPAN calculations were performed by RRC "KI" staff in Russia. The STEPAN code is used for everyday neutronic calculations at Ignalina NPP. All passport neutron kinetic characteristics of the reactor are calculated using the STEPAN code.

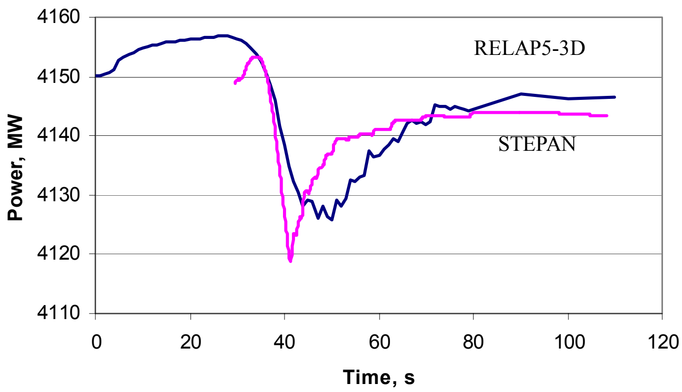


Figure 6. Total reactor core power versus time. Feedwater flow perturbation experiment

According to RELAP5-3D and STEPAN calculation results (see Fig. 6), the increase of feedwater flowrate by ~200 t/h leads to the total reactor core power decrease from the initial power of

~4150 MW(th) to ~4120 MW(th). Initial power increasing by ~5 MW was obtained by RELAP5-3D and STEPAN codes.

This can be explained by the fact, that from the very beginning of the coolant temperature decrease, pressure in the primary circuit also decreases. Pressure decrease goes with the speed of sound, while the front of the colder coolant, traveling through the primary circuit, reaches the core region only after a certain period of time. This initial pressure decrease leads to coolant density decrease in the core region, which results in the reactivity increase by ~5 MW. Afterwards, after a certain period of time, reactivity starts to decrease, because during this time the front of colder coolant already enters the core region and coolant density increases again.

According to RELAP5-3D and STEPAN calculation results (see Fig. 7), the average signal deviation of four lateral fission chambers from the set-point value at the start of the transient is equal to zero. After the stabilization of the transient, this signal deviation decreases to ~(-0.01). This corresponds to the reality, because automatic regulation rods start to move up/down when signal deviation from the set-point value reaches –1%/+1%.

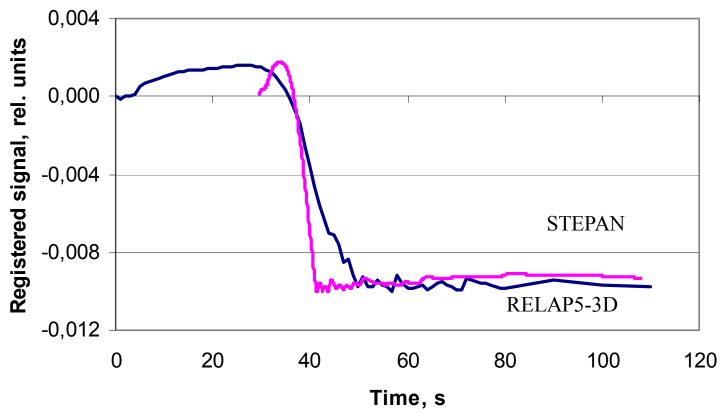


Figure 7. Summary signal of four lateral fission chambers. Feedwater flow perturbation experiment.

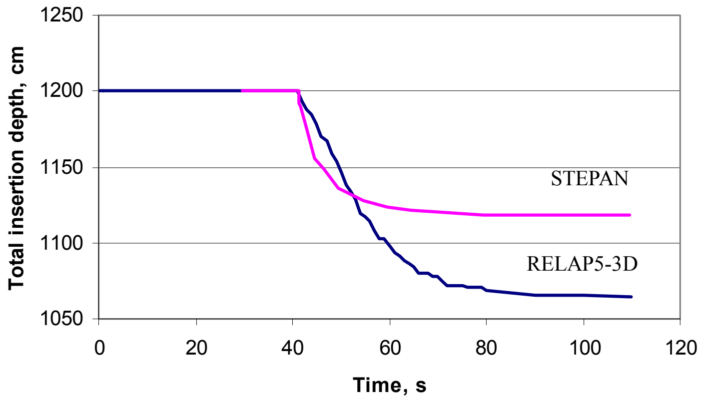


Figure 8. Total insertion depth of four automatic regulators. Feedwater flow perturbation experiment.

Both codes overestimate the total insertion depth of four automatic regulators during the transient (see Fig. 8). RELAP5-3D code overestimates this value by 85 cm, while STEPAN code overestimates it by 32 cm.

STEPAN code models this feedwater flow perturbation transient by changing coolant temperature simply by 1 ^{o}C at the core inlet. This corresponds to the increasing of feedwater rate by ~200 t/h. In RELAP5-3D code feedwater flow perturbation transient is modeled directly by changing feedwater flowrate by ~200 t/h. This also causes coolant temperature decrease by 1 ^{o}C at the core inlet. So during the modeling of this transient using RELAP5-3D code, there exists some delay between the start of feedwater flowrate change and the decrease of coolant temperature at the core inlet. That's why there can be observed the time difference between the moments of the decrease of the total reactor core power in RELAP5-3D and STEPAN cases. Zero time point in the presentation of STEPAN results is shifted to the right to make the comparison of both calculation results easier and more evident.

In general, RELAP5-3D and STEPAN codes give reasonable mutual coincidence of the calculation results and their reasonable agreement with real plant data.

4. REACTOR POWER REDUCTION BENCHMARK

The reactor power reduction transient is initiated by the reactor shutdown operation, where the parameters of the reactor are strongly changed. Besides evident decrease of reactor power, such parameters as system pressure and MCP flow are changed significantly because of the total reactor power reduction and also under the influence of the equipment functioning.

Reactor shutdown is a regular reactor operation procedure during the entire lifetime of the reactor. Usually it starts by scram signal activation by the operator. After this signal all 24 fast acting scram rods are fully inserted into the reactor core during ~6 s from the top end switch position, while all the rest control rods are fully inserted into the reactor core during ~13 s from their present actual operation positions. The above described control rods insertion causes sharp decrease of reactivity and the total reactor core power. Usually, during this transient in-core and lateral detectors measure neutron field distribution in the reactor core. Measurement results are registered by the reactor information computer system. On March 29, 1999 Ignalina NPP Unit 2 was shutdown by the operator signal. Initial reactor core conditions and neutron flux behavior were registered by in-core detectors.

The real operation conditions of Ignalina NPP Unit 2 (on March 29, 1999) were used for this benchmark. Reactor power was equal to 2065 MW(th) (as taken from the database), but just before the reactor shutdown it was decreased to 1204 MW(th). This decreased reactor core power was taken as the basis for the benchmark. Reactor core loading on this date consisted of 1251 FA with uranium-erbium fuel, 408 FA with 2.0% enr. uranium dioxide fuel, 1 water column and had 71 manual control rods of 2477-01 design (skirt type).

The reactor scram calculation was performed in two steps:
- 24 fast acting scram rods were inserted in the first step;
- all the rest control rods were inserted in the second step 13 s later.

Calculation modeling of the scram signal was performed according to the above described scenario. 24 fast acting scram rods were inserted into the reactor core from the top end switch beginning from the 48 s. 13 s later all the rest control rods were inserted into the reactor core simultaneously from their present actual operation positions. Insertion velocity of the fast acting scram rods was assumed to be 120 cm/s, while the insertion velocity of all the rest control rods, except for the short-bottom control rods, was assumed to be 80 cm/s. Short-bottom control rods were assumed to be inserted with the velocity of 40 cm/s.

Calculation results using RELAP5-3D code are presented for 220 s time period, including 48 s calculation of zero transient before the reactor scram signal initiation, to match the presented real plant data. Like in previous benchmarks water flowrate at the reactor core inlet and the steam drum pressure were used as boundary conditions for the calculation.

According to RELAP5-3D and STEPAN calculation results, the insertion of 24 fast acting scram rods beginning from 48 s of the reactor shutdown transient causes sharp total reactor core power decreasing. Fig. 9 shows, that, during this first reactor shutdown phase, in STEPAN case the total reactor core power decreases more sharply than in RELAP5-3D case. This can be explained by the fact, that STEPAN code overestimates the efficiency of 24 fast acting scram rods in comparison with RELAP5-3D code. From the beginning of the second phase of reactor shutdown (from 61 s), when AZ-1 signal is initiated and all the remaining control rods start to insert into the core from their actual operation positions, STEPAN and RELAP5-3D calculation results show exactly the same behavior of the total reactor core power versus time. So in general, STEPAN and RELAP5-3D codes calculate total reactor core power behavior in time during the reactor shutdown transient in quite a similar manner and the calculation results correspond quite reasonably to each other.

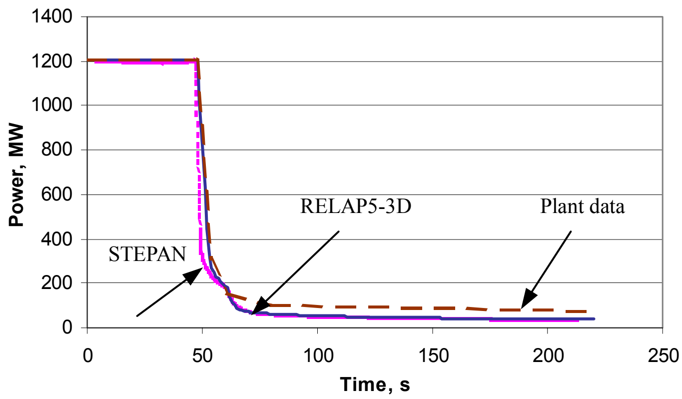


Figure 9. The total reactor core power behavior versus time during the reactor shutdown transient.

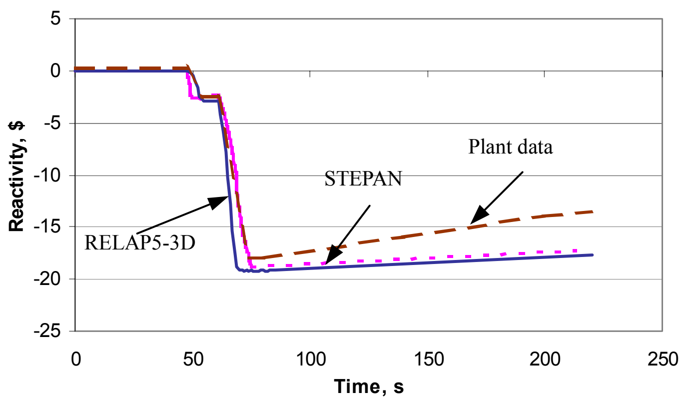


Figure 10. Calculation “counter” reactivity behavior during the reactor shutdown transient.

Fig. 10 shows the comparison of the “counter” reactivity as calculated by STEPAN and RELAP5-3D codes. Experimental results are presented in the figure as well. As in previous figure one can see, that during the first phase of reactor shutdown process, STEPAN code overestimates the efficiency of 24 fast acting scram rods in comparison with RELAP5-3D code. As a result of this, the calculated "counter" reactivity by STEPAN code decreases more sharply (beginning from 48 s), than the reactivity calculated by RELAP5-3D code. In the second phase of reactor shutdown, when AZ-1 signal is initiated and all the remaining control rods start to insert into the core from their actual operation positions, RELAP5-3D code overestimates the efficiency of CPS rods without 24 fast acting scram rods in comparison with STEPAN code. As a result of this, the "counter" reactivity calculated by RELAP5-3D code decreases more sharply (beginning from 61 s), than the reactivity calculated by STEPAN code. At the final reactor shutdown point (beginning from ~75 s), both codes calculate the total reactor shutdown effect value to be approximately the same.

In STEPAN case, the total reactor shutdown effect is equal to ~19.0β, while in RELAP5-3D case, the total reactor shutdown effect is equal to ~19.1β. The worth of 24 fast acting scram rods is equal to ~2.6β and ~2.8β, respectively. In general, the "counter" reactivity behavior in time, calculated by STEPAN and RELAP5-3D codes, is very similar and the final reactor shutdown effectiveness values correspond quite reasonably to each other.

In general, RELAP5-3D and STEPAN codes give reasonable mutual coincidence of the calculation results and their reasonable agreement with real plant data.

CONCLUSIONS

A successful best estimate RELAP5-3D model of the Ignalina NPP RBMK-1500 reactor has been developed and validated against real plant data. The validation of the model has been performed using operational transients from the Ignalina NPP. The benchmark problem analyses, that were performed during the validation of the successful best estimate RELAP5-3D model of the Ignalina NPP RBMK-1500 reactor and reported here are: feedwater flow perturbation and reactor power reduction transients. Both benchmarks were modeled using the RELAP5-3D code and the calculation results compared to the calculation results obtained using the STEPAN code, as well as to the real plant data registered by the TITAN information computer system at Ignalina NPP, where such data was available for comparison. Comparison of the results obtained, using the RELAP5-3D and STEPAN (specially designed for RBMK reactor analysis) codes, showed reasonable mutual coincidence of the calculation results and their reasonable agreement with real plant data.

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NOMENCLATURE

ACS	Accident Confinement System
ANL	Argonne National Laboratory
AZ-1,3,4,6	Emergency Protections 1, 3, 4, 6
CPS	Control and Protection System
DBA	Design Basis Accident
DKER	Russian Acronym for "Power Density (Distribution) Monitoring Sensor Radial
DS	Drum Separator
ECCS	Emergency Core Cooling System
FA	Fuel Assembly
FC	Fuel Channel
GDH	Group Distribution Header
ICS	Information Computer System
INEEL	Idaho National Engineering and Environmental Laboratory
INPP	Ignalina Nuclear Power Plant
INSP	International Nuclear Safety Program
LAR	Russian Acronym for "Local Automatic Control"
LEP	Local Emergency Protection
LOCA	Loss-of-Coolant-Accident
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
MFWP	Main Feedwater Pump
MSV	Main Safety Valve
NPP	Nuclear Power Plant
RBMK	Large Channel Type Water Cooled Graphite Moderated Reactor
RRC "KI"	Russian Research Center "Kurchatov Institute"
SDV-A	Steam Discharge Valves to ACS
SDV-C	Steam Discharge Valves to Turbine Condensers
SDV-D	Steam Discharge Valves to Deaerators and to In-house Needs
US DOE	Department of Energy of the United States of America