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RELAP5-3D code applications for RBMK-1500 reactor core analysis

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Introduction

RELAP5 code originally was designed for PWR and BWR type 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.

RELAP5-3D code was evaluated for its suitability to model specific transients that take place during RBMK-1500 reactor operation, where the neutronic response of the core is important. Using RELAP5-3D code a successful best estimate RELAP5-3D model of Ignalina NPP RBMK-1500 reactor has been developed and validated against real plant data.

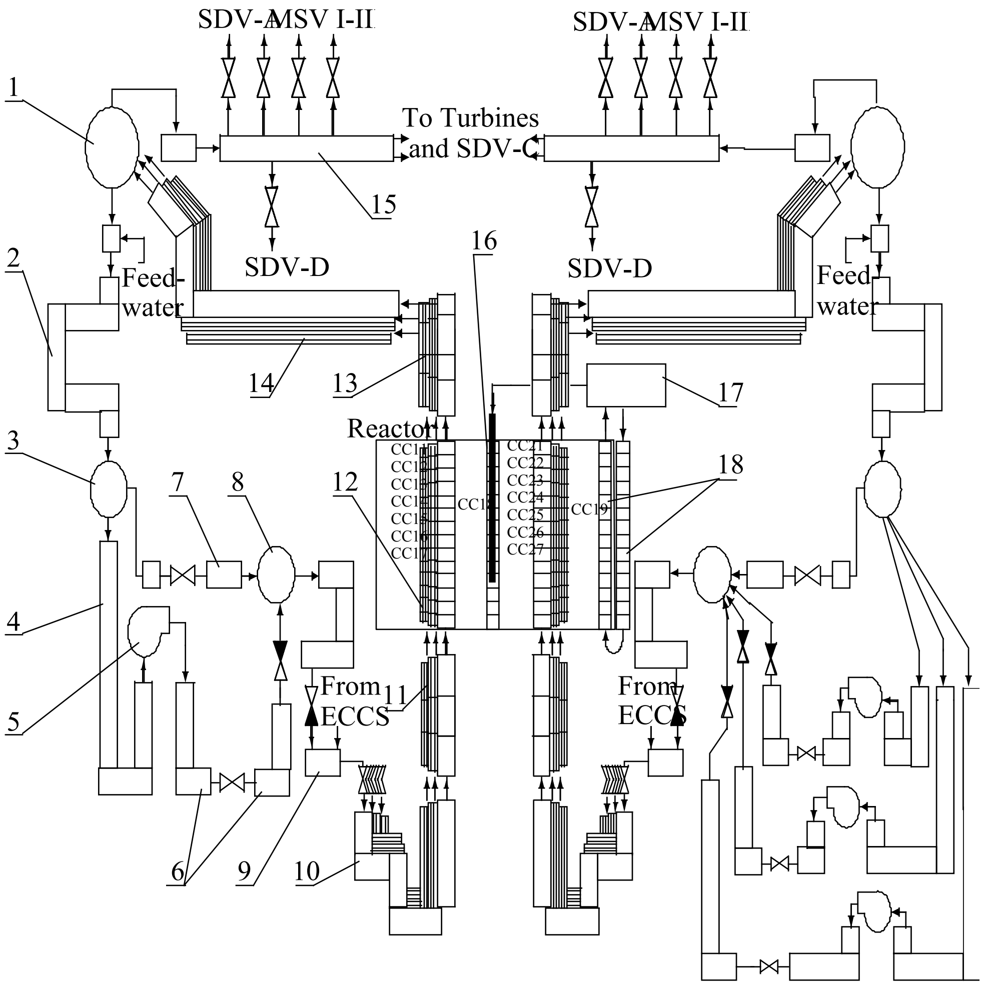
The four benchmark problem analyses, that were performed during the validation of the successful best estimate RELAP5-3D model of the Ignalina NPP RBMK-1500 reactor and will be reported here are: single control rod and group of control rods inadvertent withdrawal, feedwater flow perturbation and reactor power reduction transients. All 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, where such data was available for comparison.

Ignalina NPP RELAP5-3D model (1)

The main purpose for using RELAP5-3D code in this analysis 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.

The RBMK-1500 is graphite moderated, boiling water, multi-channel reactor. The general thermal-hydraulic nodalization scheme of the model is presented on the next slide. 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, while the right half is modeled in finer detail.

Ignalina NPP RELAP5-3D model (2)



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

Ignalina NPP RELAP5-3D model (3)

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 Tables 1 and 2, correspondingly for INPP Unit 2 reactor core states registered on November 26, 1998 and on 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 also.

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.

Ignalina NPP RELAP5-3D model (4)

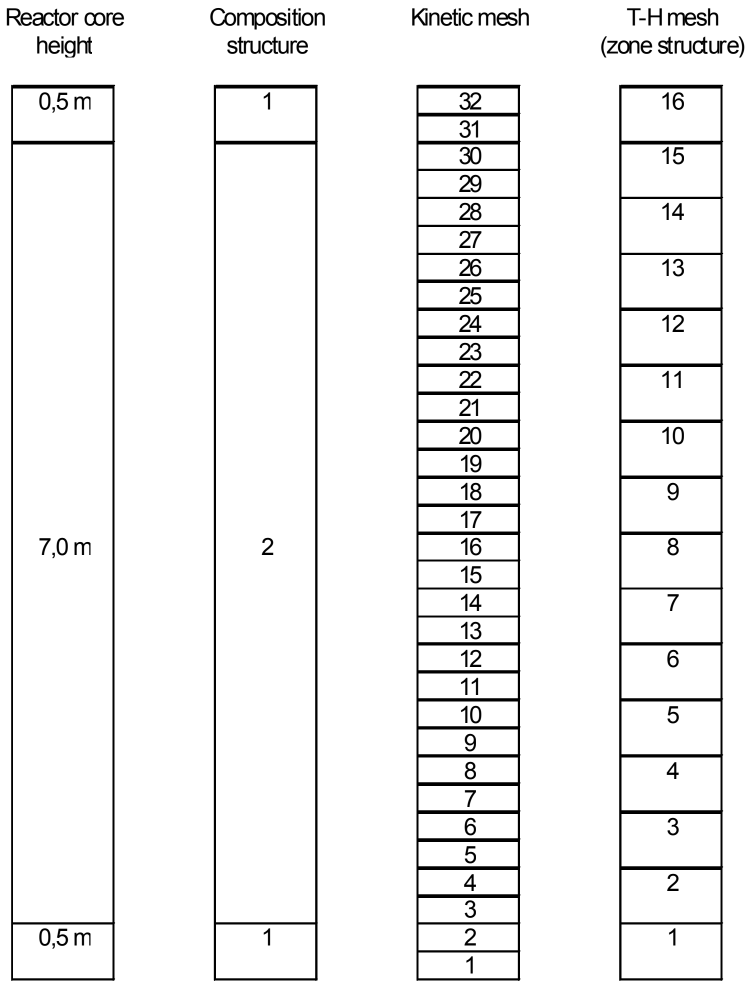
Table 1. Summary specification of the thermal-hydraulic channel groups as being modeled in the RBMK-1500 reactor RELAP5-3D model (Unit 2, November 26, 1998)

Ch. gr. 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
CC21	Right	378	2.95	28.2
CC12	Left	249	2.5	26.2
CC22	Right	234	2.5	26.2
CC13	Left	60	2.4	25.1
CC23	Right	59	2.4	25.1
CC14	Left	59	1.8	21.1
CC24	Right	55	1.8	21.1
CC15	Left	39	1.6	17.5
CC25	Right	37	1.6	17.5
CC16	Left	61	1.2	15.6
CC26	Right	70	1.2	15.6
CC17	Left	3	1.8	33.5
CC27	Right	2	1.8	33.5
CC18		235		
CC19		592*		

Table 2. Summary specification of the thermal-hydraulic channel groups as being modeled in the RBMK-1500 reactor RELAP5-3D model (Unit 2, March 29, 1999)

Ch. gr. specific.	Reactor side	No. of ch.	Av. power in ch., MW	Av. flowrate. in ch., m^3/h
CC11	Left	304	1.5	28.2
CC21	Right	308	1.5	28.2
CC12	Left	301	1.3	25.6
CC22	Right	305	1.3	25.6
CC13	Left	55	1.1	24.6
CC23	Right	51	1.1	24.6
CC14	Left	65	0.8	19.7
CC24	Right	63	0.8	19.7
CC15	Left	38	0.8	16.1
CC25	Right	35	0.8	16.1
CC16	Left	60	0.6	14.3
CC26	Right	71	0.6	14.3
CC17	Left	3	0.8	34.0
CC27	Right	2	0.8	34.0
CC18		235		
CC19		592*		

Ignalina NPP RELAP5-3D model (5)

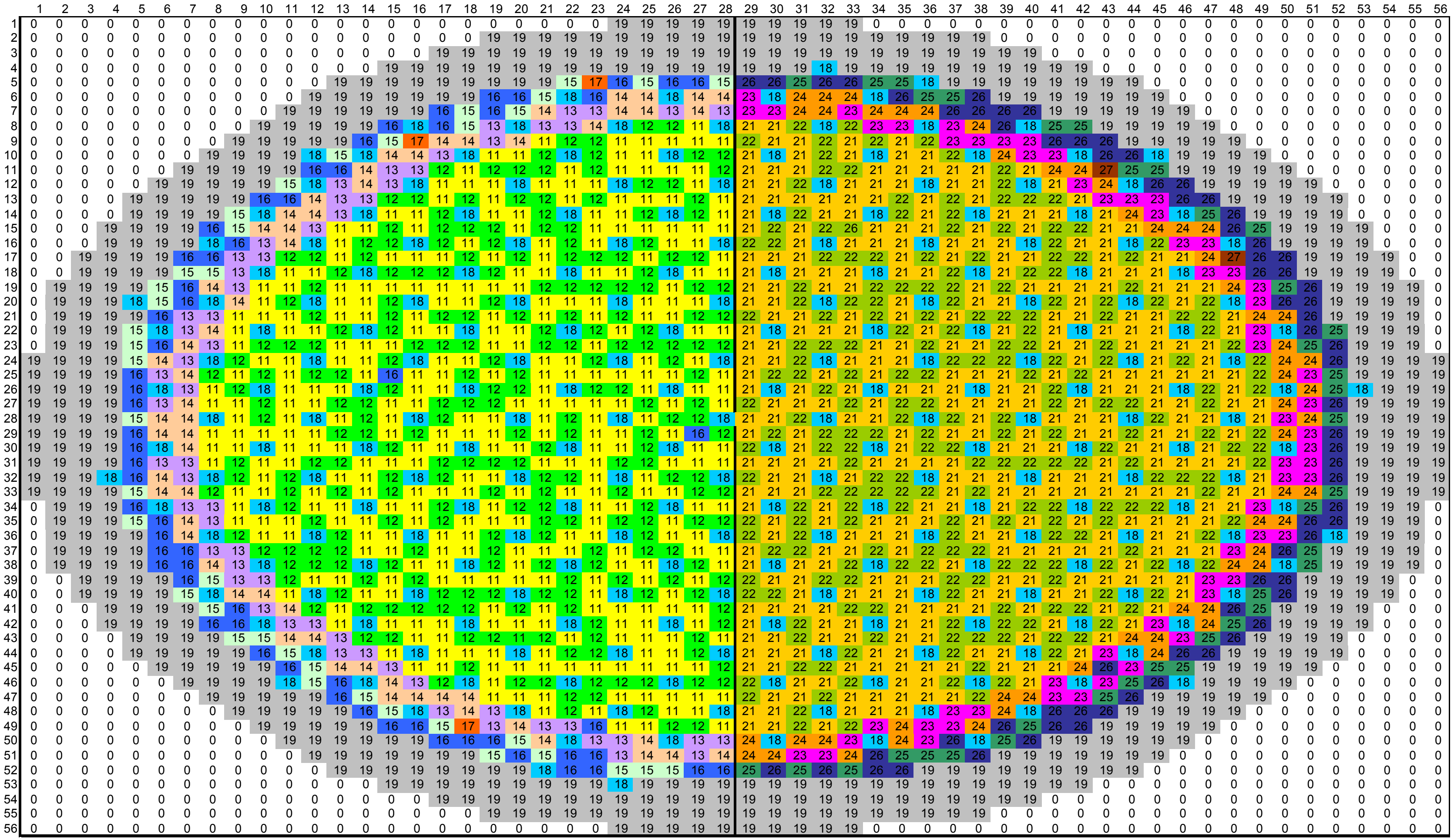


The RBMK-1500 reactor core has a 7.0 m fuel region and a 0.5 m reflector region above and below the fuel region. The overall height of the core region is 8.0 m. The neutronics mesh represents each rectangular graphite column as one individual stack in the radial plane.

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.

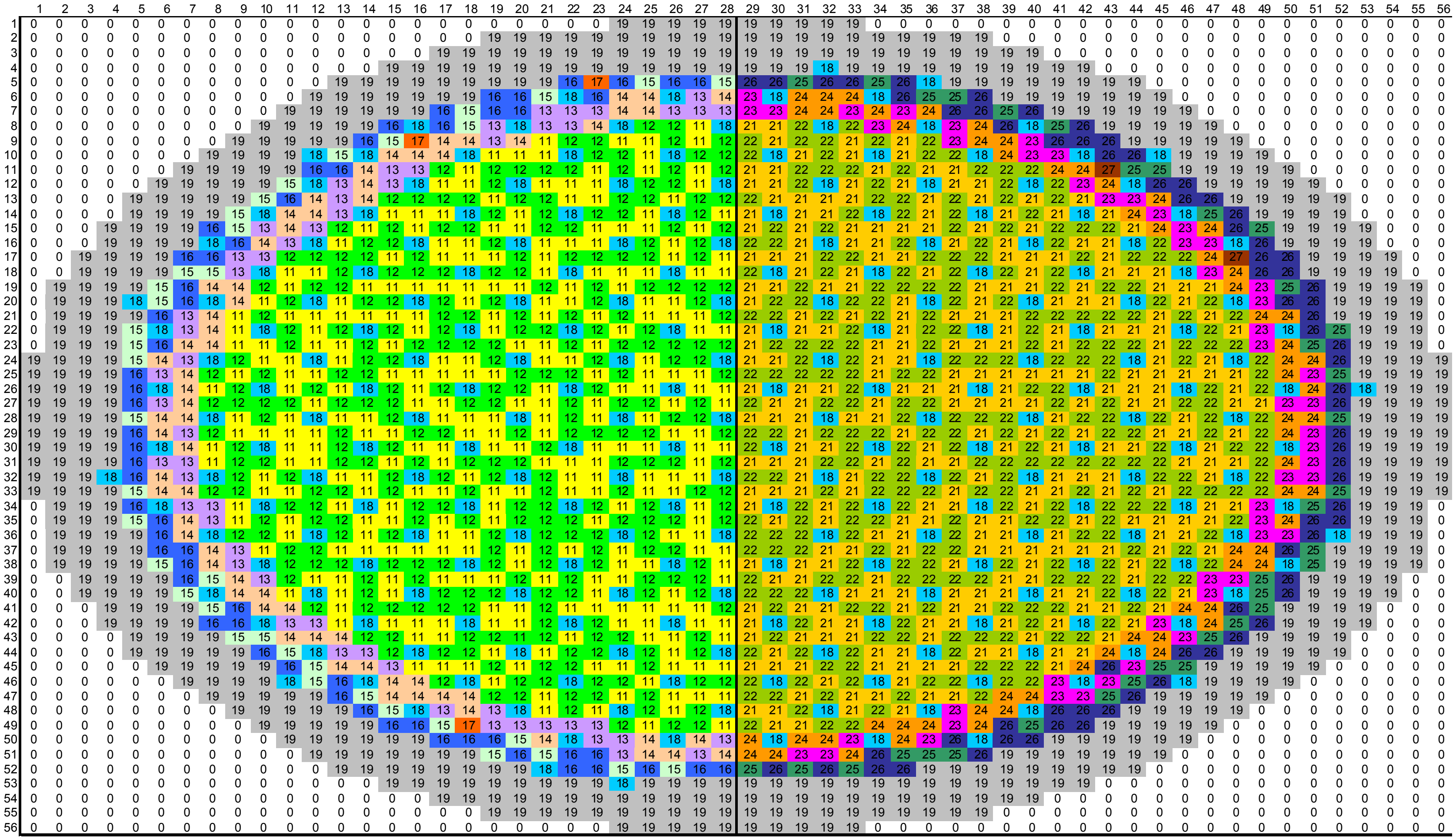
In thermal-hydraulic model of the reactor core we have 16 thermal-hydraulic meshes: 14 nodes (0.5 m each) in the fuel region and 1 node in each of the top and bottom reflector region. In this way the height of the two neutronics nodes are equal to the height of one thermal-hydraulic node.

Ignalina NPP RELAP5-3D model (6)



Nodalization scheme of the RBMK-1500 reactor core (Unit 2, November 26, 1998).

Ignalina NPP RELAP5-3D model (7)



Nodalization scheme of the RBMK-1500 reactor core (Unit 2, March 29, 1999).

Ignalina NPP RELAP5-3D model (8)

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.

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.

Ignalina NPP RELAP5-3D model (9)

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.

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. 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.

Single control rod withdrawal (1)

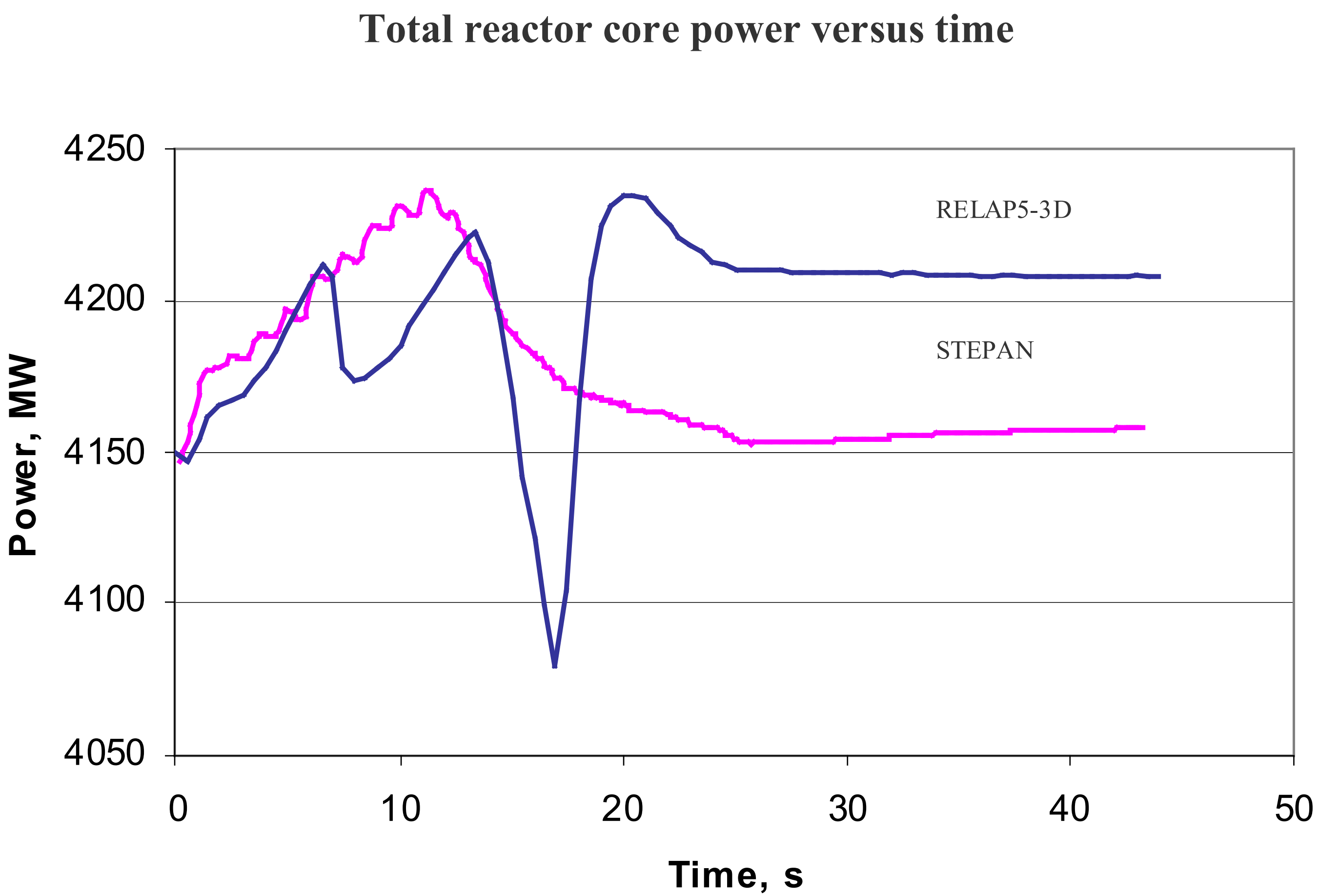
The real operation conditions of Ignalina NPP Unit 2 (on November 26, 1998) were used for this benchmark.

Transient calculation was performed for control rod 10-15 inadvertent withdrawal. This manual control rod of skirt design (mod. 2477-01) was withdrawn from its initial depth of 561 cm (by position indicator) with constant speed 40 cm/s till top end switch.

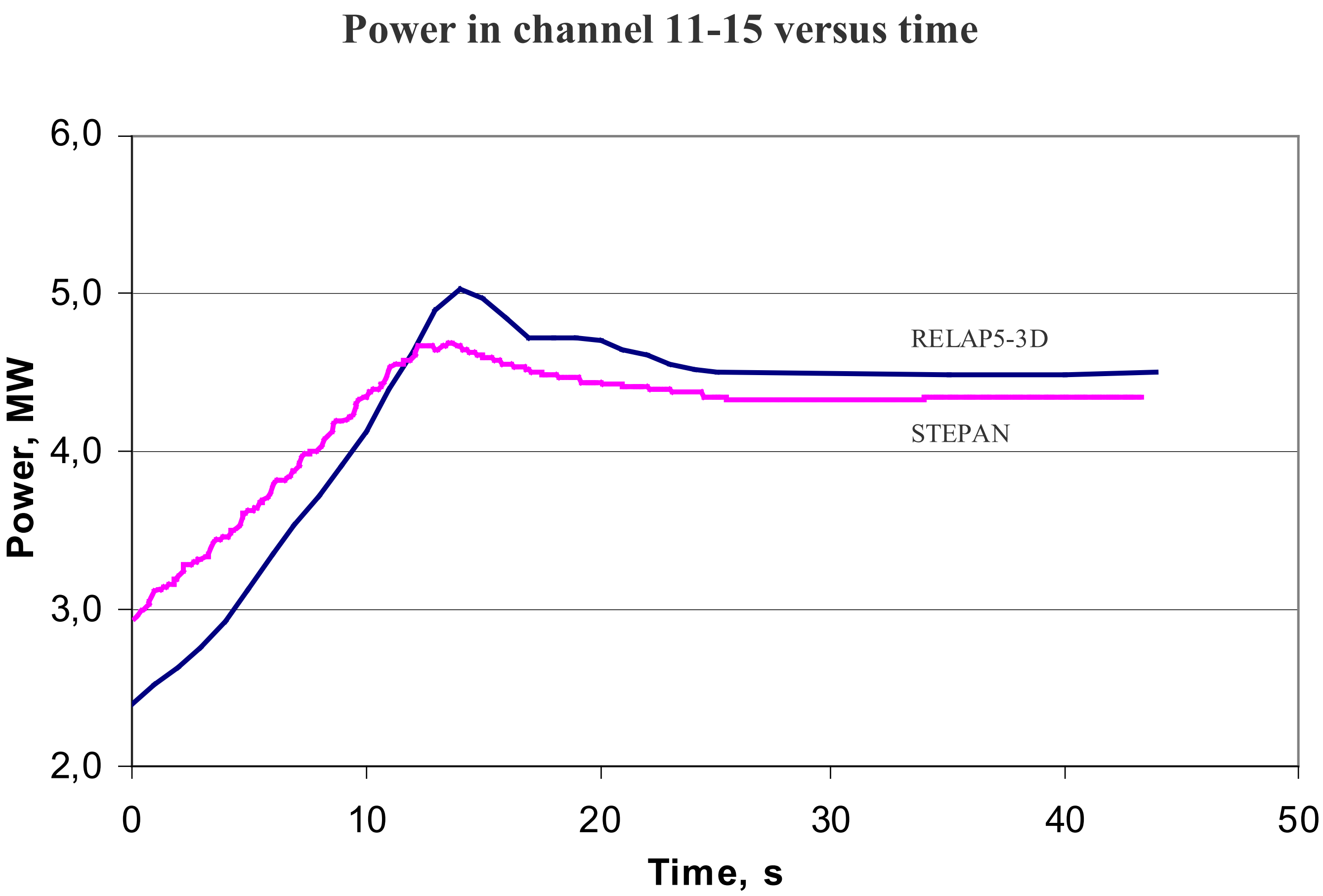
Modeled was control rod 10-15 withdrawal with full scale LAR system operation and without reactor scram activation. In this case single control rod 10-15 inadvertent withdrawal is compensated by full scale LAR system operation. All 12 LAR rods are moving to keep core power distribution steady.

Calculation results show, that LAR system successfully neutralizes perturbation caused by a single control rod inadvertent withdrawal.

Single control rod withdrawal (2)

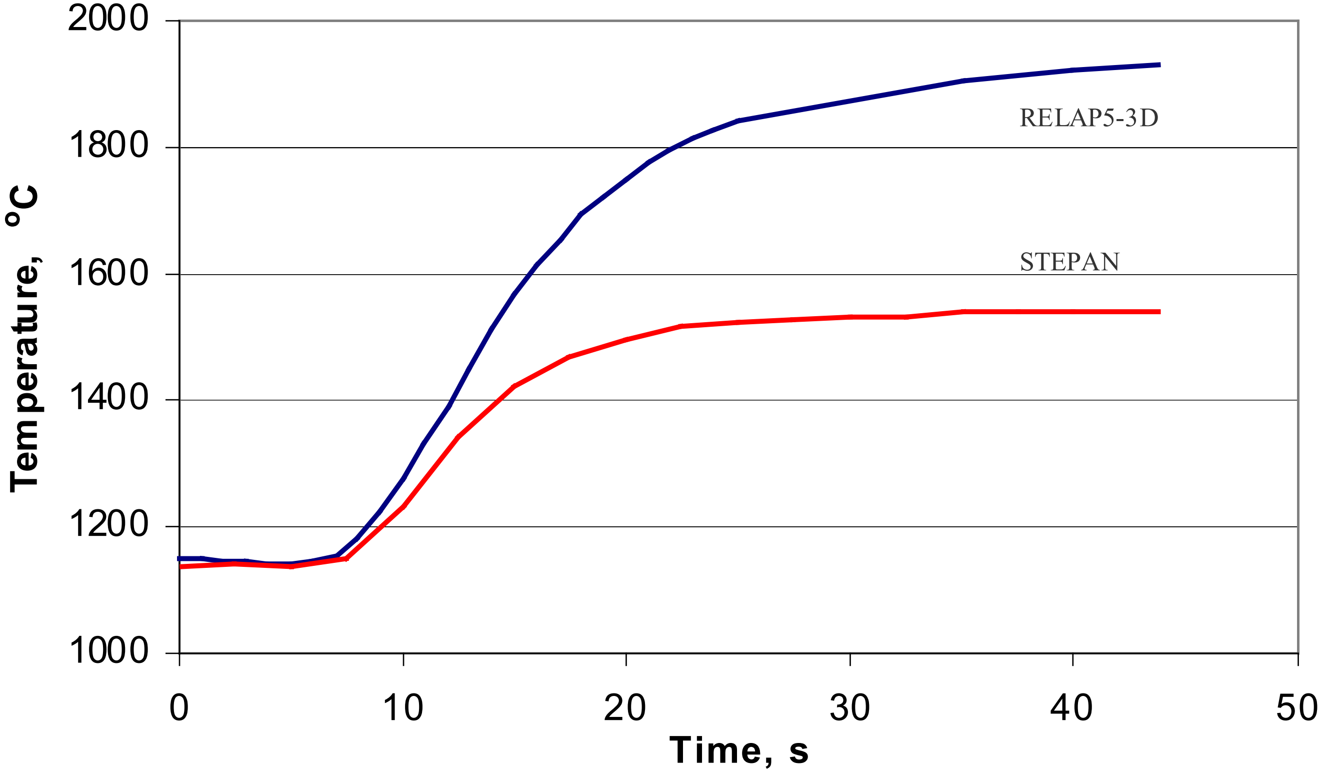


Single control rod withdrawal (3)



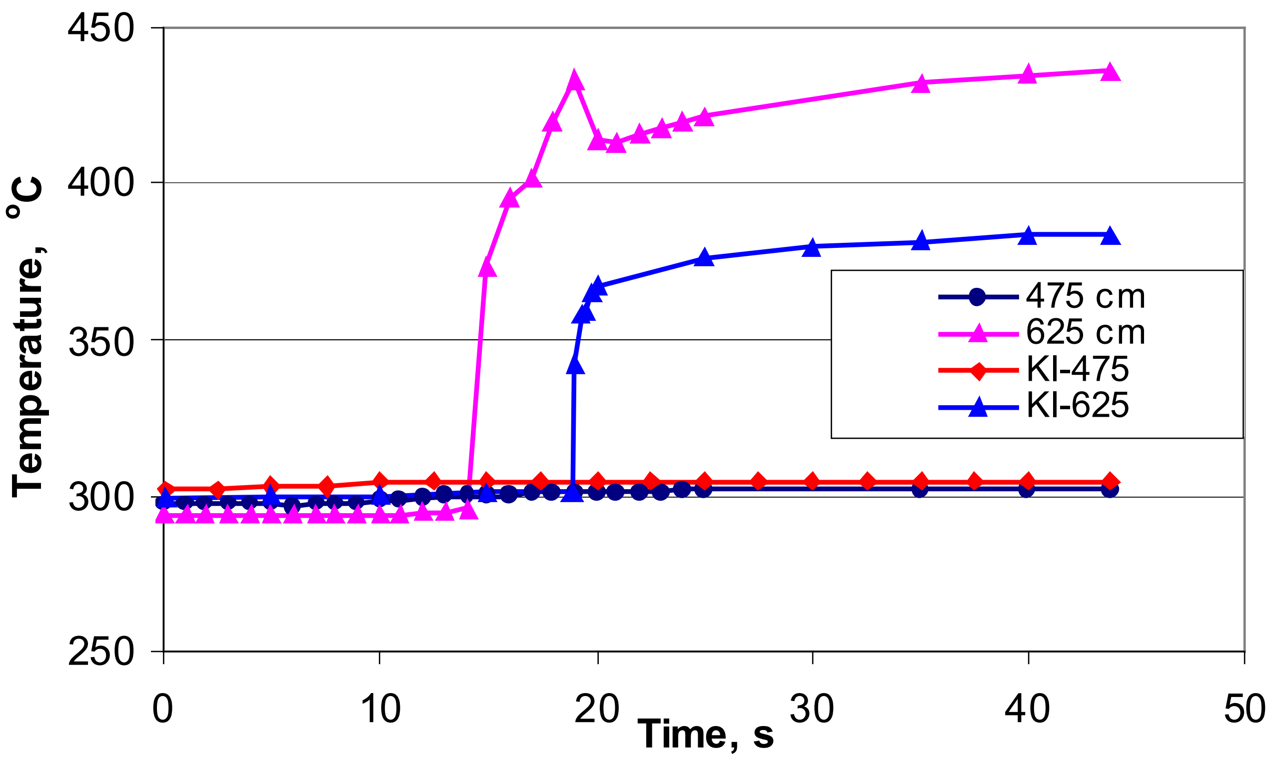
Single control rod withdrawal (4)

Fuel centerline temperature in the channel 11-15 at the level 475 cm from the core bottom versus time



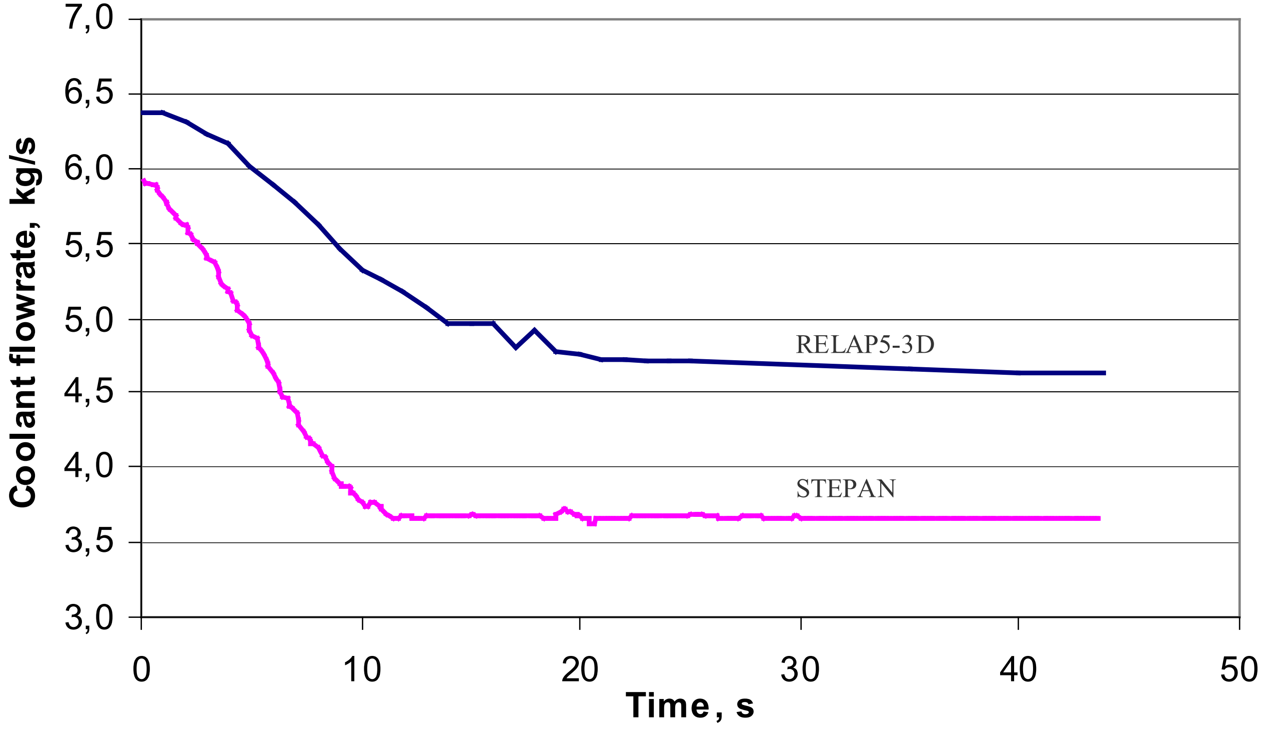
Single control rod withdrawal (5)

Fuel clad temperature in the channel 11-15 at the level 475 and 625 cm from the core bottom versus time

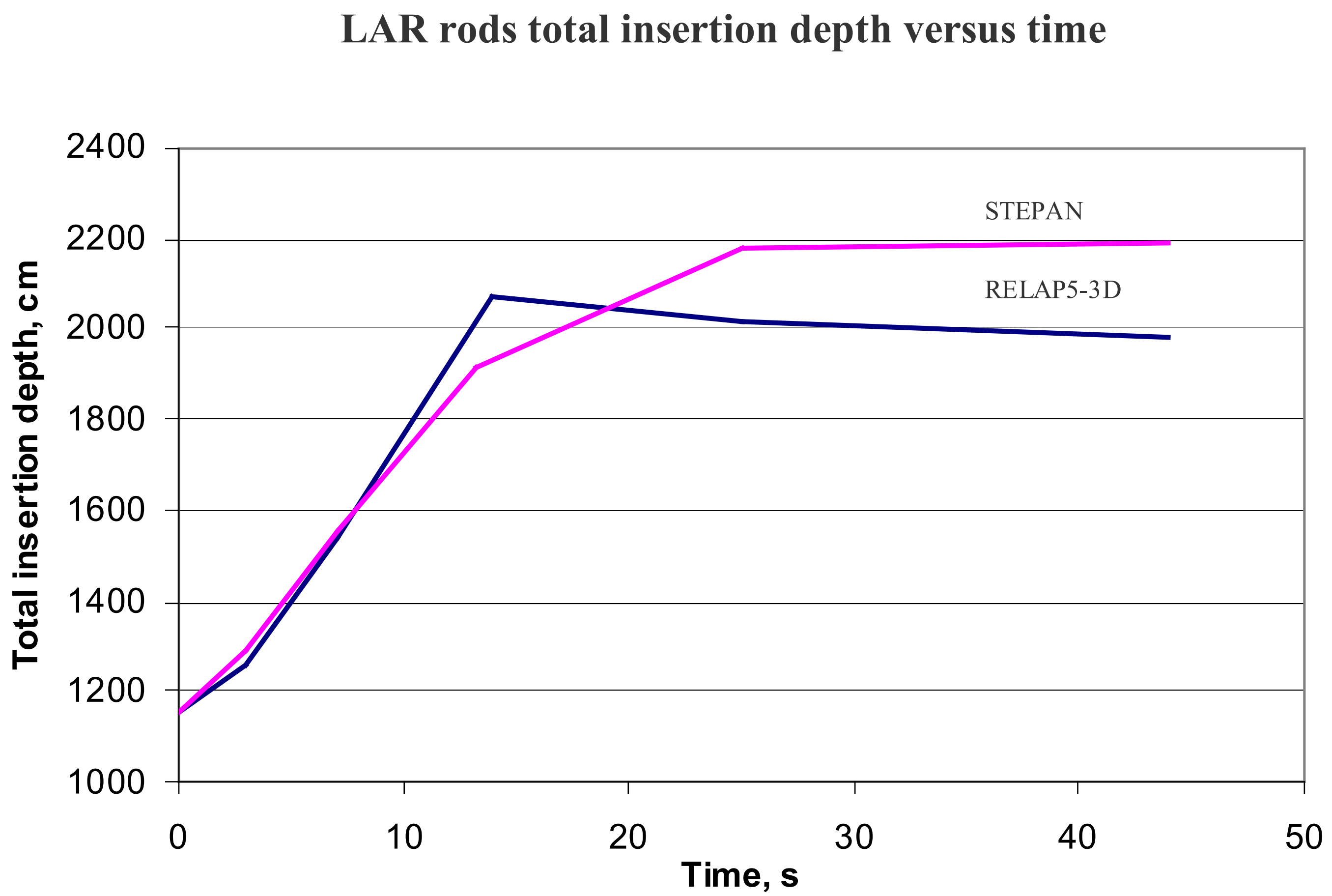


Single control rod withdrawal (6)

Coolant flowrate at the inlet of the fuel channel 11-15 versus time



Single control rod withdrawal (7)



Single control rod withdrawal (8)

In general, RELAP5-3D and STEPAN codes give reasonable agreement of the calculation results for a single control rod 10-15 inadvertent withdrawal out of the core with full scale LAR system operation.

Regarding the agreement of separate parameters behavior in time as calculated by both codes, it could be summarized as follows:

- Reasonable agreement was obtained for: i) total reactor power; ii) power in nearby channel; iii) fuel cladding temperature in nearby channel; iv) LAR rods total insertion depth;
- Minimal agreement was obtained for: i) fuel centerline temperature in nearby channel; ii) coolant flowrate in nearby channel.

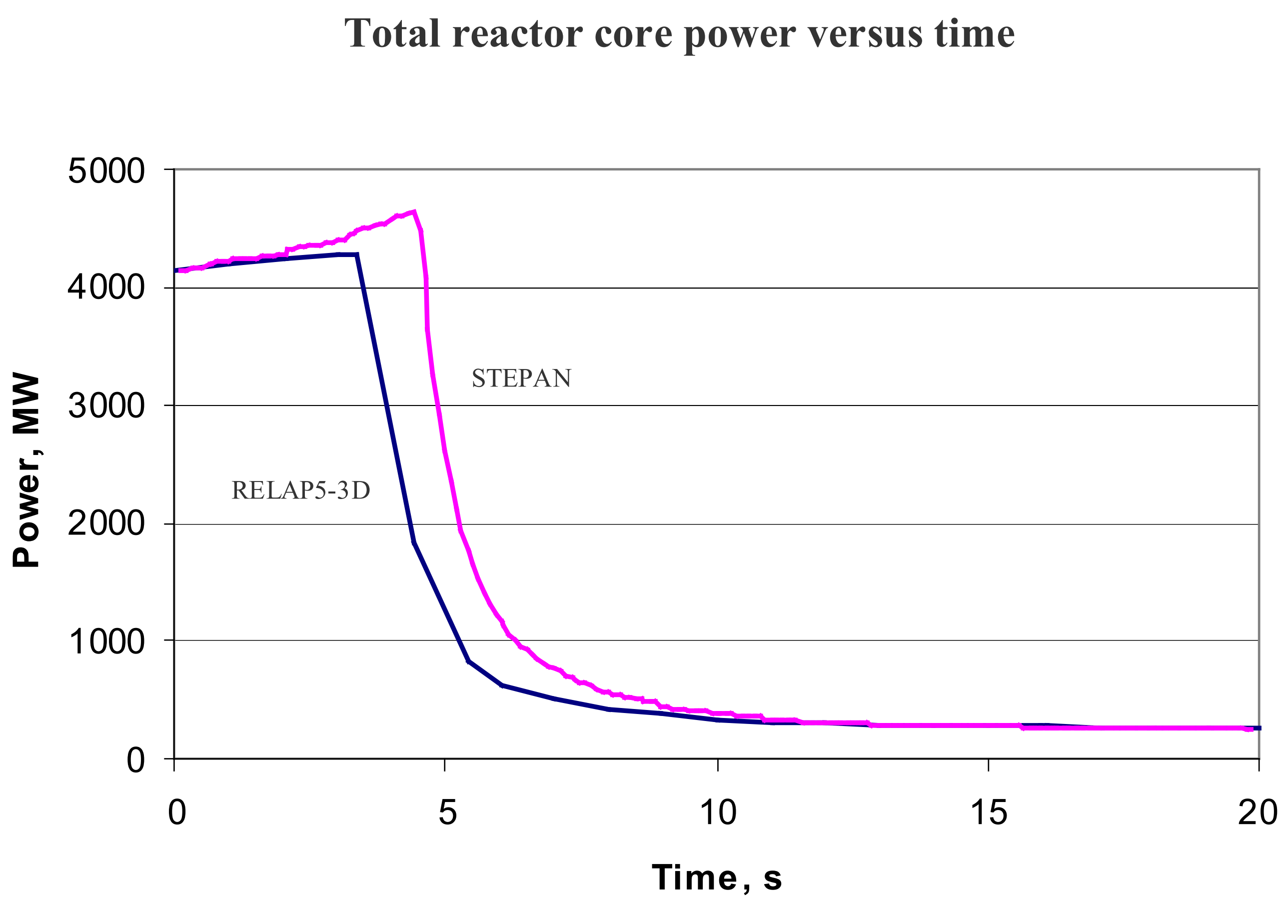
Group of control rods withdrawal (1)

The real operation conditions of Ignalina NPP Unit 2 (on November 26, 1998) were used for this benchmark.

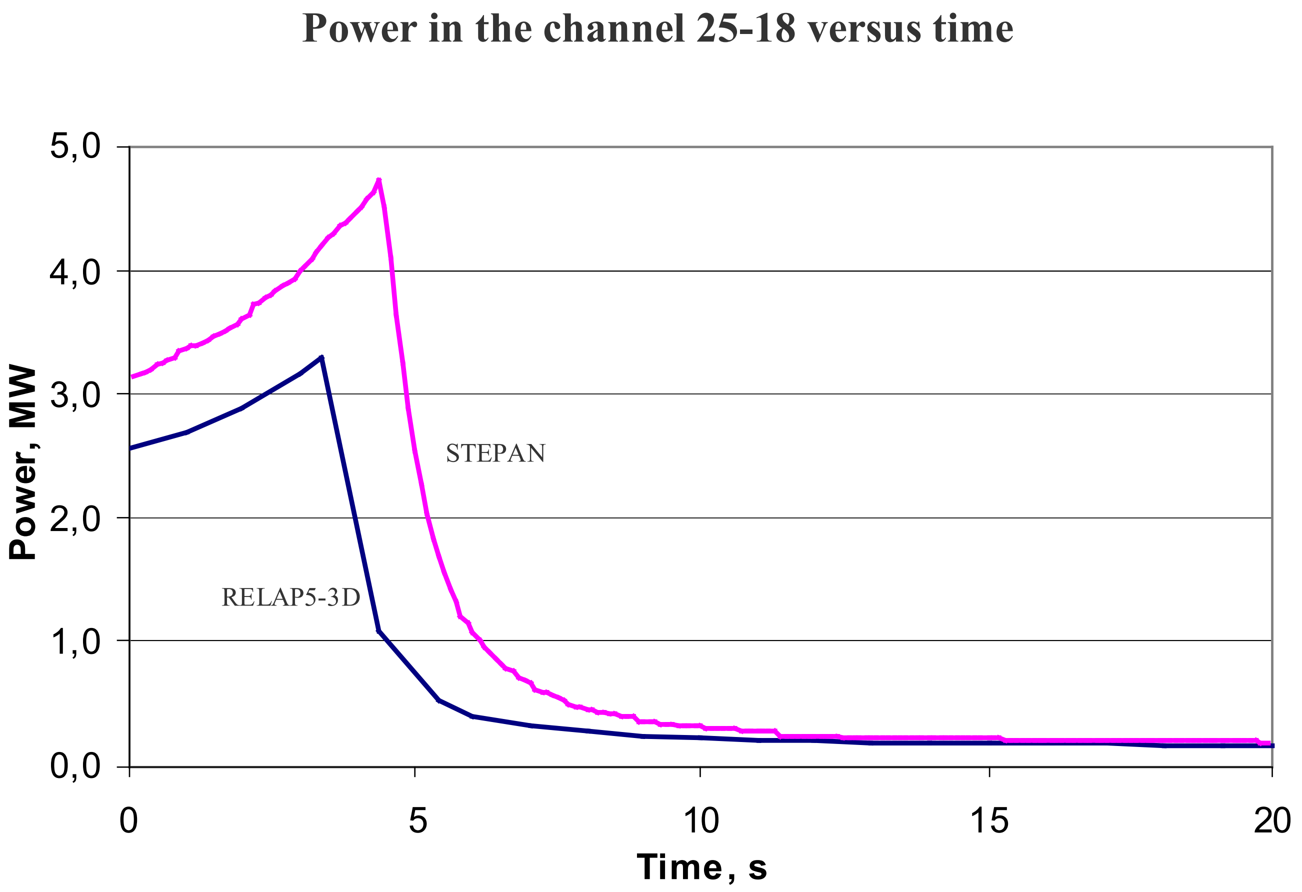
Transient calculation was performed for a group of control rods inadvertent withdrawal. A group of 4 manual control rods were withdrawn from their initial positions (CR 22-19 – 507 cm; CR 22-15 – 643 cm; CR 26-15 – 509 cm; CR 30-15 – 546 cm by position indicator) with constant speed 40 cm/s till top end switch. Reactivity insertion by a group of 4 manual control rods withdrawal is much more than reactivity insertion by one control rod withdrawal. Inadvertent withdrawal of 4 MCR without scram activation leads to unlimited power increase.

Therefore in this case transient was modeled with reactor scram activation and full scale LAR system operation. A group of control rods inadvertent withdrawal is compensated by full scale LAR system operation. All 12 LAR rods are moving to keep core power distribution steady, but this is not enough to suppress the inserted reactivity. Due to this, reactor power increases up to the moment of scram system activation.

Group of control rods withdrawal (2)

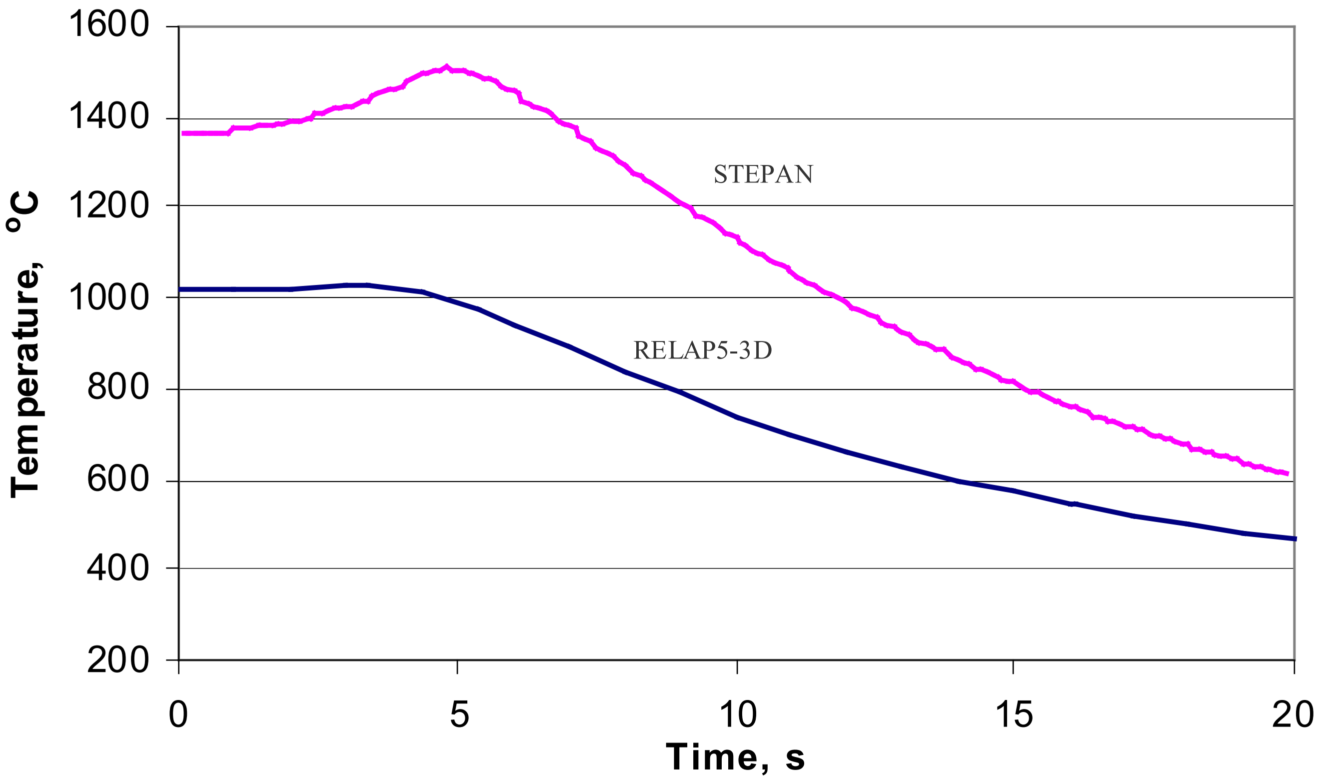


Group of control rods withdrawal (3)



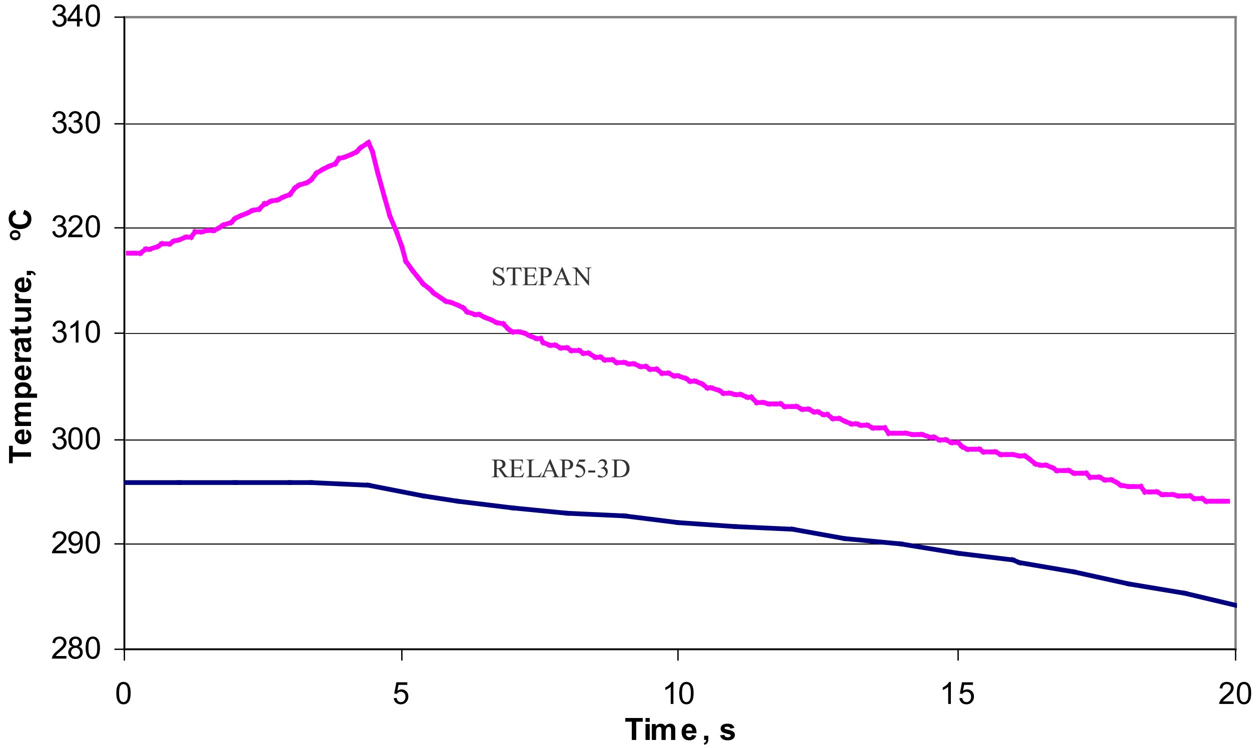
Group of control rods withdrawal (4)

Fuel centerline temperature in the channel 25-18 at the level 475 cm from the core bottom versus time



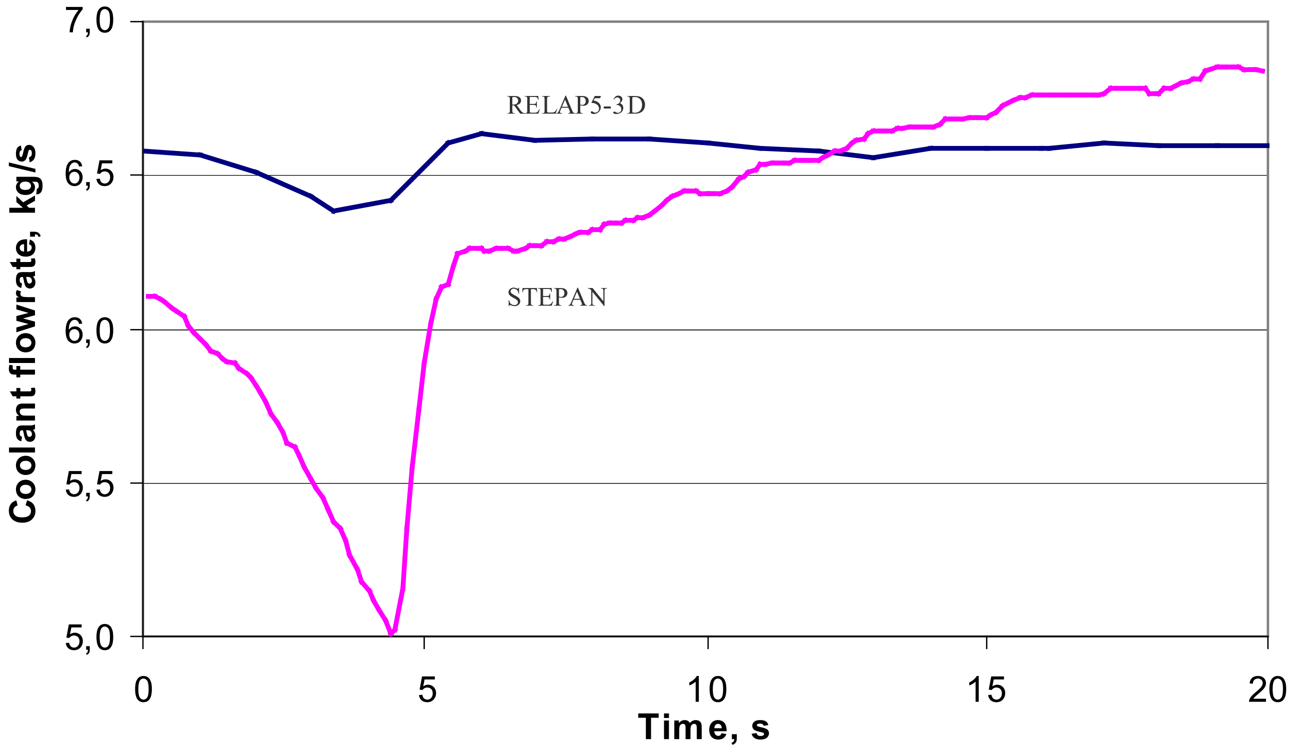
Group of control rods withdrawal (5)

Fuel cladding temperature in the channel 25-18 at the level 475 cm from the core bottom versus time

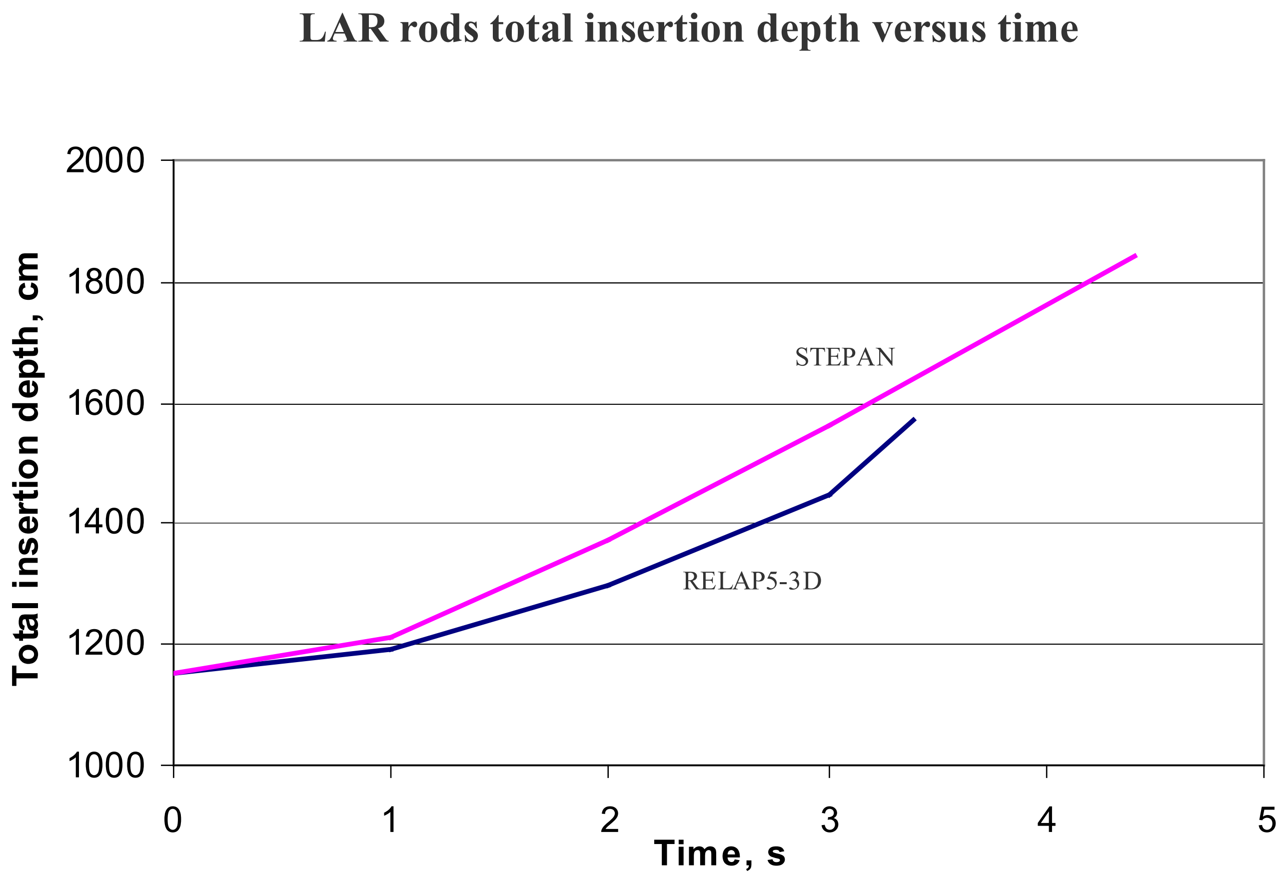


Group of control rods withdrawal (6)

Coolant flowrate at the inlet of the fuel channel 25-18 versus time



Group of control rods withdrawal (7)



Group of control rods withdrawal (8)

In general, RELAP5-3D and STEPAN calculation results show minimal qualitative and quantitative results coincidence for this transient.

Regarding the agreement of separate parameters behavior in time as calculated by both codes, it could be summarized as follows:

- Reasonable agreement was obtained for: i) total reactor power; ii) fuel cladding temperature in nearby channel; iii) LAR rods total insertion depth;
- Minimal agreement was obtained for: i) power in nearby channel; ii) fuel centerline temperature in nearby channel; iii) coolant flowrate in nearby channel.

Feedwater flowrate perturbation (1)

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 tons per hour. 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. Positions of all other control rods are not changed during this experiment.

Increasing of feedwater flowrate by 200 tons per hour 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 centimeters for each automatic regulator.

Feedwater flowrate perturbation (2)

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.

Since the real transient data is not available for the 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.

Feedwater flowrate perturbation (3)

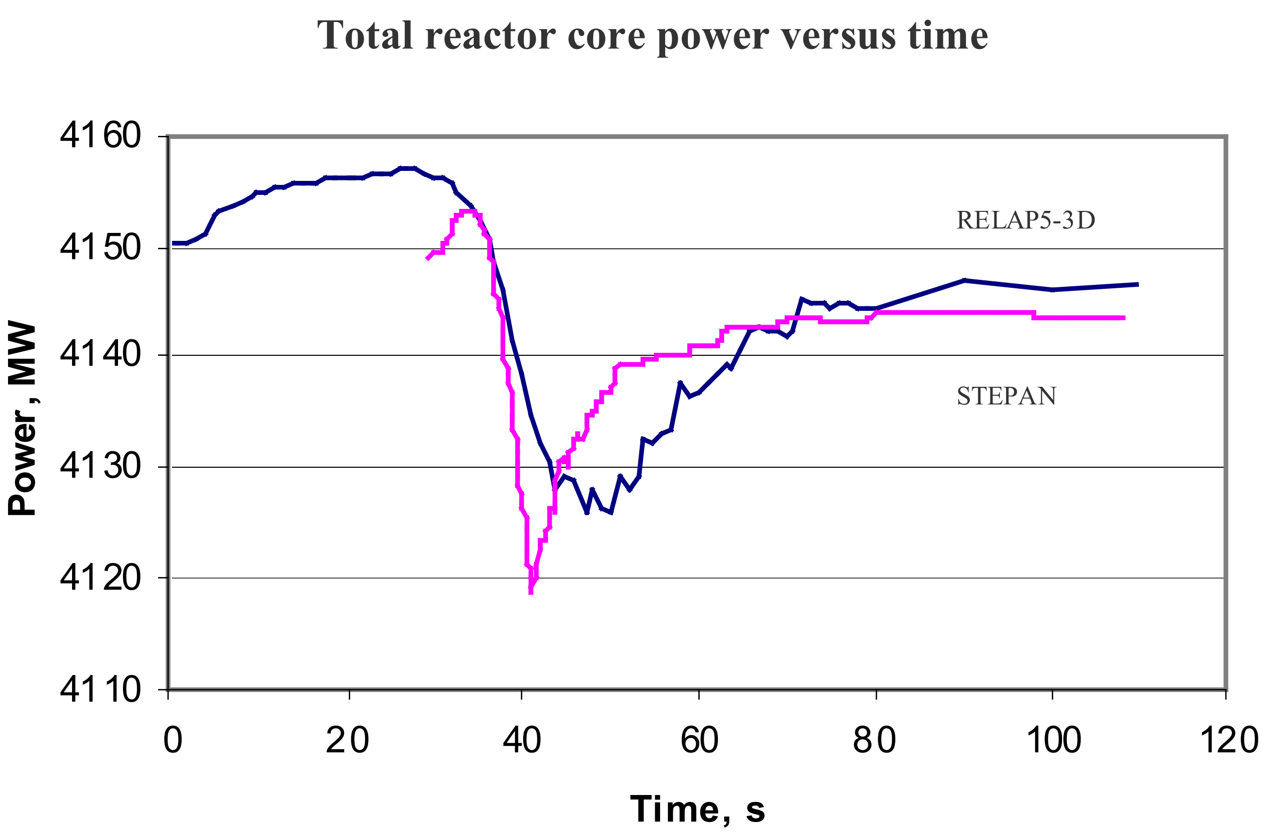
Table 3. Initial/final calculated and measured automatic control rod positions

ΔG_{fw} , 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
Calc. (Final)	285(5)	226(64)	280(10)	274(6)	266.25 (21.25)
Measu-rement (In./Fin.)	300/290	300/290	300/290	300/280	300/ 287.5

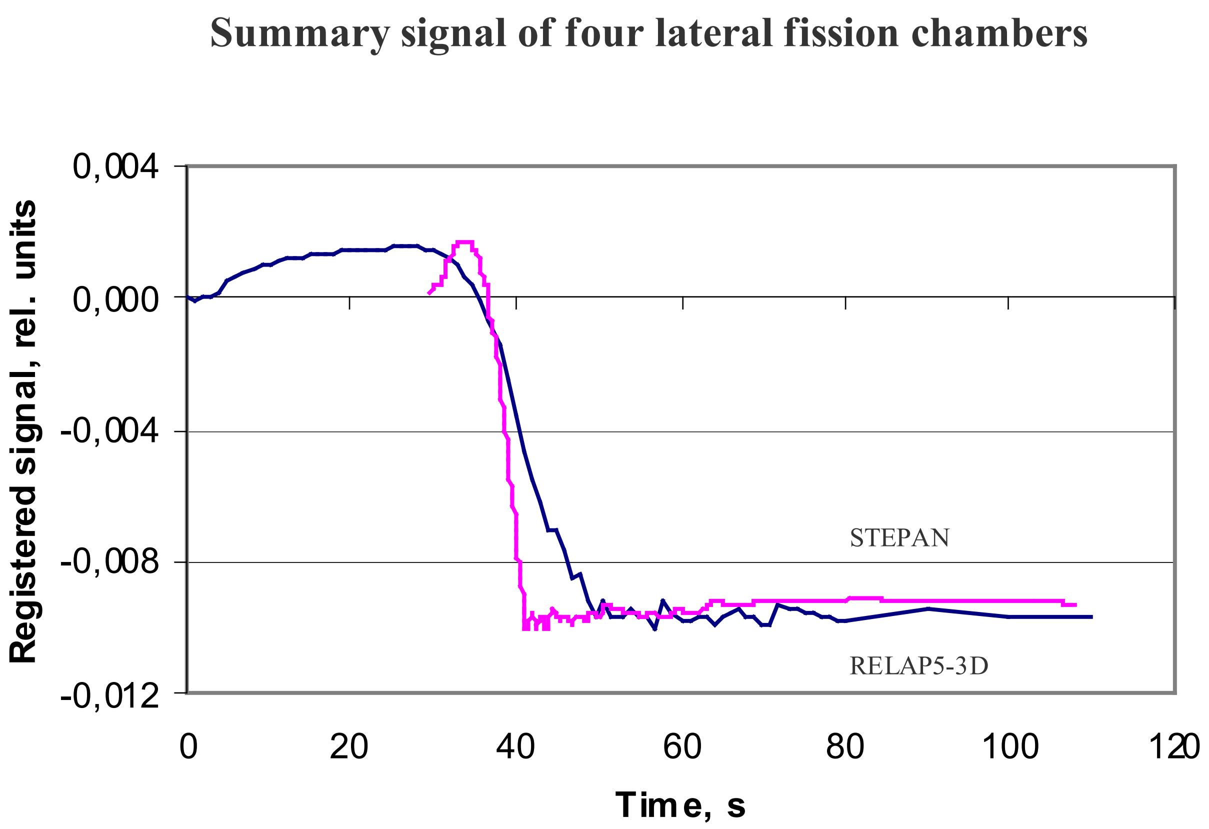
In Table 3 the initial/final calculated and measured automatic control rod positions are presented. In parenthesis differences between experimental (measured) and RELAP5-3D final calculation results are presented as well.

According to the RELAP5-3D calculation results, automatic regulators average shift is more than obtained during the experiment (33.75 and 12.5 centimeters 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.

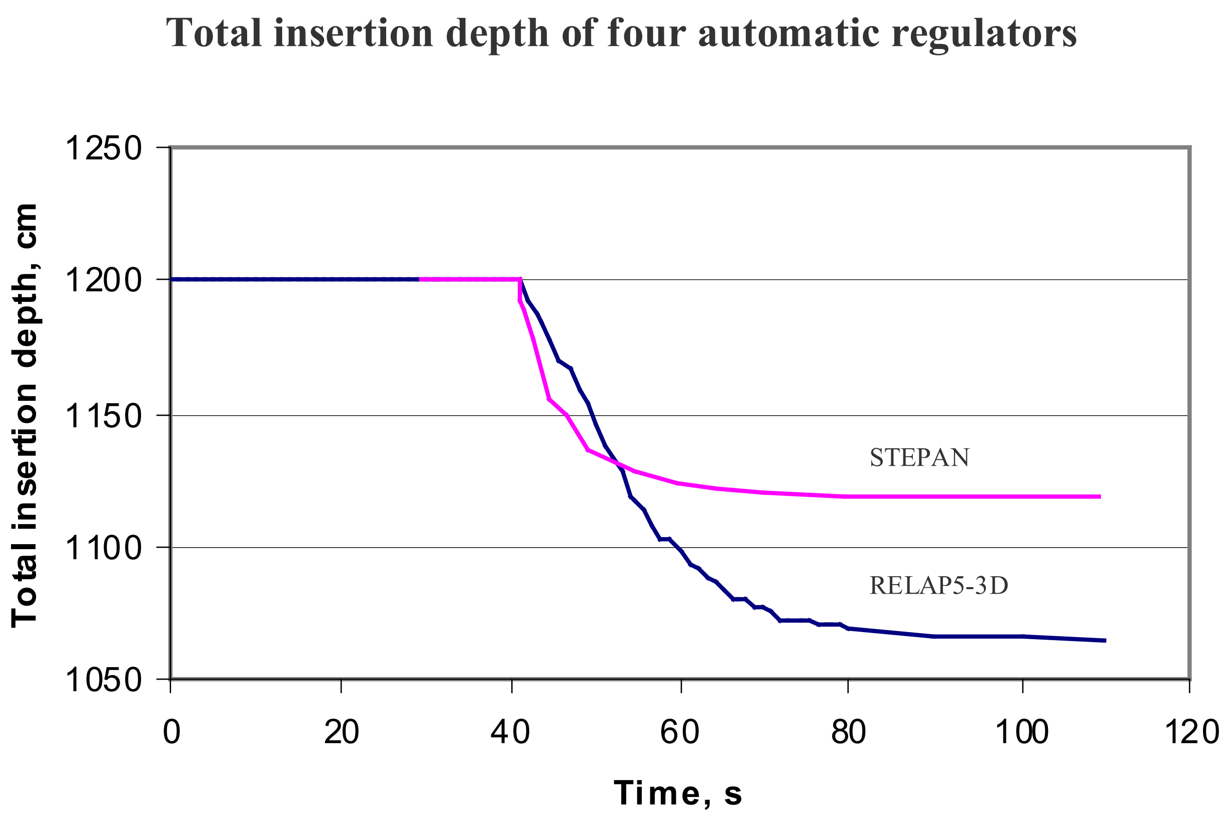
Feedwater flowrate perturbation (4)



Feedwater flowrate perturbation (5)



Feedwater flowrate perturbation (6)



Reactor power reduction (1)

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 seconds from the top end switch position, while all the rest control rods are fully inserted into the reactor core during ~13 seconds 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) - the decreased reactor core power value used in the benchmark.

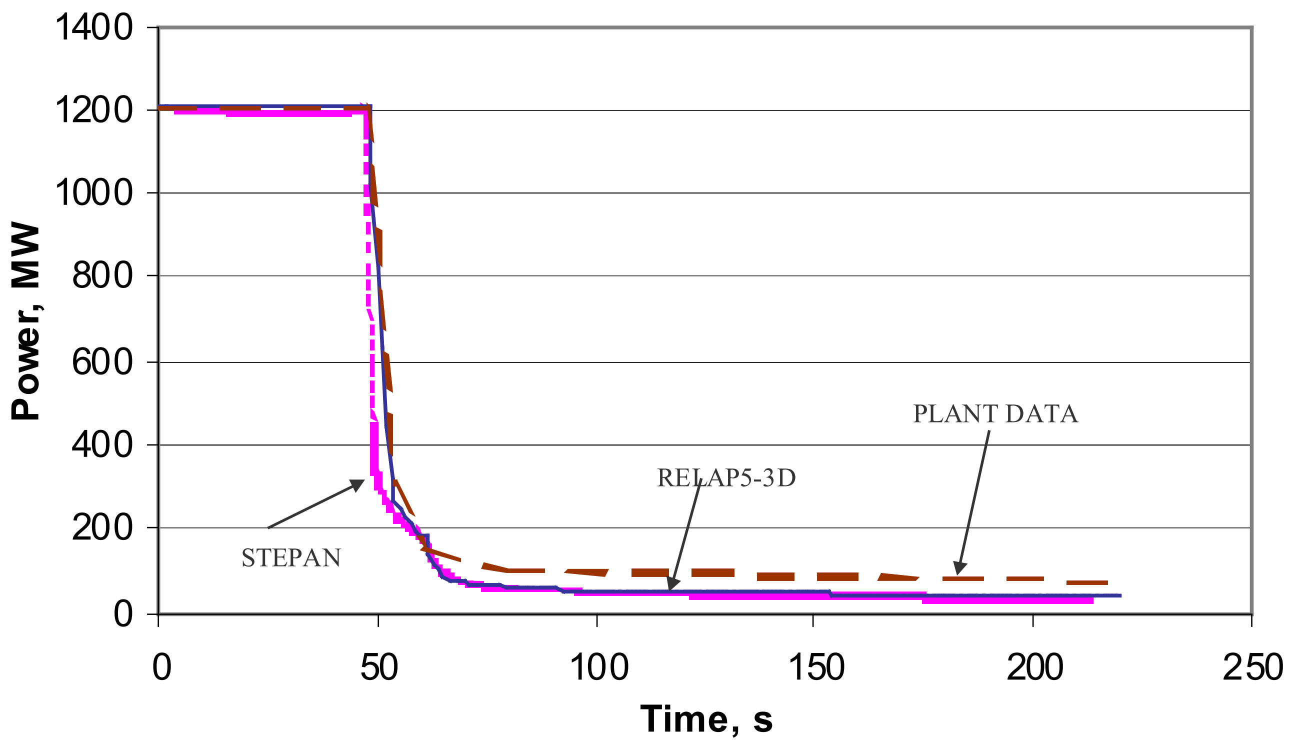
Reactor power reduction (2)

Calculation modeling of the scram signal was performed according to the following scenario: 24 fast acting scram rods were inserted into the reactor core from the top end switch beginning from the 48 second. 13 seconds 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.

According to RELAP5-3D and STEPAN calculation results, the insertion of 24 fast acting scram rods beginning from 48 second of the reactor shutdown transient causes sharp total reactor core power decreasing.

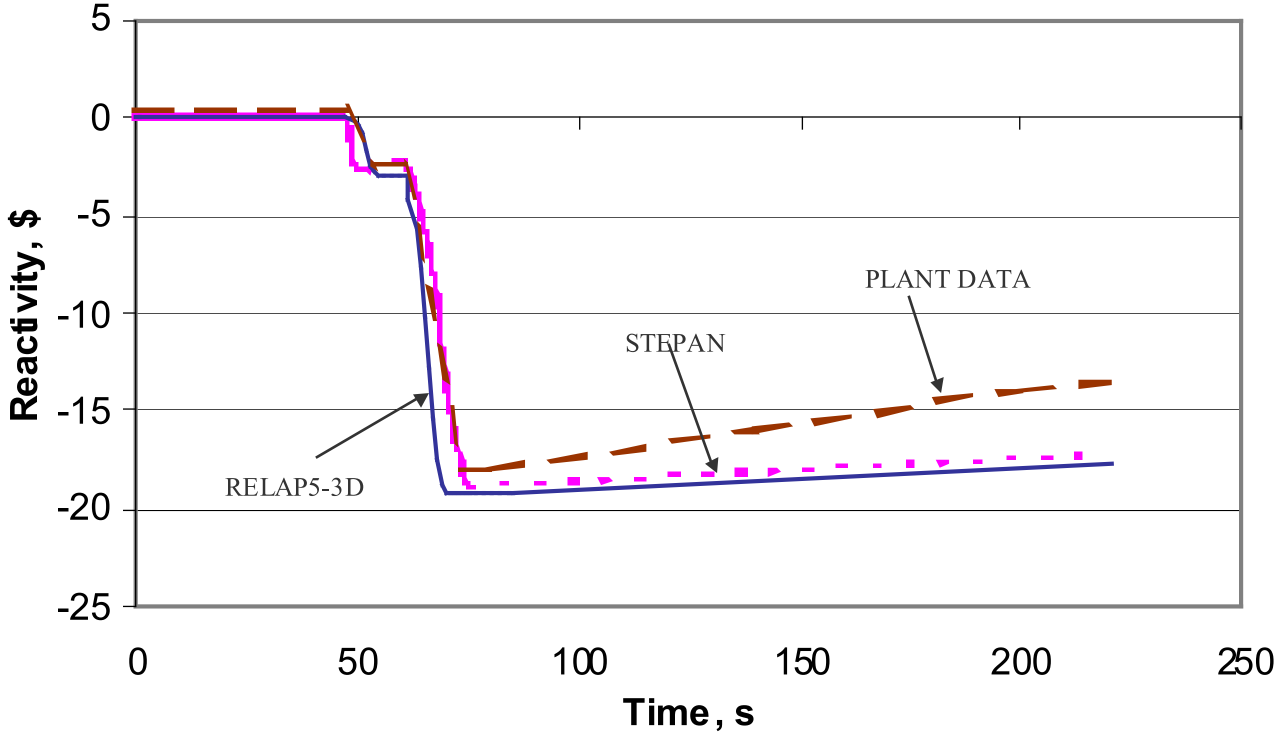
Reactor power reduction (3)

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



Reactor power reduction (4)

Calculation "counter" reactivity behavior during the reactor shutdown transient



Reactor power reduction (5)

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 well 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 (1)

- 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 four 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: single control rod and group of control rods inadvertent withdrawal, feedwater flow perturbation and reactor power reduction transients;
- All 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, where such data was available for comparison;

Conclusions (2)

- Both codes proved to be able to reproduce experimental results with certain accuracy;
- Regarding the control rod spontaneous withdrawal benchmarks - the calculation results produced by both codes are only in minimal agreement among themselves;
- Comparison of the calculation results obtained, using the RELAP5-3D and STEPAN codes for the rest two benchmarks, showed reasonable mutual coincidence of the calculation results and their reasonable agreement with real plant data;
- The proposal for the future would be to proceed with benchmark calculations using RELAP5-3D code for the transients taking place in RBMK-1500 reactors and especially for the cases, where experimental data from the Ignalina NPP

  is available for comparison with calculation results.

Acknowledgments

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- The authors would like to acknowledge also the technical support and access to the STEPAN x-section library provided by RRC “Kurchatov Institute. Especially we want to thank RRC “KI” experts Dr. A. Krajushkin, Mr. A. Balygin and Dr. A. Glembotskiy for their advises and their contribution to the successful realization of the project;
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