Proceedings of ICONE10
10th International Conference on Nuclear Engineering
Arlington, VA, April 14-18, 2002

ICONE10-22450

RELAP5-3D CODE APPLICATION FOR RBMK-1500 REACTOR CORE ANALYSIS

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ABSTRACT

The paper presents an evaluation of RELAP5-3D code 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 the Ignalina NPP RBMK-1500 reactor has been developed and validated against real plant data. 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 the RELAP5-3D and STEPAN codes, showed quite good mutual coincidence of the calculation results and good agreement with real plant data.

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. In this paper 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 [1]. The two 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, specially designed for RBMK reactor analysis, as well as to the real plant data registered by the TITAN information computer system at Ignalina NPP.

NOMENCLATURE

ACS	Accident Confinement System
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
FASS	Fast Acting Scram System
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
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
MFWP	Main Feed Water Pump
MSV	Main Safety Valve
NPP	Nuclear Power Plant
RBMK	Large Channel Type Water Cooled Graphite Moderated Reactor
RDIPE	Research and Development Institute of Power Engineering
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

DESCRIPTION OF RELAP5-3D MODEL

The main purpose for using RELAP5-3D code in our analysis was that RELAP MOD3.2 code was not capable to predict local effects taking place in such a big reactor core as that of RBMK-1500 reactor. RELAP 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.). Regarding the capabilities of the code, several key-features of RELAP5-3D should be mentioned as well: 3D hydrodynamics, multidimensional neutron kinetics, new matrix solvers for increased speed of 3D problems, improved water properties for enhanced robustness, reflood model working again, backward compatibility with MOD3.2 decks, executable under UNIX, LINUX, Windows-NT and Windows 95, user friendly RELAP graphical user interface (RGUI), external kinetics subroutine for material x-section generation for the coupled hydrodynamic-kinetics calculations.

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 [2]. 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" [3] 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 flow rate through the pump. In the RELAP5 pump model the four-quadrant characteristics are expressed by so-called homologous curves [4]. The throttling-regulating valves are used for coolant flow rate regulation through the core. These valves are modeled by employing "servo valve" [3] 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 to. These channels are cooled by separate water circuit (17).

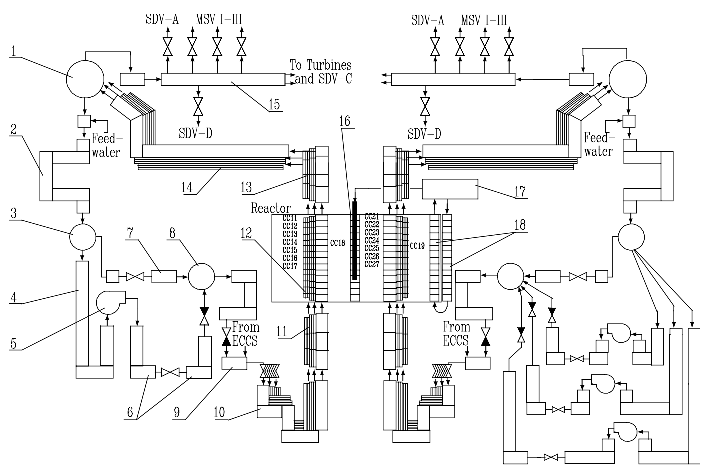

Fig. 1. Ignalina NPP thermal-hydraulic model nodalization diagram: 1 - DS, 2 - downcomers, 3 - MCP Suction Header, 4 - MCP suction piping, 5 - MCPs, 6 - MCP discharge piping, 7 - bypass line, 8 - MCP Pressure Header, 9 - GDHs, 10 - lower water communication line, 11 - reactor core inlet piping, 12 - 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 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" [3] 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 flow rate 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 feed water injection into the DS is simulated explicitly using RELAP5 "pipe", "junction", "volume" and "pump" elements. The nodalization scheme of the feed water system is not presented in this paper. The feed water from the deaerators (in which the available amount of water is 480 m^3) is supplied to the MCC by Main Feed Water 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.

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.

The energy that is dissipated around the MCC is evaluated by determining the energy added to the fluid in the MCPs. The use of RELAP5-3D code allows the description of the heat exchange between technological channels, CPS channels and the reflector cooling circuit without using the detailed reactor gas circuit model. This heat transfer is described using the special 'Conduction Input' option. The heat transfer between 14 equivalent fuel channels, one equivalent CPS channel and one equivalent reflector cooling circuit channel are modeled. Heat exchange between graphite columns occurs along all length of equivalent channel.

Thermal hydraulic part of Ignalina NPP RELAP5-3D model was validated against real plant transients and the validation results presented in [1]. The results of the calculations obtained with RELAP5-3D model on the Ignalina NPP specific base compare favorably with the real plant data.

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 (see Fig. 2). 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.

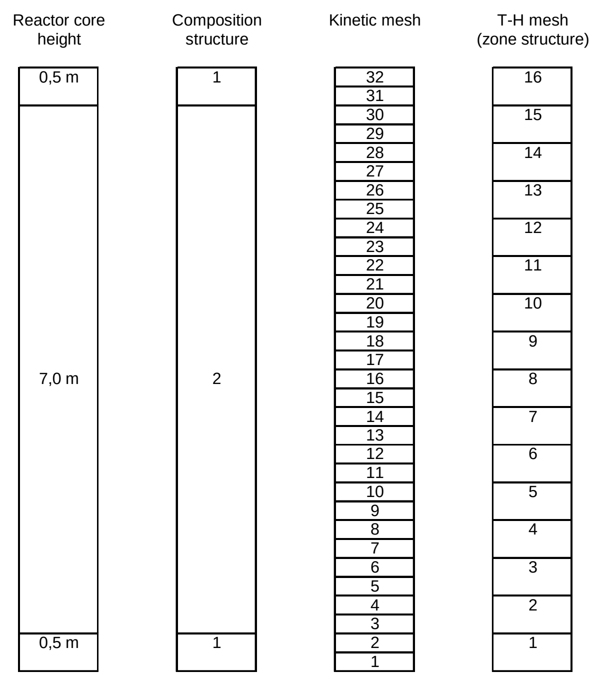

Fig. 2. Kinetic mesh, thermal-hydraulic mesh and material composition structure of Ignalina NPP RBMK-1500 reactor RELAP5-3D model

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 to us 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.

Figures 3 and 4 show the assignment of thermal-hydraulic channel groups to the radial kinetics nodes of the RBMK-1500 reactor core, correspondingly for INPP Unit 2 reactor core states registered on November 26, 1998 and on March 29, 1999. 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. Tables 1 and 2 show the assignment of thermal-hydraulic channels to each group, correspondingly for INPP Unit 2 reactor core states registered on November 26, 1998 and on March 29, 1999. The number of channels in each group varies from 2 to 378 and from 2 to 308, respectively. As shown, channel groups CC11, CC12, CC21 and CC22 are located in the center of the reactor core, and 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. ‘CC’ represents the thermal-hydraulic axial mesh number. The kinetics part of the model models each fuel and non-fuel channel individually, as shown in Figures 3 and 4.

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.

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*		

* 436 channels are radial reflector channels

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*		

* 436 channels are radial reflector channels

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.

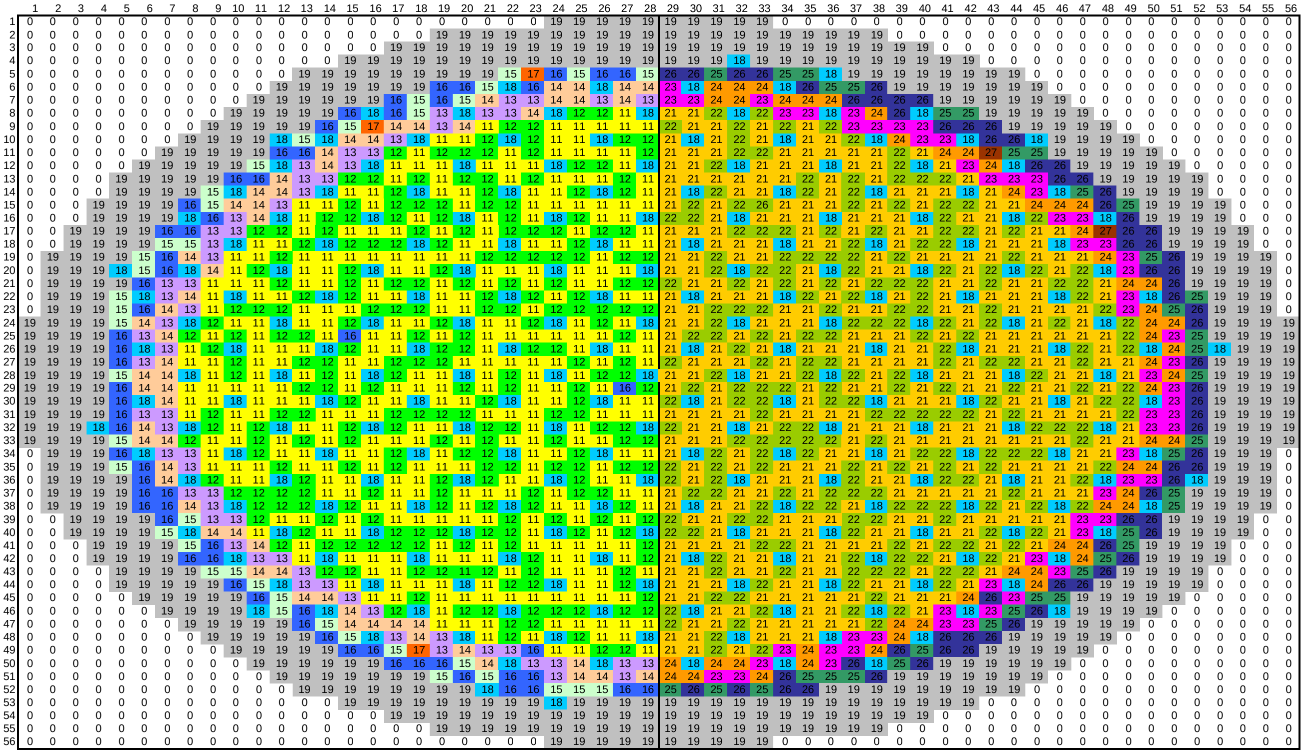

Fig. 3. Nodalization scheme of the RBMK-1500 reactor core (Unit 2, November 26, 1998).

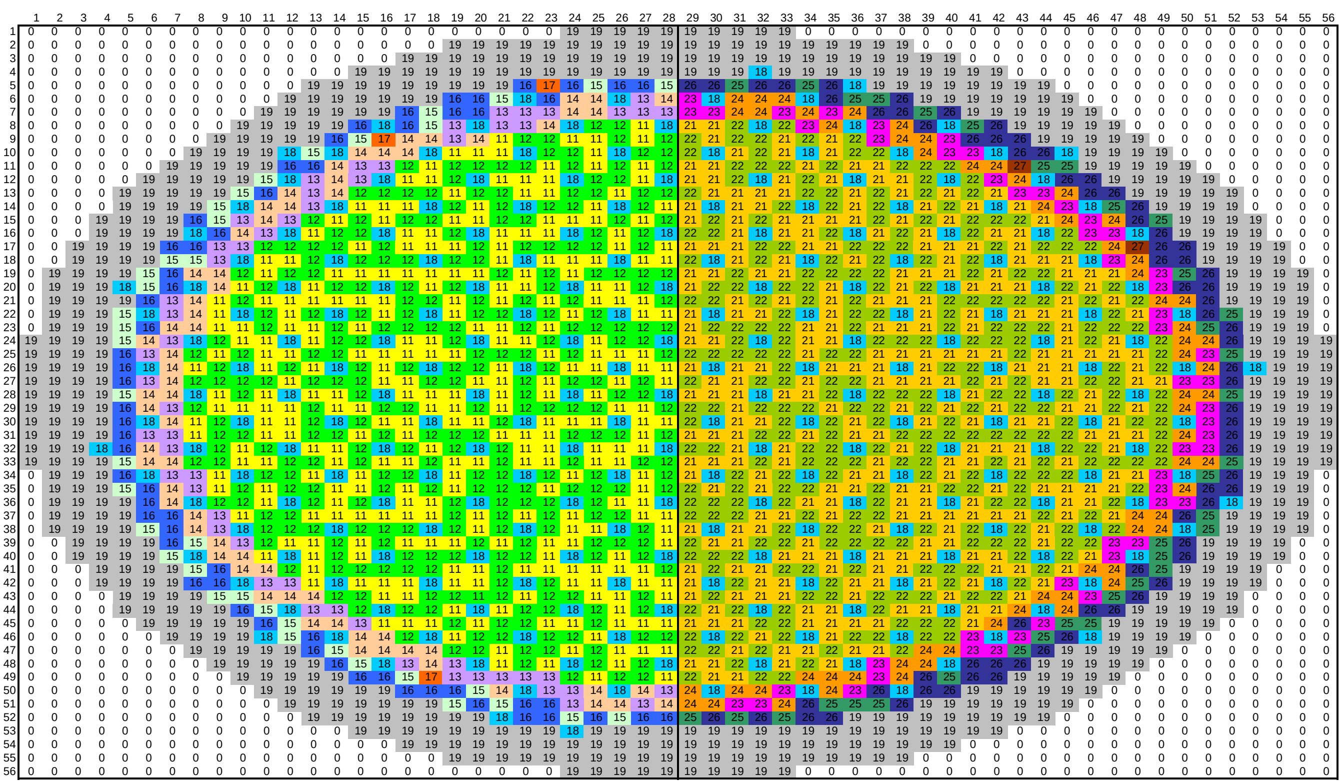

Fig. 4. Nodalization scheme of the RBMK-1500 reactor core (Unit 2, March 29, 1999).

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. Movement of LAR rods continues until the signal that initiated their movement is no longer valid. Then the LAR rods stop moving and hold their current positions until another signal to insert or withdraw a certain control rod is received. The LAR rods move individually, depending on the power deviation signals coming from radial detectors located in their corresponding LAR zones. LEP rods can move either together with the LAR rods based on the overpower signals coming from detectors located in a certain LAR zone, or they can move separately from LAR rods based on the overpower signals coming from detectors that belong to a certain LEP zone. LEP rods move only into the core, but never out of the core. If two overpower signals are coming from DKER-1 detectors of different detector groups that belong to a single LEP zone at the same time, the AZ-6 signal in that LEP zone is initiated. The AZ-6 signal initiates the AZ-3 signal if reactor power is more that ½ of design reactor power. The AZ-3 signal causes reactor power to be reduced to ½ of design reactor power. If the AZ-6 signal is still valid when reactor power is ½ of design reactor power, reactor power is reduced further until AZ-6 signal disappears.

Modeled are one more fast controlled automatic emergency reactor power decrease mode (AZ-4), which decreases reactor power to 60% of design reactor power and which is initiated by technological parameters deviation from set-points, and two reactor emergency protection modes (FASS and AZ-1), which are triggered based on neutronic and technological parameters deviation from set-points. In FASS or AZ-1 mode all 211 control rods are inserted into the core and the reactor is shutdown in 12÷14 seconds. In FASS mode fast acting scram rods are inserted in 2÷2.5 seconds into the core, while in AZ-1 mode they are inserted in 5÷7 seconds.

Nodal kinetics part of Ignalina NPP RELAP5-3D model was validated against real plant data in static neutron-physics calculations and the validation results presented in [1]. The steady-state calculation results of RBMK-1500 reactor core state obtained using RELAP5-3D code agree well to the real plant data. The RELAP5-3D nodal kinetics model represents the Ignalina NPP Unit 2 reactor power and coolant density profiles reasonable well, too. Eigenvalue close to unity indicates reasonable values are calculated for neutron fluxes.

FEEDWATER FLOW RATE PERTURBATION

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, serving as a biological protection shield. 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.

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 (AR) 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 (№ 3, 9, 15, 21) measured neutron field change. Four automatic control rods were changing their positions to compensate the reactivity change.

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.

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

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. 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 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.

According to RELAP5-3D and STEPAN calculation results (see Fig. 5), 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, travelling 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.

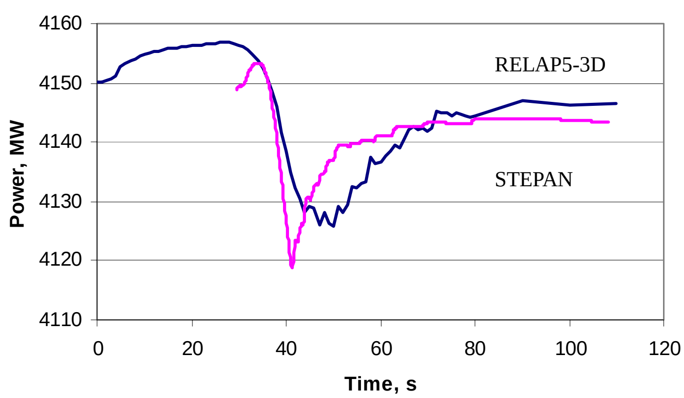

Fig 5. Total reactor core power versus time. Feedwater flow perturbation experiment

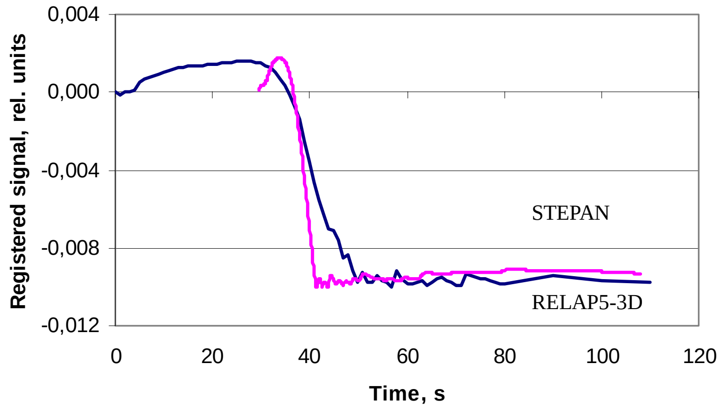

Fig. 6. Summary signal of four lateral fission chambers. Feedwater flow perturbation experiment.

According to RELAP5-3D and STEPAN calculation results (see Fig. 6), 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%.

Both codes overestimate the total insertion depth of four automatic regulators during the transient (see Fig. 7). RELAP5-3D code overestimates this value by 85 cm, while STEPAN code overestimates it by 32 cm.

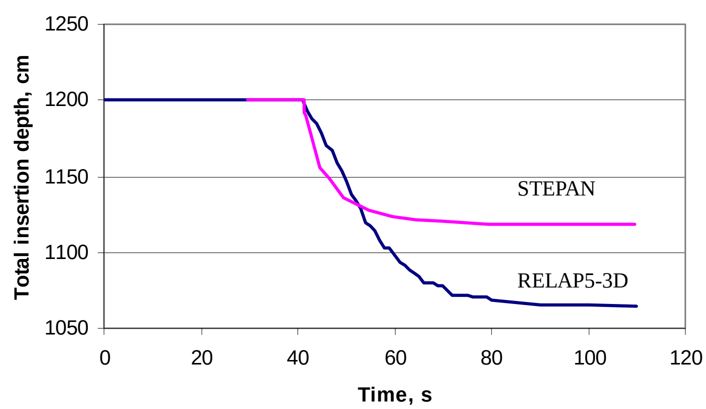

Fig. 7. Total insertion depth of four automatic regulators. Feedwater flow perturbation experiment.

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 quite good mutual coincidence of the calculation results and good agreement with real plant data.

REACTOR POWER REDUCTION

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 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). This decreased reactor core power was taken as the basis for our 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 second 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 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.

Calculation results using RELAP5-3D code are presented for 220 seconds time period, including 48 seconds calculation of zero transient before the reactor scram signal initiation, to match the presented real plant data. Like in previous benchmarks water flow rate 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 second of the reactor shutdown transient causes sharp total reactor core power decreasing. Figure 8 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 second), 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 well to each other.

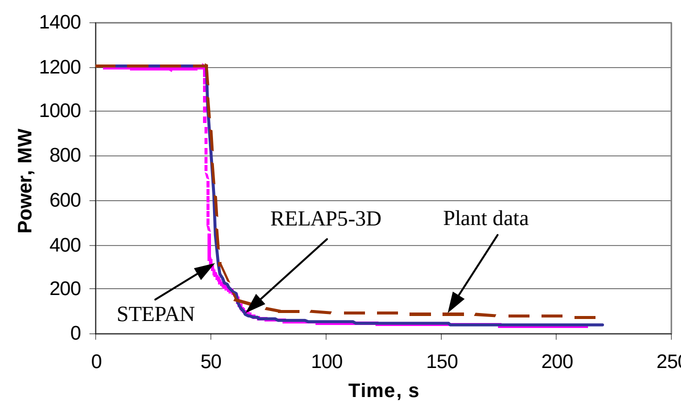

Fig. 8. The total reactor core power behavior versus time during the reactor shutdown transient.

Figure 9 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 second), 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 second), than the reactivity calculated by STEPAN code. At the final reactor shutdown point (beginning from ~75 second), 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 well to each other.

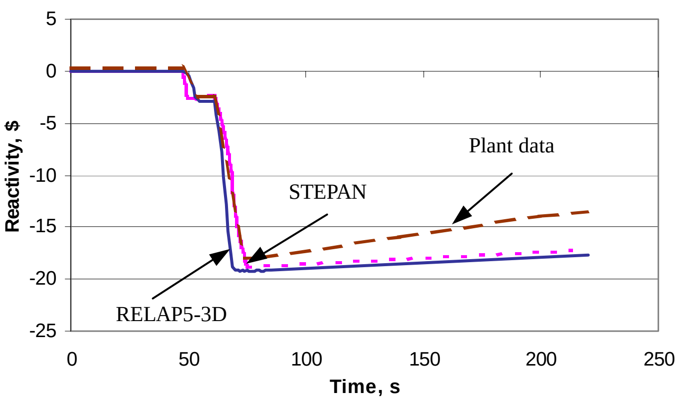

Fig. 9. Calculation "counter" reactivity behavior during the reactor shutdown transient.

In general, RELAP5-3D and STEPAN codes give quite good mutual coincidence of the calculation results and good 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 two 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. Comparison of the results obtained, using the RELAP5-3D and STEPAN (specially designed for RBMK reactor analysis) codes, showed quite good mutual coincidence of the calculation results and good agreement with real plant data.

ACKNOWLEDGMENTS

The authors would like to acknowledge the technical (from INEEL), financial support and access to the code RELAP5-3D provided by the US DOE, the US International Nuclear Safety Program (INSP). Especially we want to thank two experts from INEEL, Dr. James Fisher and Dr. Paul Bayless, who helped a lot with their advises and sharing their experience. All this made it possible to bring together the analytical methodology developed in the US with the design knowledge of the unique RBMK systems available in Lithuania. 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, Dr. A. Balygin and Dr. A. Glembotskiy for their advises and their contribution to the successful realization of the first part of the project. We also want to extend our thanks to the administration and technical staff of the Ignalina NPP, for providing information regarding operational procedures and operational data.

This work was supported by the U.S. Department of Energy, National Nuclear Security Administration. The U.S. Government makes no endorsement of the results of this work.

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