VALIDATION OF COUPLED RELAP5-3D CODE IN THE ANALYSIS OF RBMK-1500 SPECIFIC TRANSIENTS

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Abstract

This paper deals with the modelling of RBMK-1500 specific transients taking place at Ignalina NPP. These transients include: measurements of void and fast power reactivity coefficients, change of graphite cooling conditions and reactor power reduction transients. The simulation of these transients was performed using RELAP5-3D code model of RBMK-1500 reactor. At the Ignalina NPP void and fast power reactivity coefficients are measured on a regular basis and, based on the total reactor power, reactivity, control and protection system control rods positions and the main circulation circuit parameter changes during the experiments, the actual values of these reactivity coefficients are determined. Graphite temperature reactivity coefficient at the plant is determined by changing graphite cooling conditions in the reactor cavity. This type of transient is very unique and important from the gap between fuel channel and the graphite bricks model validation point of view. The measurement results, obtained during this transient, allowed to determine the thermal conductivity coefficient for this gap and to validate the graphite temperature reactivity feedback model. Reactor power reduction is a regular operation procedure during the entire lifetime of the reactor. In all cases it starts by either a scram or a power reduction signal activation by the reactor control and protection system or by an operator. The obtained calculation results demonstrate reasonable agreement with Ignalina NPP measured data. Behaviours of the separate MCC thermal-hydraulic parameters as well as physical processes are predicted reasonably well to the real processes, occurring in the primary circuit of RBMK-1500 reactor. Reasonable agreement of the measured and the calculated total reactor power change in time demonstrates the correct modelling of the neutronic processes taking place in RBMK-1500 reactor core. And finally, the performed validation of RELAP5-3D model of Ignalina NPP RBMK-1500 reactor allowed to improve the model, which in the future would be used for the safety substantiation calculations of RBMK-1500 reactors.
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. 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: the simulation of the measurements of void and fast power reactivity coefficients, change of graphite cooling conditions and reactor power reduction transients that were registered at Ignalina NPP. All benchmarks were modelled using RELAP5-3D code and the calculation results compared to the real plant data registered by the TITAN information computer system at Ignalina NPP.

Description of RELAP5-3D model

The RBMK-1500 is graphite moderated, boiling water, multi-channel reactor [1]. The main purpose for using RELAP5-3D code for the above mentioned transient 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 modelling 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.

Thermal-hydraulic part of Ignalina NPP RELAP5-3D model

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. Two downcomers DS in each MCC loop are modelled by generalised “separator” element (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 “pump” 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 modelled by employing “servo valve” [2] elements. The normalized flow area versus normalized stem position is described in the RELAP5 model. The bypass line (7) between the pump suction header and the pump pressure header is modelled 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 pass 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 modelled 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 modelled 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 modelled too. These channels are cooled by a separate water circuit (17).

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 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 modelled by “servo valve” 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 ACS 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.

The reactor core has 14 thermal-hydraulic channels for the fuel channels and 2 thermal-hydraulic channels for the non-fuel channels (CPS cooling circuit channels and radial reflector cooling channels). The fuel channels were divided into 7 groups according to power and coolant flowrate values. Square profile 0.25 x 0.25 m graphite blocks are modelled by cylindrical heat structures with
the equivalent cross-section area. Each equivalent channel is modelled using 16 axial nodes of 0.5 m length each. The fuel element is modelled 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 modelled 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.

**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. Besides, the kinetics part of the model models each fuel and non-fuel channel individually.

The developed models of nodal reactor kinetics are based on two real states of the reactors of Ignalina NPP. The first database that was used was a database of INPP Unit 1, registered by INPP ICS on November 22, 2000. This database was used for the modelling of reactor power reduction by the actuation of AZ-1 emergency protection signal, as well as for the modelling of the change of graphite cooling conditions. The second database was a database of INPP Unit 2, registered by INPP ICS on October 12, 2000. This database was used for the modelling of void and fast power reactivity coefficients measurement experiments done at Ignalina NPP, as well as for the modelling of reactor power reduction by the actuation of AZ-3 emergency protection signal. Reactor core loading information was obtained from the plant as a part of the databases from the main information computer system “TITAN”. Besides the reactor core loading information, the databases 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 variables.

Cross sections for the different compositions of the RBMK-1500 reactor core were obtained from two-group macroscopic x-section library of the STEPLAN code that was provided by Russian Research Centre “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 kinetic part of the RELAP5-3D code.

The CPS control rods and the CPS operation logic is another complicated part of RBMK-1500 reactor RELAP5-3D model. All CPS 211 control rods are modelled individually, because all of them have different insertion depths into the reactor core. Four types of control rods are modelled: two types of manual control rods, fast acting scram 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 modelling. Movements of the CPS control rods are controlled by the CPS logic, based on the power deviation signals coming from 127 radial detectors. The radial detectors are modelled 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 radial detector location. All radial detectors 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.

**Change of graphite cooling conditions simulation benchmark**

Change of graphite cooling conditions from helium-nitrogen gas mixture to nitrogen (during reactor planned shutdown or vice versa during reactor startup) is performed during the reactor operation at the power level of not more than 2100 MWt. Composition of helium-nitrogen gas mixture: ~90% helium and ~10% nitrogen (by volume). Presented below are the RELAP5-3D calculation results, obtained during the modelling of the graphite cooling conditions change, performed at Ignalina NPP Unit 1 on August 8, 1999. In reality, the change of the graphite cooling conditions results in a slow change of the graphite outer surface temperature. The graphite outer surface temperature change process due to the graphite cooling conditions change takes ~7 hours. The simulation of ~7 hours process of the graphite cooling conditions change is impractical with RELAP5-3D code because of the huge computer time resources and disc space that are needed. So during the simulation of the change of the graphite cooling conditions process it was assumed, that the gas mixture composition in the reactor gas circuit changes step-wise. The whole range of gas mixture composition change (from -15% N₂ and 85% He to -96.4% N₂ and 3.6% He) was simulated by five intermediate gas mixture composition change steps.

Initial calculations were performed using only the thermal-hydraulic part of RELAP5-3D model, where artificially the total reactor power value was kept constant and no reactivity feedbacks were evaluated. Due to the gap between the fuel channels and the graphite bricks characteristics change, the graphite stack temperature was changing as well. According to the methodology presented in [4], the thermal conductivity coefficient values for the gap between the fuel channels and the graphite bricks were calculated for each of the five intermediate steps of the graphite cooling conditions change process. Such initial calculations allowed us to determine, what graphite stack temperature values would be after a certain gas mixture composition change in the reactor gas circuit. Further calculations were conducted already using the full basic RELAP5-3D model of RBMK-1500 reactor, where the graphite temperature reactivity feedback was evaluated and the reactor power control system was activated. Using the ‘restart’ option of the RELAP5-3D code and starting from steady-state conditions, thermal conductivity of the gap between the fuel stacks was recalculated for each of the five intermediate steps of the graphite cooling conditions change process. The result was a series of five input files, each containing the temperature distribution across the fuel channels and the graphite bricks for a specific gas mixture composition change step. These input files were then used as initial conditions for the full RELAP5-3D simulation of the graphite cooling conditions change process. The simulated graphite temperature change process was then compared with the experimental measurements, which were performed using an in-reactor probe (INPP measurement).

![Graph showing recalculated average graphite temperature during the transient](image)

Figure 2. The recalculated average graphite temperature during the transient
channels and the graphite bricks and the graphite stack temperature values were changed step-wise during the transient calculation of the modelled graphite cooling conditions change process. It needs to be stressed here, that only the results of the graphite cooling conditions change, performed at Ignalina NPP, might lead to the validation of the gap between the fuel channels and the graphite bricks and graphite temperature reactivity feedback models. This is because only during this transient the graphite temperature changes with constant reactor power conditions.

Calculation of the five stages of gas mixture composition change was simulated for 100 s each. As the real graphite cooling conditions change process at the Ignalina NPP lasted for ~7 hours, for the comparison of the calculated and measured reactor parameters, only final points from the RELAP5-3D calculations are plotted in Figs. 2÷3 at the corresponding time points.

During the initial calculations it was determined that the contact resistance between the graphite rings and the pressure tube or the graphite bricks changes. It is likely that when the graphite stack temperature increases, graphite rings expand more and the contact gap between the graphite rings and the pressure tube walls or the graphite bricks decreases. Based on the average graphite temperature measurement, the dependence of the contact gap versus the average graphite temperature was determined. While estimating this contact gap change, the thermal conductivity values of the gap between the pressure tube walls and the graphite bricks were calculated. Using the calculated thermal conductivity values of the gap, the dependence of the average outer graphite stack surface temperature values on the amount of helium in the reactor gas circuit was obtained. The calculated graphite stack temperature values show reasonable agreement with the measured plant data (see Fig. 2). During this transient the total reactor power was kept constant at the level of ~1370 MWt.

The stable total reactor core power during the simulation of the gas mixture composition change in the reactor gas circuit was achieved by the proper operation of the LAR/LEP control rods of the CPS. While LAR/LEP rods were compensating for the reactivity rise due to the graphite stack temperature rise as a result of the change of graphite cooling conditions, the operative reactivity margin was increasing during the whole simulation time. The behaviour of the reactor operative reactivity margin during the simulated process is shown in Fig. 3. In general the calculated operative reactivity margin value corresponds quite reasonably to the measured plant data, but the calculated ORM value is clearly overestimated by RELAP5-3D code during all the simulated process of the graphite cooling conditions change. A certain time delay between the end time point of each of the simulated step of the reactor cavity gas composition change and the stabilization of the corresponding ORM value was taken into account when plotting the final ORM values calculated by RELAP5-3D code for each simulated step of the overall graphite cooling conditions change process. One can see from Fig. 3, that the final calculated ORM value is ~2% higher than the corresponding ORM value measured at the plant. This difference can be explained by the RELAP5-3D model peculiarities and/or the materials cross-section library used for the calculations.
In general, the graphite cooling conditions change process, modelled with the RELAP5-3D code, corresponds reasonably to the real physical process taking place in RBMK-1500 reactor. All calculation results are in reasonable agreement with the measured plant data and are well within the measurement/calculation error range.

**Void reactivity coefficient measurement simulation benchmark**

During the void reactivity coefficient measurement, which was performed at Ignalina NPP Unit 2 on October 12, 2000 [5], local automatic control system of CPS was switched off. Switched on were four 2AC control rods in the channel (individual) control mode. The initiator of the transient was the additional water supply from AFWP, starting from time $t=12$ s (see Fig. 4). According to the real plant data, at time $t=12$ s the increase of feedwater flowrate through AFWP by 190 t/h (52.8 kg/s) for both reactor core halves was modelled. Later, at time $t=157$ s the decrease of feedwater flowrate through AFWP also by 190 t/h was modelled as well. During the void reactivity coefficient measurement, feedwater flowrate through MFWP was artificially kept constant in the model (feedwater automatic control was switched off at Ignalina NPP).

Additional feedwater flowrate through AFWP to the DS of both reactor core halves disturbed the total coolant mass balance in the reactor (feedwater flowrate to the reactor was greater than the steam flowrate from DS to turbines). Following the feedwater flowrate through AFWP increase at time $t=12$ s, water volume in DS of both reactor core halves started to increase (see Fig. 5). RELAP5-3D calculation results show that water level in the DS reaches its maximum value at time $t=220$ s of the transient and after that remains constant. But according to the Ignalina NPP data, water level in DS is increasing slightly during the time period of $t=220\div300$ s (see Fig. 5) during the void reactivity coefficient measurement. Such small difference in the calculated and the measured water level change can be caused by slightly less amount of steam supplied to the turbines during the real plant measurements. Unfortunately such assumption is possible to make only based on the change of the secondary parameters, because the steam supply into turbines was not measured during this transient.
As it was mentioned above, coolant temperature start to decrease in the MCC due to the feedwater flowrate increase through AFWP at time moment t=12 s. Due to the cold water supply, the amount of steam generated in the reactor core decreases. Void fraction of the coolant leaving the reactor core decreases, too. This explains the negative reactivity insertion into the reactor core (see Fig. 6) and the subsequent total reactor power decrease (see Fig. 7) at time t≈50 s.

Following feedwater flowrate increase by 190 t/h and the subsequent reactivity and the total reactor power decrease, four 2AC control rods started to withdraw from the core trying to compensate the negative reactivity insertion caused by the colder coolant flow through the reactor. According to RELAP5-3D calculation results the average shift of the four 2AC control rods was 29 cm versus 13 cm as it is indicated by the real plant data. The difference in the final insertion depths of the four 2AC control rods might be due to the local effects present in the RELAP5-3D code RBMK-1500 reactor core model. After the reduction of the feedwater flowrate through AFWP to its initial value, due to the coolant temperature in the MCC increase by ~1 °C, with a certain time delay positive reactivity was inserted into the core that resulted in the subsequent total reactor power increase at time t=200 s. Four 2AC control rods were automatically inserted again into the core to make the total reactor power decrease.

**Fast power reactivity coefficient measurement simulation benchmark**

During the simulation of the fast power reactivity coefficient measurement, performed at Ignalina NPP Unit 2 on October 12, 2000 [5], in the first step insertion depth of the four 2AC control rods is set to ~3.5 m. LAC system of the CPS is switched off. Later on, according to the procedure of fast power reactivity coefficient measurement, at the start of the transient four 2AC control rods are inserted manually from their initial positions of ~3.5 m to their final positions of 4.0 m. This results in a sharp total reactivity decrease from its initial steady-state value of ~0.0 $\beta_{\text{eff}}$ down to ~$(-0.03)$ $\beta_{\text{eff}}$ (see Fig. 8). Such reactivity change is caused by 2AC control rods insertion into the core in a short period of time (according to the plant data, four control rods are inserted 0.5 m each during the time interval of 2.5 s).
This was the initiator of the transient behaviour of all the reactor parameters presented further in the text below. Negative reactivity insertion into the reactor core results in the total reactor power decrease (see Fig. 9). In a short period of time (in \(\approx 5\div 10\) s) reactor power decreases from \(\approx 3050\) MWt down to \(\approx 2880\) MWt. After the full insertion of 2AC control rods, no actions are done in the time period of \(\approx 1\) min (no manual actuation of control rods). After the time period of \(\approx 1\) min, LAC system of CPS is switched on.

According to the RELAP5-3D calculation results (see Fig. 8), shortly after the end of the reactivity decrease (at time \(t=3.4\) s from the beginning of the transient), reactivity starts to increase and at time \(t=60\) s reaches the reactivity value of \(-0.0065\ \beta_{\text{eff}}\). According to the real plant data, reactivity value of \(\approx (–0.005)\ \beta_{\text{eff}}\) is reached already on \(\approx 30\) s of the transient process and is maintained constant for \(\approx 50\) s. The obvious difference is that the reactivity rises more slowly in RELAP5-3D case than during the experiment. The reactivity rise beginning from the time \(t=3.4\) s from the beginning of the transient is due to the power reactivity effect feedback. Some discrepancies of the reactivity behaviour in time during this transient might be due to the RELAP5-3D model peculiarities and the materials cross-section library used during the calculations.
Due to the reduced reactor power the amount of the generated steam in the core decreases, too. Decrease of the generated steam amount results in the drop of the pressure in DS (see Fig. 10). Drop of pressure in DS activates the pressure controller and turbine control valve of the turbine, operated in the pressure regulation mode, starts to close. Steam supply to this turbine decreases, but the steam supply to the turbine operated in the load maintenance mode remains constant. As could be seen from Fig. 10, operation of the pressure controller stabilizes pressure in DS at the level of ~6.80 MPa. Reasonable agreement of Ignalina NPP measurement data and the calculation results demonstrates adequate operation of the pressure controller model.

The pressure in DS change has its influence on other MCC components as well. For example, pressure in the MCP pressure header decreases also by 0.06 MPa, correspondingly.

Following the reactor power reduction, water level in DS decreases also (see Fig. 11). This decrease is caused by the decrease of the steam part (void fraction) in the coolant. In other words, part of water from DS flows back into the fuel channels. Water level in DS begins to increase only after ~20 s from the beginning of the transient, because steam supply into the turbine decreases correspondingly.

**Reactor power reduction on AZ-1 emergency protection signal simulation benchmark**

Presented below are the RELAP5-3D calculation results, obtained during the modelling of the reactor power reduction transient on the actuation of AZ-1 emergency protection signal that took place at Ignalina NPP Unit 1 on November 22, 2000 [6]. On that date Unit 1 of Ignalina NPP was shutdown upon AZ-1 signal generation based on the erroneous actuation of the emergency protection by the reduction of the operative reactivity margin (ORM).

As the initiating signal (AZ-1) for the reactor emergency shutdown in reality was generated based on an
erroneous signal in the CPS logic circuit, so during the modelling of this transient the initiating AZ-1 signal was simulated manually at time \( t=0 \) s. According to AZ-1 signal, all 211 control rods of the CPS system start to move into the core and the reactor is fully shutdown in \( \sim 14 \) s from the AZ-1 signal generation moment. During that time the negative reactivity of \( \sim 18 \beta_{\text{eff}} \) is being inserted into the core (see Fig. 12). Such a significant negative reactivity insertion into the reactor core results in the significant total reactor power decrease (see Fig. 13). In the time period of \( \sim 14 \) s reactor power decreases from \( \sim 2420 \text{ MWt} \) down to \( \sim 150 \text{ MWt} \). Later on (see Fig. 12), starting from \( \sim 70 \text{ s} \) of the transient, one can see a slight increase of reactivity. This reactivity increase is due to the power reactivity effect feedback. When the total reactor power sharply decreases, the power reactivity effect tries to eliminate the negative reactivity insertion into the core and makes the reactivity increase again, although the total reactor power at the same time continues to decrease slightly (see Fig. 13) and reaches the value of \( \sim 70 \text{ MWt} \) at time \( t=210 \text{ s} \). As could be seen from Fig. 12 and Fig. 13, reactivity and total reactor power obtained using RELAP5-3D model are in reasonable agreement with Ignalina NPP measurement data.

As it was mentioned above, the signal for the reactor shutdown (AZ-1 signal) was simulated at time \( t=0 \) s. During

Figure 13. Thermal reactor power change in time during the event

Figure 14. Water level in DS change in time during the event for both reactor core halves

Figure 15. Pressure in DS change in time during the event for both reactor core halves
the first 40 s of the transient, water level in DS decreases (see Fig. 14) because of the increase of coolant density following the reactor shutdown. Later on, water level in DS starts to increase due to the water supply from MFWP. The obvious difference between the measured plant data obtained from the Ignalina NPP and the calculation results can be explained by the fact, that DS model in RELAP5-3D code not quite accurately repeats the real geometry of the DS. The real DS is a horizontal cylinder, while in RELAP5-3D model, although DS has the same diameter and volume as the real two DS at the plant, it is simulated by three interconnected rectangular volumes.

Pressure in DS change in time during the transient is shown in Fig. 15. Some discrepancy between the pressure values in DS of the left and right MCC loops might be due to the measurement errors or due to the different power release in different MCC loops. Because during the simulation it was assumed that both MCC loops are identical, pressure in DS of the left and right MCC loops as calculated by RELAP5-3D code is the same. As could be seen from Fig. 15, during the first 20 s from the start of the transient, pressure in DS starts to decrease rapidly. This is due to the fact, that after the actuation of the AZ-1 emergency protection, reactor power rapidly decreases, but turbine reloading process (the decrease of the amount of steam supplied to the turbines) progresses rather slowly. Following the full closure of the TCV (24÷28 s after the actuation of AZ-1 signal), pressure in DS decrease rate slows down, but that does not result in the pressure increase. This means that a small amount of steam, even after the closure of TCV, is being removed from the MCC for the in-house needs.

**Reactor power reduction on AZ-3 emergency protection signal simulation benchmark**

Below one will find the RELAP5-3D calculation results, obtained during the modelling of the reactor power reduction transient on the actuation of AZ-3 emergency protection signal that took place at Ignalina NPP Unit 2 on August 22, 2000 [7]. Following the closure of the turbine control valve of one turbo generator at time t=0 s, steam flowrate from DS in both reactor core halves decreases very rapidly (see Fig. 16). From this time point steam is being supplied only to a single turbo generator. As it was already mentioned above, the trip of one turbo generator initiated the AZ-3 emergency protection actuation at time t=0.5 s.

Following the AZ-3 signal, all power setpoint devices of the 12 LAR/LEP zones started to decrease their power setpoints, according to which LAR/LEP detectors are generating their power excess signals. Due to the power setpoints decrease, LAR detectors in all 12 LAR/LEP zones started to generate excess power signals and LAR rods started to insert into the reactor core to decrease the total reactor power to 50% of the nominal power. Due to the negative reactivity insertion, in ~16 s the total reactor power was reduced from its initial power level...
of 3964 MWt down to ~2400 MWt. During that time ~0.2 $\beta_{\text{eff}}$ negative reactivity is being inserted into the core (see Fig. 17). Due to the peculiarities of the RELAP5-3D model of RBMK-1500 reactor, the total reactor power still kept decreasing for about 10 s and reached its value of ~2220 MWt (see Fig. 18). Then, due to the power effect, it started to increase again and reached the value of ~2440 MWt at time $t=76$ s. Reactivity due to the power effect started to increase even earlier. This inconsistency with the plant data could be explained by the model sensitivity to the control rods movements in the core and the active feedback effects. Finally the total reactor power reached its new steady-state value of ~2260 MWt at time $t=200$ s. This is only slightly lower than the plant value for the total reactor power equal to ~2330 MWt. As could be seen from Fig. 17 and Fig. 18, reactivity and total reactor power obtained using RELAP5-3D model are in reasonable agreement with Ignalina NPP measurement data.

Due to a sudden trip of one turbine, balance between the steam generation in the reactor core and the steam flowrate from the DS of both reactor core halves is being disturbed. This results in a sharp increase of the pressure in DS (see Fig. 19). At the same time SDV-A and SDV-C valves of both reactor core halves open and the excess steam is being discharged to the turbine condensers and condensation plates of the ACS towers. Due to the decrease of the reactor power...
(after the actuation of AZ-3 emergency protection), the amount of steam, generated in the reactor core, decreases and the pressure in DS decreases as well. After the time period of 30-40 s following the actuation of AZ-3 emergency protection, pressure in DS decreases to such a value that SDV-A and SDV-C valves start to close. The above mentioned steam relief valves completely closes only in ~55 s after the generation of AZ-3 signal. Later on pressure in DS is being controlled by changing the steam flowrate supplied to the operating turbine. This is performed by the help of the pressure regulator.

Looking at the calculation results of the four benchmarks (void and power reactivity coefficients measurement simulation, as well as reactor power reduction on AZ-1 and AZ-3 emergency protection signals simulation) presented above, one can see that most of the compared parameters are in reasonable agreement between the calculated parameter values with RELAP5-3D code and the measured plant data. In general, physical processes occurring in the MCC and simulated using RELAP5-3D model correspond reasonably well to the real processes occurring in the primary circuit of RBMK-1500 reactor. Slight differences could be seen in the reactivity and total reactor thermal power behaviour in time during the transient as calculated by RELAP5-3D code and compared to the measured plant data, but in general, RELAP5-3D code predicts reactivity and the total reactor core power behaviour during the transient in a reasonable manner which confirms the correct modelling of the neutronic processes taking place in RBMK-1500 reactor core.

Conclusions

A 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 RELAP5-3D model of the Ignalina NPP RBMK-1500 reactor and reported here are: the simulation of the measurements of void and fast power reactivity coefficients, change of graphite cooling conditions and reactor power reduction transients registered at Ignalina NPP. The simulated benchmarks were modelled using RELAP5-3D code and the calculation results compared to the real plant data registered by the TITAN information computer system at Ignalina NPP. In general, the calculation results obtained using RELAP5-3D code are in reasonable agreement with the measured plant data. RELAP5-3D code predicts thermal-hydraulic and neutron-physical processes taking place in the reactor during the analysed transients in a reasonable manner and could be used further for the analysis of transients taking place during RBMK-1500 reactor operation where the neutronic response of the core is important.

Nomenclature

<table>
<thead>
<tr>
<th>2AC</th>
<th>Automatic Controller (2nd group)</th>
<th>LEP</th>
<th>Local Emergency Protection</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACS</td>
<td>Accident Confinement System</td>
<td>MCC</td>
<td>Main Circulation Circuit</td>
</tr>
<tr>
<td>AFWP</td>
<td>Auxiliary Feedwater Pump</td>
<td>MCP</td>
<td>Main Circulation Pump</td>
</tr>
<tr>
<td>ANL</td>
<td>Argonne National Laboratory</td>
<td>MFWP</td>
<td>Main Feedwater Pump</td>
</tr>
<tr>
<td>CPS</td>
<td>Control and Protection System</td>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
<tr>
<td>DS</td>
<td>Drum Separator</td>
<td>ORM</td>
<td>Operative Reactivity Margin</td>
</tr>
<tr>
<td>FC</td>
<td>Fuel Channel</td>
<td>PH</td>
<td>Pressure Header</td>
</tr>
<tr>
<td>GDH</td>
<td>Group Distribution Header</td>
<td>RBMK</td>
<td>Large Channel Type Water Cooled Graphite Moderated Reactor</td>
</tr>
<tr>
<td>ICS</td>
<td>Information Computer System</td>
<td>RRC “KI”</td>
<td>Russian Research Centre “Kurchatov Institute”</td>
</tr>
<tr>
<td>INEEL</td>
<td>Idaho National Engineering and</td>
<td>SDV-A</td>
<td>Safety Discharge Valve (to ACS towers)</td>
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</tbody>
</table>
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References