Application of the Best–Estimate Approach for the NPP Licensing Process

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Abstract

According the international practice the best-estimate approach in safety analysis of Nuclear Power Plants (NPPs) is used mainly for the Loss of Coolant Accident (LOCA) in reactor cooling circuit type accidents. In Lithuania the best estimate approach was successfully applied not only for LOCA but also for reactor transients, reactivity initiated accidents and accident confinement system response analyses. This paper presents the four examples of best-estimate accident analysis developed for Ignalina NPP licensing, covering the LOCA, transients, reactivity initiated accidents and the accidents in the confinement system. In the paper the so called "partially-conservative" approach, also used for the Ignalina NPP licensing, is introduced. The comparison of both approaches are performed and the recommendations for employment of approaches presented.

1. Introduction

Deterministic safety analyses present the most part in the scope of works performed for nuclear power plants licensing activities. This analysis is performed through the calculation of plant parameters (responses) with complex computer codes, solving a set of mathematical equations describing a physical model of the plant. Historically, a conservative approach has been taken for licensing analysis, including making conservative assumptions on plant data, system performance and system availability. The conservative approach means that use of conservative codes is combined with conservative boundary and initial plant conditions. This approach gives the results deliberately biased in a pessimistic manner. Now almost in all countries the best-estimate approach is widely used in order to avoid the unnecessary conservatism and to properly assess and to address the existing safety margins. In this approach the use of best-estimate codes is combined with realistic boundary and initial plant conditions. Together with use of best-estimate code the uncertainty analysis is required.

In Lithuania the best-estimate approach was successfully applied in licensing practices of Lithuanians only nuclear power plant – Ignalina NPP starting 2003. Two units of RBMK-1500 reactors were built in Lithuania, in Ignalina NPP. RBMK (Russian abbreviation for: "Large-power channel-type reactor") belongs to a class of graphite-moderated nuclear power reactors and were designed in the Soviet Union in 1970s. At present both reactors at Ignalina NPP are shutdown (first unit was shutdown in 2004, second – in 2009). Recently there are no plans to build new RBMK type reactors, however there are 11 RBMK reactors operating in Russia (4 reactors in Saint Petersburg, 3 – in Smolensk and 4 – in Kursk). The main findings of this paper may be applied for the RBMK-1000 reactors, operating in Russia (because of the similar design of the reactors).
In this paper are presented four examples of best-estimate accident analysis performed for Ignalina NPP with RBMK-1500 reactors licensing:

- Limiting Reactivity Initiated Accident (RIA) for RBMK-1500;
- Break of fuel channels inside reactor cavity;
- Large LOCA case in main circulation circuit (break of main circulation pump pressure header);
- Blocking of coolant flow rate in group of fuel channels (blocking of coolant flow rate in group distribution header).

These selected bounding cases with its consequences covers all possible RIAs, transients, LOCA type accidents and the accidents in reactor cavity. The RBMK-type reactors do not have the continuous containment, which covers all reactor and cooling circuit. There are Accident Localisation System (ALS), which covers the part of reactor cooling system equipment (main circulation pumps, headers and water supply pipelines) and the Reactor Cavity (RC), which enclose the graphite stack with fuel channel. Both these leak tight, complex geometry systems plays a role of containment and are the last barrier preventing the radioactivity release into atmosphere. For the analysis the different best estimate computer codes were used:

- QUABOX/CUBBOX-HYCA code [1], [2] – for the calculation of power distribution in fuel assembly;
- RELAP5 Mod3.2 [4], [5] – for the analysis of reactor cooling circuit response;

2. Models for the simulation of processes in RBMK-1500

The RBMK reactor is a channel-type graphite-moderated boiling water reactor [9]. It has a huge graphite block structure as the moderator that slows down the neutrons produced by fission. The feature of RBMK type reactor is that each fuel assembly is positioned in its own vertical Fuel Channel (FC). The water is supplied to fuel channels bottom, where it is heated to saturation and partially evaporates. The steam produced passes to the steam separators, which separates water from the steam. The fuel channels are made of Zirconium and Niobium alloy similar to that used for the fuel claddings. The graphite structure is contained in a steel vessel, which is called reactor cavity.

2.1. Simulation of power distribution in reactor core and processes in fuel matrix

For the modelling of processes in RBMK –1500 reactor core and the analyses of reactivity initiated accidents the QUABOX/CUBBOX-HYCA (Q/C-H) code was used in Lithuanian Energy Institute (LEI). The core model Q/C-H was originally developed by GRS for core calculations of Light Water Reactors (LWRs) [1], [2]. Since 1990 the code was adapted to the features of RBMK-1000 reactors and since 1995 additionally adapted to account for the special requirements of RBMK-1500 reactors. During this time the code was continuously used for RBMK-1500 core calculations and for the surveillance of the reactivity behaviour of the changing core loading using higher enriched uranium-erbium mixed fuel and Control Rods (CR) of new design [10], [11]. The code was used for audit calculations during the review of the Ignalina NPP Safety Analysis Report (SAR) for Unit 1 and for the preparation of the SAR for Unit 2. Later on, the Q/C-H code was applied for the independent assessment of the shutdown systems modification at Unit 2 [10], [11].

Design of fuel pellets and separate fuel rods for RBMK reactor differs very little from fuel rods manufactured for standard BWR-type reactors [9]. In RBMK reactor the fuel assembly is fit into a circular fuel channel with inside diameter of 80 mm and an active core height of 7 m. In order to achieve the required height, the RBMK fuel assembly consists of two fuel bundles placed one above another. Each fuel bundle includes 18 fuel rods placed in two circles around the carrying rod. For the modelling of processes in the fuel rod of RBMK-1500 the FEMAXI-6 code [3] is used in LEI. FEMAXI-6 code can analyse the integral behaviour of the whole fuel rod throughout its length.

Figure 1: Model of RBMK-1500 fuel rod (bottom bundle), developed using FEMAXI-6 code
life as well as the localized behaviour of a small part of fuel rod (temperature distribution in the fuel rod, thermally induced deformation of fuel pellet and cladding, fission product gas release). This code, designed for vessel type reactors, was adapted for modelling of processes in fuel rods of RBMK-1500 reactor, introducing the material properties for the RBMK fuel and cladding, [12], [13]. The RBMK-1500 fuel rod model, developed using FEMAXI-6 code, is presented in Figure 1. The single fuel rod from bottom fuel bundle in fuel assembly was modelled, because the generation of power peak is met in the position 1.07 m from the bottom of the core [9].

2.2. Simulation of reactor cooling circuit response

Reactor Cooling System (RCS) of RBMK has two loops, which are interconnected via the steam lines and do not have a connection on the water part. The best estimate system thermal-hydraulic code RELAP5 [4] has been adapted to model the RBMK type reactors and used since 1993 at the Lithuanian Energy Institute [22], [23]. The RELAP5 nodalization scheme, which was used for the modelling of processes in RCS, is presented in Figure 2.

The left loop of RCS model consists of one equivalent core pass. Two Drum Separators (DS) are modelled as one “branch” type element (1). All down comers are represented by a single equivalent pipe (2), further subdivided into a number of control volumes. The Main Circulation Pumps (MCPs) suction header (3) and the pump pressure header (8) are represented as branch objects. Three operating MCPs are represented by one equivalent element (5) with check and throttling-regulating valves. The stand-by MCP is not modelled. All 830 fuel channels of this left core pass are represented by an equivalent channel (12) operating at average power and coolant flow. In the right circulation loop, the
MCP system model consists of two equivalent core passes. One core pass represents one Group Distribution Header (GDH). Fuel channels from this GDH are represented by few equivalent channels of different power levels. For the core power of 4200 MWth, the channel average power is assumed to be 2.53 MWth, the maximum channel power is 3.75 MWth, and minimum channel power is 0.88 MWth. The other core pass represents the other 19 GDHs. The channels of this pass are simulated by an equivalent FC of average power. The steam separated in the separators is directed to turbines via steam pipes (15). Two turbine control valves divert steam supply to the turbines. The guillotine break of MCP pressure header (17) in the right loop model of RCS is modelled by a valve (18). The flow area of this valve is double of pressure header flow area. The valve (18) is connected to the volume (19), which represents the compartments covering RCS pipelines. The detailed description of this model is presented in papers [8], [14].

2.3. Simulation of reactor cavity response

The reactor and a large part of the reactor cooling circuit of RBMK-1500 are enclosed within the accident localization system, which consists of a number of interconnected compartments with 10 condensing pools to condensate the steam, discharged during the accident [9]. In this respect, the ALS may be called a pressure suppression type confinement. The reactor cavity in RBMK-type reactors is the compartment surrounding reactor core (fuel channels inside a graphite stack). The RC is protected against overpressure by the reactor cavity venting system, which directs the released coolant from the broken fuel channels to ALS. The RC and ALS are very important systems of RBMK reactors as they perform the containment function, i.e. form the last barrier preventing radioactive material release to the environment. The nodalization scheme of ALS and Reactor Cavity Venting System (RCVS) is presented in Figure 3 and general description of ALS and RC – in [15].

The model, developed by employing CONTAIN code [5], consists of 35 nodes and 178 structures for heat exchange modelling. Nodes 32 and 33 simulate RCVS pipelines from the upper part of RC. The four lower condensing trays of each ALS tower were connected to separate nodes 5 and 14. In the case of fuel channels rupture, the coolant from the RC is directed only to a half of the 5th condensing tray of left ALS tower. This is why the 5th condensing tray of the left tower of ALS is divided into two parts and simulated as two pools (Figure 3). The steam to the pool 5th in node 21 is provided from the RC through 2 steam distribution devices (node 20) as well as from the top steam reception chamber through 8 steam distribution devices (node 19). The steam to the pool in node 27 is provided from the top steam reception chamber through 10 steam distribution devices (node 26). These two pools are interconnected at water part by the pipe of 100 mm diameter. The pools located in nodes 6 and 15 represent the water in hot condensate chambers. The initial level in these pools assumed to be 2.5 m (according to the design). The overflow of water from Hot Condensate Chamber (HCC) to nodes 3 and 12 (representing bottom steam reception chamber) is considered in the model. The air release from the towers of ALS (from nodes 8 and 17) to the environment is simulated employing special junctions that close in 5 min after the accident starts. The blow-out panels (nodes 7 and 16), which open if excess pressure in gas delay chambers (nodes 6 – 8 and 15 – 17) increases to 80 kPa, are installed in compartments in both towers of ALS. An assumption is made in the model that six Membrane Safety Devices (MSD) (nodes 9 and 33) open when the excess pressure in RCVS pipelines before MSD increases to 60 ± 10 kPa (i.e. in accordance with MSD design data). The drainage
water removal from RC as well as from the top and bottom RCVS pipelines is taken into account. The arrows on the nodalization scheme in Figure 3 indicate the places where the structural leaks are simulated (break of fuel channels inside the reactor cavity).

3. Results of the Best-Estimate Analysis

3.1. Analysis of the limiting Reactivity Initiated Accident

Modelling of reactivity-initiated accidents involves the simultaneous solution of equations for neutron transport, heat transport within the fuel rods and across the clad-to-coolant interface, mechanical behaviour of fuel and cladding, and coolant thermal-hydraulics. These equations are strongly interconnected and dependent on both space and time. Since it was no possibilities to solve it in full detail in core-wide analyses, the separate codes and simplifications were used: the power distribution in fuel assembly was calculated using Q/C-H code, while the processes in fuel matrix of RBMK-1500 were evaluated using FEMAXI-6 code.

All possible RIAs were analysed during SAR preparation: (1) single CR withdrawal; (2) spurious withdrawal of CRs group; (3) drop out of CR with shorted absorber; (4) erroneous refuelling; (5) loss of water in the channels, where control rods are placed [19]. The accidents related to spurious withdrawal of single Control Rod (CR) were selected for more detailed investigation, because these events lead to more significant change of the core parameters. For the maximum permissible thermal reactor power (4200 MW), the withdrawal of the rod in the centre of core leads to high peak of linear power for fuel rod. In the base calculation the peak value of 530 W/cm was reached, that exceeds the acceptance criterion (485 W/cm) [17]. For this case uncertainty and sensitivity analysis were performed using GRS methodology. For the evaluating possible uncertainties of the calculation results a total number of 16 uncertain input parameters were taken into account defining model options (heat transfer parameters in fuel matrix, radial zones in fuel pin, etc.) and physical boundary conditions (CRs insertion depth, reactor power, etc.). The analysis of the calculation results allows identify the main contributors to the uncertainty of investigated output parameters as well as to determine their two-sided (95%, 95%) statistical tolerance intervals. Based on gained results it was confirmed that during the single CR withdrawal accident the maximal temperatures of fuel pellet and cladding were 1865 °C and 375 °C respectively and did not exceed the safety limits. The peak of linear power in the maximum loaded fuel channel reaches 589 W/cm (Figure 4). The linear power exceeds the acceptance criterion (485 W/cm) in 25 fuel channels, located around the failed control rod. The violation of safety limit continues ~20 s. The power in maximal loaded FC exceeds it’s the highest values in 14th second of the accident. The power peak during transient always was located in the bottom part of the core.

The maximum bounding linear power from Figure 4 was assumed as power behaviour in selected fuel assembly for the analysis of process in fuel rod. The FEMAXI model of RBMK-1500 fuel rod, presented in Figure 1 was used for the simulation. The temperatures of fuel pellet and fuel cladding temperatures start to increase after beginning of withdrawal of the failed CR, due to increase of heat generation in the core. Later, after finishing of CR movement (8th sec) temperatures start to decrease. The peak temperature in the fuel pellet centre is 2270 °C and on the inner cladding surface 400 °C. These temperatures are below acceptance criteria (2800 °C and 700 °C accordingly) [17].

The fuel cladding outer surface temperature remains constant, because the condition for boiling crisis is not reached in the fuel channel. The intensification of releases of gases (Kr and particularly Xe) from fuel into gap between pellets and cladding is insignificant during this accident, thus, the increase of pressure of gases inside fuel rod is caused due to increase of pellets temperature mainly. After decreasing of neutronic linear power in the fuel rod, the pressure in the gap between fuel pellets and cladding decreases down to the initial value. Because the fuel burnup is low, the fuel grain growth, cracking of pellet, formation of fission gas bubble in fuel and following swelling of fuel pellets do not occur. The outer diameter of fuel pellets is increasing only due to radial displacement. The radial displacement of fuel cladding is insignificant, because temperature of fuel cladding remains approximately constant during the all investigated period of time.

The results of FEMAFI calculation - the change of the gap between fuel pellets and cladding is presented in Figure 5. As can see in time interval 5 - 20 s the gap between fuel pellets and cladding disappears in segments No. 2 – 9 (at ~1-3 m from fuel rod bottom). The gap closure
appears because the radial displacement of fuel cladding is much slower than displacement of fuel pellets and fuel pellet comes in contact with fuel cladding. When the pellet reaches the cladding, the fuel pellet-clad interaction phenomenon is met. However, after reactor shutdown the radial dimensions of pellets and cladding returns back to the values close to the initial conditions. The equivalent stresses of fuel cladding are presented in Figure 6, which shows that the highest equivalent stresses are in the range of 110 MPa. Peak of stress is related to fuel pellet-clad interface. According to [20], the yield stress for Zr + 1 % Nb alloy is 180 - 220 MPa for 300 °C temperature and 320 – 380 MPa for 20 °C temperature. After exceeding the yield stresses limit, the fuel cladding will be affected by plastic deformation that leads to cladding failure. In the analysed case the calculated maximal value of equivalent stress in the cladding is much lower than the yield stress. Thus, the fuel cladding remains intact. Summarizing, it could be concluded that in case of the investigated limiting reactivity initiated accident in RBMK-1500, evaluating possible uncertainties in the modelling of neutron dynamic processes in reactor core, the peak of thermal linear power in fuel assembly could exceed the acceptance criterion (485 W/cm) for short term, but the fuel rods remain intact.

3.2. Analysis of the multiple fuel channels rupture inside reactor cavity

During the analysis of the multiple FCs rupture in the reactor cavity it was assumed that the reactor is operating on the maximum permissible thermal power of 4200 MWth. In this case the steam-water mixture from the ruptured FCs is released to the reactor cavity. The energy flow (mass flow and enthalpy) from the simultaneously ruptured 9, 11 and 16 FCs were calculated using RELAP5 model of reactor cooling circuit (see Figure 2). Further these calculation results were applied as the boundary data for the analysis of thermal hydraulic parameters behaviour in RC and ALS employing CONTAIN model (see Figure 3).

According to the performed modification in reactor cavity, the RCVS must withstand 9 fuel channels rupture [15], [17], [21]. As it is shown in [15], the excess pressure 214 kPa inside reactor cavity is assumed as an acceptance criterion because it is the smallest allowed load on the reactor cavity structures. The pressure increase (depending of number of ruptured fuel channels) beyond this value can lead to the failure of RC, loss of core cooling and large radioactivity release into the atmosphere. Thus, the pressure inside RC was selected as the analysis output result for the uncertainty analysis. The uncertainty and sensitivity analysis was performed using the GRS methodology [6] together with the developed package SUSA 3.2 [7]. The parameters which may influence the uncertainty of calculation results were compiled into two main groups: initial conditions of discharged flow (pressure, temperature and mass of water) and CONTAIN code models, assumptions and correlations. The first group of parameters, which could have the impact on the simulation results are the following: (1) heat transfer coefficient in Condenser Tray Cooling System (CTCS) heat exchangers; (2) service water temperature for CTCS heat exchangers cooling; (3) water temperature in the 5th condensing tray; (4) water temperature in the 1 to 4 condensing trays; (5) water temperature in the HCC; (6) gas temperature in RCVS pipelines; (7) air temperature in ALS leak-tight compartments; (8) air temperature in ALS tower compartments; (9) heat transfer area in the graphite stack; (10) heat transfer area of FC; (11) MSD opening pressure; (12) volume of the accident zone; (13) the connection area between the volume inside a graphite stack and the bottom (top) part of RC; (14) water level in the lower (1 to 4) condensing trays in the left ALS tower; (15) water level in the lower (1 to 4) condensing trays in the right ALS tower; (16) water level in HCC; (17) water level in the 5th condensing tray in the left ALS tower. The second group of parameters

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Figure 5: Variation of fuel-clad gap during limiting RIA

![Figure 5](image5.png)

Figure 6: Change of equivalent stresses in cladding during limiting RIA (at 1.07 m from core bottom)

![Figure 6](image6.png)
(CONTAIN code parameters) contains the following: (1) model of water deposition in the reactor cavity (i.e. in one case it is assumed that water discharged in the reactor cavity filled by helium-nitrogen mixture is deposited to the sump, and in the other case that water is transported between the nodes by the gases flow); (2) interaction of water with hot surfaces located in RC.

The uncertainty analysis with one-sided tolerance limits (59 runs) was performed for each of the selected cases - 9, 11 and 16 FCs ruptures. The calculated pressures in RC in the case of multiple fuel channel ruptures are presented in Figure 7. Figure 7 a) and b) shows that the maximum pressures of all calculated variants for the rupture of 9 and 11 FCs do not exceed the criterion of RC integrity. It is also shown in Figure 7 that calculated curves of pressure in RC form three groups. The maximum pressures were received assuming, that all the liquid fraction of discharged through the ruptured FCs coolant remains in RC in a dispersed condition and interact with hot surfaces of steel and graphite in RC. This case is not expected in reality. The bottom group of curves represents pressures calculated considering the models for water deposition from the atmosphere, but without consideration of direct interaction of this water with hot surfaces in RC. The results of the middle group of curves (Figure 7 b), which represents the greatest part of the calculated cases, reflect the most probable behaviour of the pressure in RC.

The performed sensitivity analysis demonstrated that the heat transfer area of metal, model of water deposition from the atmosphere and model of water interaction with hot surfaces located in RC have the largest influence on the pressure in RC. The uncertainty analysis showed that in the case of 11 FCs rupture the limiting pressure was not exceeded even in the most unfavourable combinations of the initial conditions and modelling parameters (see the top curve in Figure 7 b). The extrapolation of the bottom curves of the calculated pressures for the rupture of 9, 11 and 16 FCs (Figure 7) shows that maximal number of ruptured FCs, when limiting pressure is not exceeded, is 19 FCs. Thus, summarizing the results of the uncertainty analysis, it is possible to conclude that the capacity of RCVS comprises from 11 up to 19 FCs, i.e. 15±4 FCs.

3.3. Analysis of large LOCA in RBMK-1500

Considering the consequences of all LOCA type accidents and taking into account the rupture size and peak temperature of the fuel cladding [16], it was found that the worst consequences for RBMK-1500 are in the case of MCP pressure header (the largest diameter pipeline) break with a failure to close the check valve of one GDH at maximum permissible reactor power level of 4200 MWth. The analysis of such large LOCA was performed according to the “best-estimate” approach. Thermal hydraulic analysis was performed by employing the best estimate thermal-hydraulic RELAP5 Mod3.2 code Ignalina NPP model (see Figure 2). In order to identify the main contributors to the uncertainty of the calculation results, the GRS methodology [6] together with the developed package SUSAN 3.2 was used [7]. More details about the application of uncertainty and sensitivity analysis methodology for the best-estimate analysis of RBMK are presented in [8]. The parameters, which may impact the uncertainty of calculation results, can be divided into two main groups:

• Initial and boundary conditions with values that may be impacted by measurement errors;
• RELAP5 code models, assumptions and correlations.

The six following plant parameters, which may have the greatest impact on the simulation results due to knowledge from earlier performed benchmarking calculations, are selected for the analysis: (1) pressure in the DS; (2) coolant flow rate through the MCPs; (3) feed water temperature; (4) amount of steam for in-house needs; (5) reactor thermal power; (6) delay time for reactor scram initiation. The deviation values for these parameters are known from the Ignalina NPP documentation: they vary from 1.5 to 2%. Additionally, the 8 RELAP5 code parameters and models, such as water packing scheme, vertical stratification model, counter current flow limit model, thermal front tracking model, mixture level tracking model and others were selected for the analysis. It was assumed that the selected RELAP5 code parameters vary in the area where mainly two-phase flow conditions might occur: in the vertical heated channels, above the heated channels, steam-water pipes modelling elements and break location. Other areas (especially with single-phase conditions) are excluded due to the fact that these parameters do not have impact on the results in such regions.

Straight after the MCP pressure header break, the water from MCP piping and DSs is discharged through the break (18, see Figure 2). The reactor emergency scram system is activated due to the pressure increase in the compartments where the coolant is discharged. GDH check valves (16), which are located downstream to the break, close and prevent the loss of coolant from the fuel channels of the affected RCS loop. The coolant flow stops in the affected RCS loop. However, within approximately two seconds cold water is supplied from the Emergency Core Cooling System (ECCS) fast acting subsystem to these channels. This enables supplying reliable cooling of these channels. The coolant flow in the FCs, connected to the GDH with the check valve which failed to close (16) becomes stagnant for a moment and later changes its flow direction. Due to the coolant flows behaviour, the fuel cladding temperature (in channels connected to GDH with the check valve which failed to close) increases more than in the other channels of the affected RCS loop within the first seconds of the accident. When the saturated water from DS reaches the overheated fuel assemblies, fuel cladding temperature drops down. At the beginning of the accident, channels connected to the GDH with the check valve which failed to close are cooled by saturated water flow; however, later (after DS gets empty) they are cooled only by saturated steam. It should be noted that the first fuel cladding temperature increase asserts only at the very beginning of the accident and takes a very short time, i.e. no more than 15 seconds. Later, all fuel assemblies are reliably cooled by water, supplied from ECCS. The results of the analysis [16] shows, that the peak cladding temperature in fuel channels with 3.0 MW power is close to the acceptance criterion for fuel cladding (700 °C) [17]. Therefore, this code output quantity was selected for the uncertainty analysis. The aim of the analysis was to evaluate the number of channels with affected fuel rods. Due to the fact that for the selected case only the upper limit of technological parameter is important, only one-sided tolerance limit is used in the uncertainty analysis. For the uncertainty analysis and according to Wilk’s formula, one-sided tolerance limit (with 0.95 of probability and 0.95 confidence) requires at least 59 runs to be performed [6], [18]. The behaviour of the calculated fuel cladding temperature in 3.0 MW power FC for all 59 calculation runs is presented in Figure 8. As it is shown in this figure, the fuel cladding temperatures band does not exceed the acceptance criterion of 700 °C. Therefore, while evaluating possible uncertainties of calculation, acceptance criterion is exceeded in the fuel channels with power higher than 3.0 MW and, thus, fuel cladding integrity in these FCs can be violated. For the evaluation...
of the number of affected fuel channels, the real distribution of FCs power in the GDH taken from Ignalina NPP data was employed in this paper. Figure 8 shows a histogram of the reference channel power distribution in one GDH at the maximum permissible thermal operating power (i.e., 4200 MW). As it may be seen in Figure 8, there is a group of 12 fuel channels with power exceeding 3.0 MW; therefore, the integrity of fuel claddings in remaining 31 FCs will not be violated with 95% of probability and 95% of confidence. This information about the number of FCs (with possibly affected fuel claddings) was further used in the analysis of radiological consequences. Since the number of possibly affected fuel rods is small, this does not have any considerable impact to the radiological consequences.

3.4. Analysis of coolant flow rate blockage in group of fuel channels

The blocking of coolant flow rate in group distribution header leads to the considerable coolant flow rate decrease in a group of 38-42 fuel channels connected to the affected GDH. Previous analyses showed that the consequences of this event are the worst (the highest fuel cladding and fuel channel wall temperatures are reached) in comparison to other transients. During the analysis of coolant flow rate blockage it was assumed that up to the beginning of the initiating process (before GDH blockage), the reactor operates at the power of 2900 MWth. The coolant is supplied through the core by two MCPs in each RCS loop. This reactor state was selected because in such conditions reactor cooling of the core is the most complicated. 2900 MWth is the maximum allowable power level when four MCPs in both circulation loops are in operation, i.e. the worst power and coolant flow rate ratio is selected. During such type of events this fact has high effect on the results. In calculations it is assumed that the manual valve (see (3) in Figure 10) in pipeline connecting MCP pressure header (2) and one GDH (6) is closed by mistake. It results in the drop of the pressure in the failed GDH (6). Under the influence of the pressure difference between MCP pressure header and failed GDH, the coolant from a MCP pressure header (2) starts to flow through bypass line (4) into the ECCS header (5) and further is directed into the failed GDH (6). Thus, 38 – 43 fuel channels, connected to this GDH are cooled only by water supplied through this ECCS bypass line (4). Due to this fact the coolant flow rate through affected FCs decreases. The decreased coolant flow rate removes less amount of heat from the fuel assemblies, thus, this leads to the critical heat flux in these fuel channels. Fuel cladding and channel wall temperatures rapidly increase before the reactor is shutdown due to the coolant flow rate decrease in one GDH. After reactor shutdown energy generation in the affected fuel channels decreases, fuel channel cooling mode from the post-CHF returns back to the bubbly regime and fuel cladding and channel wall temperatures decrease.

For the present transient analysis the same parameters (initial and boundary plant conditions, RELAP5 code models, assumptions and correlations) as for the analysis of large LOCA were selected. The uncertainty analysis with one-sided tolerance limits (according to Wilk’s formula 59 calculations are required) was performed. The analysis results - behaviour of the most important RELAP5 output result - fuel cladding
temperature in the location 3.75 m from the core bottom – for all 59 calculation runs is presented in Figure 11. As it is seen from the presented figure the acceptance criterion (temperature of 700 °C for fuel cladding) is not exceeded in any FC with 95% of probability and 95% of confidence.

4. Comparison of “Best-Estimate” and “Partially-Conservative” Calculation Results

The uncertainty and sensitivity analysis by employing the GRS methodology [6], [7] requires a certain number of calculations to perform. The use of “partially-conservative” approach leads to minimization the number of calculations. In this case the best-estimate code RELAP5 with conservative boundary and initial plant conditions were employed. The conservative initial conditions are assumed as the worst possible initial conditions or the conditions increased (or decreased – depending on what value results in more conservatism) by possible measurement errors. For the MCP pressure header rupture with failure to close check valve of one GDH analysis at maximum permissible thermal reactor power level of 4200 MW the conservative initial conditions were assumed:

- The pressure in DS is equal to 6.95 MPa. It is the maximum possible pressure in the DS. This pressure is bounded by activation of equipment, protected the RCS from the overpressure (the lowest set point of activation of this equipment is equal 6.96 MPa).
- The coolant flow rate through each MCP is assumed equal 6500 m³/h. This coolant flow rate is minimum possible and it is limited by the position of throttling regulating valves.
- The feed water temperature is assumed pessimistically high and equal 195 °C. This value is equal to the maximum possible temperature of feed water 190 °C, taking into account 1 % of measurement error.
- The reactor thermal power is assumed equal to the maximum allowable reactor thermal power level 4200 MW increased by 1.06 times (3 % of measurement error and plus 3 % due to the first active control system interaction).

For the “partially-conservative” calculations, the RELAP5 code modelling parameters, which had been recommended by user manuals and were established during the RELAP5 model validation procedure [14], are used.

The comparison of “partially-conservative” calculation and upper boundary of “best-estimate” results (with realistic boundary & initial conditions plus uncertainty and sensitivity analysis) for MCP pressure header rupture case is presented in the Figure 12. In the “partially-conservative” calculation the maximum temperatures is 10 – 15 °C higher as the upper boundary of results using “best-estimate” approach with uncertainty and sensitivity evaluation.

The comparison of “partially-conservative” calculation and upper boundary of “best-estimate” results (with realistic boundary & initial conditions plus uncertainty and sensitivity analysis) for GDH blockage event is presented in Figure 13. The results of “best-estimate” approach shows that none of acceptance criterion are exceeded. As it is seen from the figure, the “partially-conservative” values of peak cladding temperatures are approximately 6 – 8 centigrade higher and the acceptance criterion for fuel cladding temperature is reached.
The comparison of results of “partially-conservative” calculation and “best-estimate” calculation with uncertainty and sensitivity analysis enables to draw a conclusion that in most cases both approaches either “best-estimate” or “partially-conservative” can be applied for the accident analysis. The latter approach looks tempting, since in this case only one calculation is sufficient; while in the case of “best-estimate” approach at least 59 calculations are required. Thus, “partially-conservative” approach requires considerably less computational time.

These two methods (“best-estimate” and “partially-conservative”) can be compared by comparing the margin to the acceptance criterion (see Figure 14). The results, calculated using “partially-conservative” method should be more conservative in comparison with the results of “best-estimate” approach (using realistic boundary & initial conditions plus uncertainty and sensitivity analysis). Analysed accident situations consequences can be acceptable if calculated parameters’ values are below the acceptance criteria. Thus, if the results obtained using “partially-conservative” method do not meet acceptance criteria (as it was in the GDH blockage case), the complete analysis by employing “best-estimate” approach is necessary.

5. Conclusions

The analyses presented in this paper were performed for the licensing needs of RBMK-1500 reactors. The performed analyses demonstrated a wide range of application of the best-estimate methodology with uncertainty and sensitivity analyses. The deterministic calculations were performed using QUABOX/CUBBOX-HYCA code (for the calculation of power distribution in fuel assembly), FEMAXI-6 code (for the modelling of processes in fuel matrix of RBMK-1500) and best estimate thermal-hydraulic code RELAP5 Mod3.2 (for analysis of reactor cooling circuit response). The integrated analyses code CONTAIN 1.2 was used for the analysis of reactor cavity response. The GRS statistical method based on propagation of input errors, together with the package SUSU was used for the uncertainty and sensitivity analyses. The presented applications demonstrated that the best-estimate methodology, used during nuclear power plants licensing activities, allows to avoid the unnecessary conservatisms and to assess and address the existing safety margins. The use of uncertainty analysis allows to assess the capacity of safety systems more accurately. This method also allows to analyse the influence of separate parameters on the calculation results, that creates the necessary conditions for further improvement of used computational models.

The performed comparison between two approaches shows that in general “partially-conservative” approach provides higher peak temperatures. This approach could be recommended for safety calculations if obtained results do not violate safety criteria. Otherwise, “best-estimate” approach would be appropriate. It is necessary to point that for the “partially-conservative” approach proper boundary and initial conditions should be selected. This selection requires experience from
user, since for different phenomena, which are observed during the accidents, initial and boundary conditions can differ.

References


