Comparison of modern analysis approaches based on the postulated GDH blockage event at Ignalina NPP

Rolandas Urbonas,
Algirdas Kaliatka,
Eugenijus Ušpuras

Lithuanian Energy Institute,
Breslaujos 3,
LT-3035, Kaunas, Lithuania
rolandas@isag.lei.lt, algis@isag.lei.lt,
eugeniju@isag.lei.lt

The paper presents an analysis of the postulated blocking of coolant flow rate in Group Distribution Header event at the Ignalina NPP. Two types of analysis – best estimate and partially-conservative – are performed. The best-estimate approach is based on the best-estimate codes with realistic boundary and initial conditions plus uncertainty analysis, while partially-conservative approach is based on the best estimate codes with conservative boundary and initial conditions plus conservative assumptions. A comparison of both calculations showed that the peak fuel cladding temperatures for best-estimate calculation, taking into account the uncertainty analysis, is slightly lower in comparison with partially-conservative calculation. The partially-conservative approach requires considerably less computational time. However, when with this methodology obtained results do not meet acceptance criteria (as it was in this particular case), a complete analysis employing the best-estimate approach is necessary.

Key words: group distribution header, thermal-hydraulic, uncertainty analysis, best estimate and partially-conservative approaches

1. INTRODUCTION

Different methodologies are used in the safety justification process of the NPP. Twenty or thirty years ago only conservative codes RELAP2, RELAP4 and conservative boundary & initial conditions plus conservative assumptions were used. Such approach is called conservative. Later, when best-estimate codes (RELAP5, ATHLET, CATHARE) were developed, they began to replace the conservative codes. In many countries the accident analysis is performed by using best estimate codes, however, the boundary & initial conditions and assumptions remain conservative. This approach is called partially-conservative.

Since the best-estimate code predictions are uncertain due to a number of uncertainty sources (code models, initial and boundary conditions, plant state, scaling and numerical solutions algorithm), at present it is agreed that the code and model uncertainty should be evaluated. The approach when in the modeling process the best-estimate code is used and uncertainty evaluation is performed is called best-estimate.

The methodologies of uncertainty analysis have been developed at Pisa University (Italy) [6–9], GRS (Germany) [3–5], NRC (USA) [10, 11], IPSN (France) [12, 13], to name a few. The main groups of uncertainty methodologies are described below:

- Uncertainty Method based on Accuracy Extrapolation (UMAE) is based on the accuracy extrapolation of the modeling of thermal hydraulic experiments towards the modeling of postulated accidents.
- Method used in Great Britain is based on definition by experts of initial uncertainties in some confidence boundaries, and the impact of these uncertainties is further investigated in terms of calculations with variations of limiting parameters.
- GRS, IPSN and ENUSA (Spain) uncertainty and sensitivity analysis methodologies are based on statistical (probabilistic) uncertainty extrapolations, when uncertainties are assumed in terms of random values with selected distributions.

A comparison of uncertainty methodologies is presented in a joint work of experts from different scientific centers [14]. It has shown that all methodologies provide similar results. However, when the methodologies themselves are compared, one can see that the methodology based on statistical (probabilistic) approach enables to evaluate more of uncertainty sources, including the user's effect, and in some cases enables to compensate the lack of measurement data or experience.

Since there is a lack of experimental data and experience in empoying the Western codes for RBMK-type reactors, the GRS method for the determination of uncertainties [4] was selected for accident analysis of the Ignalina NPP Unit 2.

The paper demonstrates a deterministic best-estimate analysis of coolant flow rate in GDH event using the best-estimate code RELAP5/MOD3 and uncertainty analysis by employing the GRS (Germany) System for Uncertainty and Sensitivity Analysis (SUSA) [3]. For that purpose, a RELAP5 Ignalina NPP model with the realistic boundary and initial conditions of RBMK-1500 was developed and the main contributors to the uncertainty of the results were identified. Since the selected uncertainty analysis methodology is a statistics-based methodology, a certain number of calculations were performed. The results showed a dependence on the initial and boundary conditions and code selected models. This modern approach applying these two types of analysis, which complement each other, is used in the Ignalina NPP Unit 2 safety analysis report.

The results obtained by this methodology are compared to ones obtained by employing the partially-conservative approach. This kind of comparison enables to evaluate both methodologies.

2. THE IGNALINA NPP RELAP5 MODEL

The RBMK-1500 is a graphite-moderated, boiling water, multichannel reactor. It is designed to provide a saturated steam at a pressure of 7.0 MPa. The design reactor power is 1500 MW_e and 4800 MW_{th}. The currently allowed thermal power is 4200 MW. The main distinguishing characteristic of an RBMK-type reactor is that each core fuel assembly is housed in an individual pressure tube [1]. The RBMK-1500 core contains 1661 fuelled channels separated from the nearest neighbors by the walls of the pressure tubes and graphite blocks. The Main Circulation Circuit is divided into two halves – left and right loops. In Fig. 1 only one loop of MCC is presented. Each loop has two drum separators (1), which separate the steam from the steam-water mixture exiting from the core block. For the cooling water forced circulation through the RBMK-1500 reactor, at the Ignalina NPP eight Main Circulation Pumps are employed. The MCPs (4) are joined in groups of four pumps each (three for normal operation and one on standby). The MCPs feed common pressure headers (5) on each side of the reactor. Each pressure header provides the coolant to 20 Group Distribution Headers (9), each of which in turn feeds from 38 to 43 fuel channels (14) (pressure tubes which contain fuel assembly inside). The flow in each pipe is set by isolation and control valves and is measured by a ball flow meter (12). The coolant is forced upwards through the reactor core block. Following through the core it acquires about 95% of the total energy emitted by the fuel elements. The steam-water mixture generated in the fuel channel flows through the steam-water pipes (15) to the Drum Separators.

The RELAP5 computer code was developed by Idaho National Engineering Laboratory [2]. It is a one-dimensional non-equilibrium two-phase thermal-hydraulic system code. The RELAP5 code has been successfully applied to PWR and BWR reactors. Since 1989 the RELAP5 model of the Ignalina NPP was used at the Lithuanian Energy Institute for analysis of the thermal-hydraulic response of the plant to various transients. The key features of the RELAP5/MOD3.2 model of the Ignalina plant are as follows:

• Both loops of the MCC are represented. Flow paths within a loop are modeled by one or more passes. In turn, a core pass model uses one or more equivalent fuel channels. The equivalent fuel channels modify heat generation in a group of real channels, as well as the hydraulic properties of this group.

• Heat structures of the equivalent FC are modelled by multiple axial and radial control volumes.

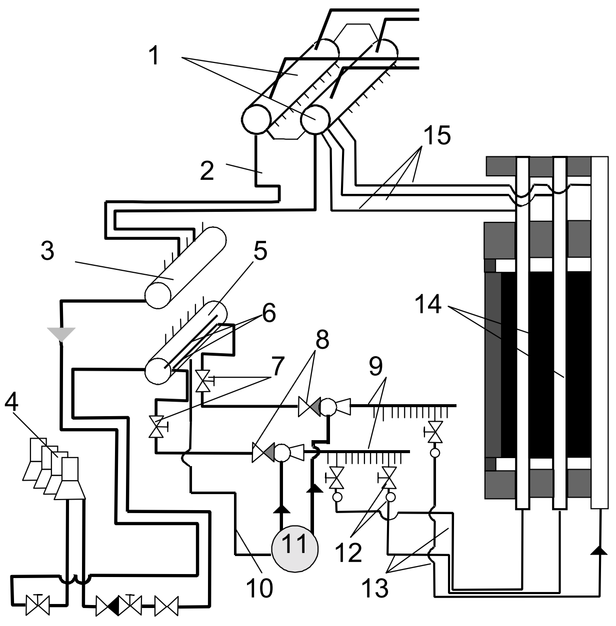

Fig. 1. Ignalina NPP circulation circuit scheme (only one half is presented): *1* – DS, *2* – downcomer, *3* – MCP suction header, *4* – MCP, *5* – MCP pressure header, *6* – filter inside pressure header, *7* – valve, *8* – check valves, *9* – GDHs, *10* – ECCS bypass line, *11* – water supply from ECCS, *12* – isolation and control valve and ball-type flow meter, *13* – lower water communication line, *14* – fuel channels, *15* – steam-water pipes

• Heat transfer among the equivalent fuel channels is approximated by means of heat exchange through the graphite moderator gaps to the reactor cavity gas circuit.

• Steam paths that remove the vapor from DSs are represented explicitly, including steam lines, steam relief valves, etc.

• The feed water system and ECCS are represented explicitly.

3. DESCRIPTION OF GDH BLOCKING EVENT

In the model it was assumed that up to the beginning of the accident process (before GDH blockage), the reactor operates at a power of 2900 MW_{th}. The coolant is supplied through the core by two MCPs in each MCC loop. This state of the reactor was selected because in such conditions the reactor cooling of the core is most complicated. 2900 MW_{th} 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 this type of accident this fact has a high effect on the results. In calculations it is also assumed that the coolant flow rate through the connection pipeline into GDH is completely stagnated within 0.01 s. The coolant flow rate stagnation results in a drop of pressure in the failed GDH. Under the influence of the pressure difference, the coolant from the MCP pressure header flows through bypass pipelines (see (10) in Fig. 1) into the ECCS header and further through ECCS pipelines is directed into the failed GDH. The 38–43 fuel channels connected to this GDH are cooled only by water supplied through the ECCS bypass pipeline. The calculations performed taking into account the throughput of the flow limiter in the ECCS mixer show that the coolant flow rate decreases (Fig. 2). The decreased coolant flow rate removes a less amount of heat from the fuel assemblies, thus leading to the critical heat flux in these fuel channels. The temperature of the fuel cladding and of the pressure tube wall sharply increases. The peak of fuel cladding temperature in the maximum power channel is close to the acceptance criterion temperature (700 °C for fuel cladding). CHF in turn initiates instabilities of the coolant flow rate in the fuel channels. The protection against coolant flow rate decrease through GDH initiates the reactor shutdown. After reactor scram energy generation in the affected fuel channels decreases, the fuel channel cooling mode from the post-CHF returns back to the bubbly mode, and this leads to a stabilization of the coolant flow rate (Fig. 2).

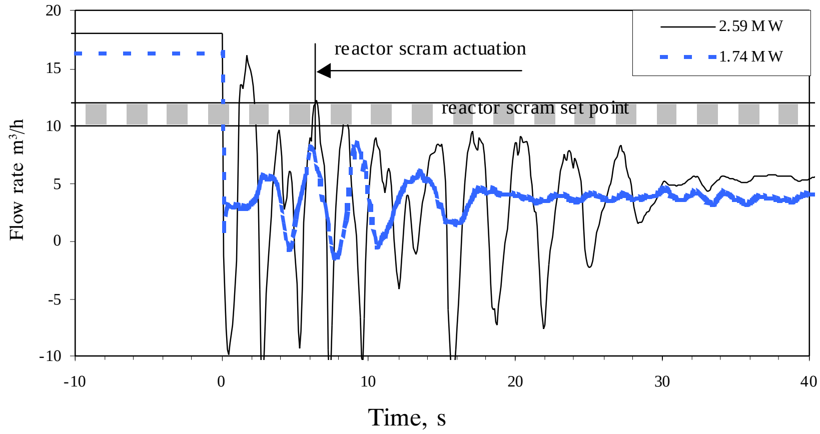

Fig. 2. Coolant flow rate through fuel channels connected to affected GDH

4. UNCERTAINTY ANALYSIS

4.1. Selection and quantification of uncertainty parameters

The parameters that may impact the calculation results, can be divided into two main groups:

1. Initial conditions (coolant pressure, temperature and flow rate or power. These values may be impacted by measurement errors).
2. RELAP5 code models, assumptions and correlations (in the RELAP5 model different correlations for the calculation of friction loss, critical heat flux and heat transfer may be used) [2].

For the present postulated accident analysis the following parameters whose initial values have the greatest impact on the simulation results are selected:

• pressure in the drum separator;
• coolant flow rate through the MCPs;
• feed water temperature;
• amount of steam for in-house needs;
• reactor thermal power.

From the Ignalina NPP documentation the deviation values ±p % are known for these parameters. In the uncertainty analysis, the maximum and minimum values of the parameters, taking into account the possible measure errors of these parameters, were determined (Table). The values *m* of the realistic initial & boundary conditions, which are between the maximum and minimum values, are used in the basic case calculation.

For the analysis, also the following RELAP5 code modeling parameters were selected:

• Water packing: it specifies if the volume filling with water scheme is to be used.

• Vertical stratification: it specifies if the model of two-phase media vertical stratification is enabled or disabled.

• Modified PV term in the equations: it specifies if the modified potential pressure energy model is applied or not.

• CCFL (counter current flow limit): it specifies if the model is applied or not.

• Thermal front tracking: it specifies if the model is enabled or disabled.

• Mixture level tracking: it specifies if the model is enabled or disabled.

Such a wide variation of parameters is assumed taking into account the possible user effects. The selected RELAP5 code parameters vary in the area where two-phase flow conditions might occur: in the vertical section before the heated channels, in the heated channels, above the heated channels, and in the steam water communications modeling elements. The areas with single-phase conditions are excluded due to the fact that these parameters have no impact on the results in this region. In the basic case of calculations some of the code models were disabled, as they had no impact on the results. However, in the uncertainty analysis none of the potential contributors to the uncertainty of the results can be excluded.

For uncertainty analysis in the case of coolant flow blocking through the GDH, additionally one parameter that might influence the fuel cladding and fuel channel wall temperatures was selected. It is reactor protection against the accident initiation signal (the time when the reactor is shut down in the BAZ regime). In the modeling the insertion delay of CPS rods was assumed by 1 second. In the uncertainty analysis the effect of one-second delay on the event consequences, *i.e.* fuel cladding and fuel channel temperatures, was analyzed.

Table. **Selection of the parameters that may impact the uncertainty of calculation results**

#	Parameter	Width of parameter distribution		Value of the parameter in the basic case calculation (m)	Mean deviation (s) and error [p%]	Probability distribution	Note
		Min. (i)	Max. (a)				
				Initial conditions			
1	Pressure in DS, Pa	$6.79 \cdot 10^6$	$6.93 \cdot 10^6$	$6.86 \cdot 10^6$	$3.43 \cdot 10^4$ [1%]	Normal	Measurement error
2	Coolant flow rate through MCPs, m^3/h	6860	7140	7000	70 [2%]	Normal	Measurement error
3	Feed water temperature, K	458.52	467.78	463.15	2.32 [1%]	Normal	Measurement error
4	Resistance of valve which regulates steam flow rate to the in-house needs header	227.7	232.3	230	1.15 [1%]	Normal	Measurement error
5	Reactor thermal power, W	$2.81 \cdot 10^9$	$2.99 \cdot 10^9$	$2.90 \cdot 10^9$	$4.5 \cdot 10^7$ [3%]	Normal	Measurement error
				Modeling parameters			
6	Water packing	0 (on)	1 (off)	1 (off)	–	Non parametric	Model assumption
7	Stratification	0 (on)	1 (off)	0 (on)	–	Non parametric	Model assumption
8	PV term	0 (off)	1	0 (off)	–	Non parametric	Model assumption
9	CCFL	0 (off)	1 (on)	0 (off)	–	Non parametric	Model assumption
10	Thermal front tracking	0 (off)	1 (on)	0 (off)	–	Non parametric	Model assumption
11	Mixture level tracking	0 (off)	1 (on)	1 (on)	–	Non parametric	Model assumption
12	BAZ initiation time	5.3	6.3	6.0	0.25	Normal	Signal delay in logic schemes

4.2. Uncertainty analysis

Before the uncertainty analysis, from the many best estimate code output quantities only a few important calculation results should be selected (usually the peak fuel cladding and pressure tube wall surface peak temperature, pressure inside the pressure tubes and drum separator pressure), which can be compared with acceptance criteria and for which uncertainties are evaluated. As mentioned above, the fuel cladding temperature peak in the maximum power channel is close to the acceptance criterion in the case of GDH blockage event. Therefore this code output result was selected for uncertainty analysis. Due to the fact that for the selected case only the upper limit technological parameters are of importance, in the analysis only one-sided tolerance limit is used.

For the uncertainty analysis for one-sided tolerance limits (with 0.95 probability and 0.95 confidence), according to Wilk's formula [4], it is necessary to perform at least 59 runs. In this case 60 runs were performed.

The behavior of the most important RELAP5 output result – the fuel cladding temperature in the location 3.75 m from the core bottom – for all 60 calculation runs is presented in Fig. 3. As is seen from the presented figures, the fuel cladding temperatures do not exceed the acceptance criterion of 700 °C. In Fig. 4 the maximum and minimum values of all calculations are compared to the basic case calculation value.

The analysis showed that the greatest impact on the calculation results has the selection of the mixture level tracking mixture and thermal front tracking model selection and

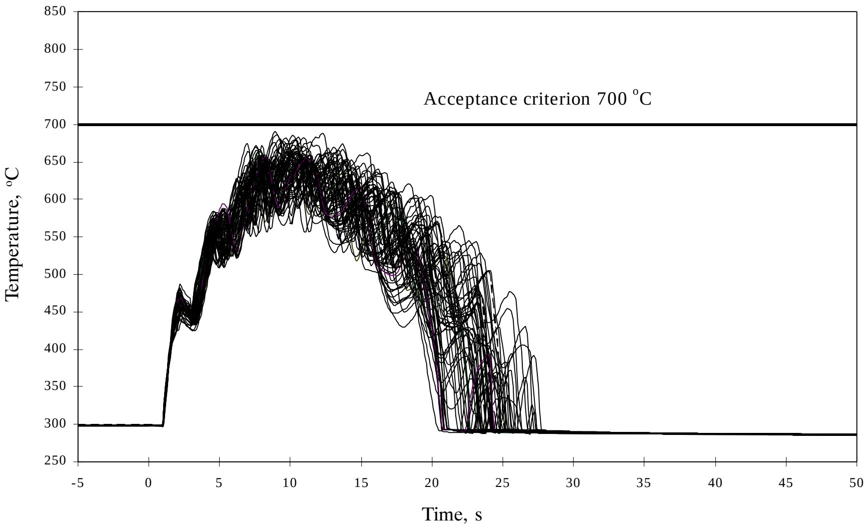

Fig. 3. Fuel cladding temperatures obtained using SUSA generated runs from RELAP5 calculations

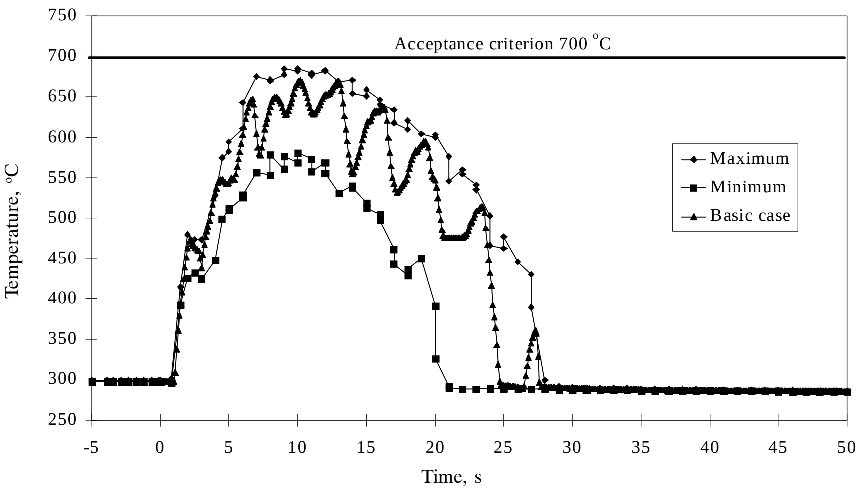

Fig. 4. Maximum, minimum and mean fuel cladding temperatures obtained using SUSA generated runs from RELAP5 calculations

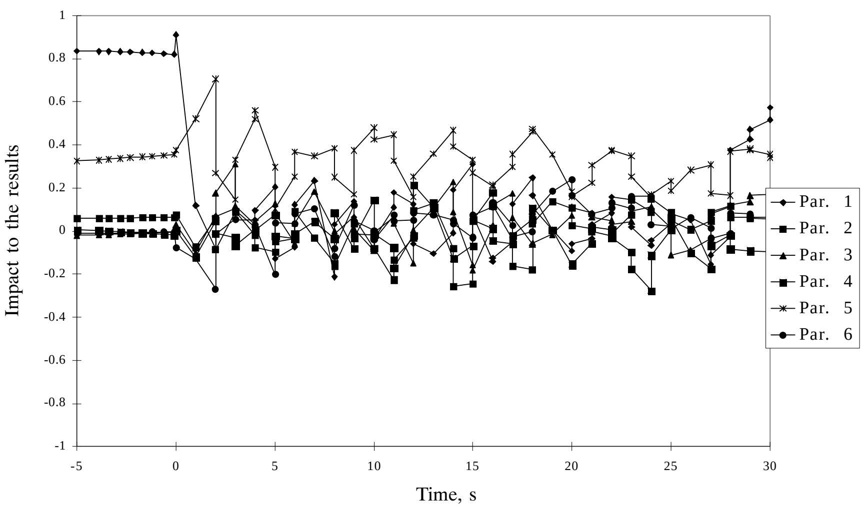

Fig. 5. Impact of the parameters 1–6 from Table on the fuel cladding temperature

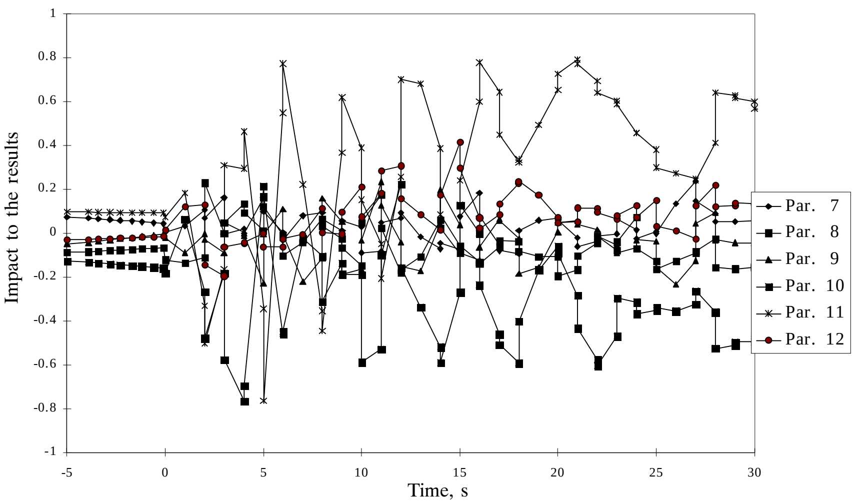

Fig. 6. Impact of the parameters 7–12 from Table on the fuel cladding temperature

reactor thermal power (see Figs. 5 and 6). Interestingly, some parameters have a positive impact on the results, while the effect of the others is negative. The meaning of the positive impact is that the higher value of the parameter is selected, the higher temperature value is obtained. In this particular case, the higher reactor thermal power and mixture level tracking is initiated, the higher fuel cladding temperature is calculated. On the other hand, when the thermal front tracking model is initiated the calculated temperature is lower, because the impact on the results in Fig. 6 of this parameter has mainly a negative value.

5. COMPARISON OF BEST ESTIMATE AND PARTIALLY-CONSERVATIVE CALCULATION RESULTS

As mentioned above, in the many countries the accident analysis is performed by using the partially-conservative approach (best estimate codes, but the boundary & initial conditions and assumptions remain conservative). The conservative initial conditions are assumed to be the worst possible initial conditions and increased (or decreased, depending on the value of more conservative results) by possible measurement errors. According to this methodology, the calculation results should be more conservative in comparison with the results obtained in the best estimate approach (using the realistic boundary & initial conditions plus uncertainty analysis). This section presents a comparison of results obtained using the best-estimate approach described earlier and a calculation obtained by employing the partially-conservative approach. These two methods can be compared by comparing the margin to the acceptance criterion. The consequences of the analyzed accident situations can be acceptable if the values of the calculated parameters are below the acceptance criteria. This comparison enables to verify the accuracy of the uncertainty calculation.

In the partially-conservative approach the initial operating conditions of the plant were set at their bounding limits (the conservative boundary & initial conditions were used):

- 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 protecting the MCC from the overpressure (the lowest set point of activation of this equipment is equal to 6.96 MPa).
- The coolant flow rate through each MCP is assumed to equal 6500 m^3/h. This coolant flow rate is minimum possible and is limited by the position of the throttling regulating valves.
- The feed water temperature is assumed to be pessimistically high and reach 467.78 K. This value is equal to the maximum possible temperature of feedwater (463.15 K), taking into account 1% of measurement error.
- The reactor thermal power is assumed to be equal to the maximum allowable power level when four MCPs are in operation ($2.90 \cdot 10^9$ W), increased 1.06 times (3% of measurement error and plus 3% due to the first active control system interaction).
- It is assumed in the modeling that the reactor scram (BAZ signal) is initiated taking into account all possible delays and occurs 6.3 s after the GDH blockage.

For the partially-conservative calculations, the RELAP5 code modeling parameters recommended by user's manuals and established during the RELAP5 model validation procedure are used. These modeling parameters are listed in Table, in the column "Value of the parameter in the basic case calculation". A comparison of partially-conservative calculation and the upper boundary of best-estimate results (with realistic boundary & initial conditions plus uncertainty analysis) is presented in Fig. 7. Only the peak fuel cladding temperatures in the maximum loaded fuel channel are presented. The results of the best-estimate approach show that no acceptance criteria are exceeded. As is seen from Fig. 7, the partially-conservative values of peak cladding temperatures are approximately 6–8 centigrade higher and the acceptance criterion for fuel cladding temperature is reached.

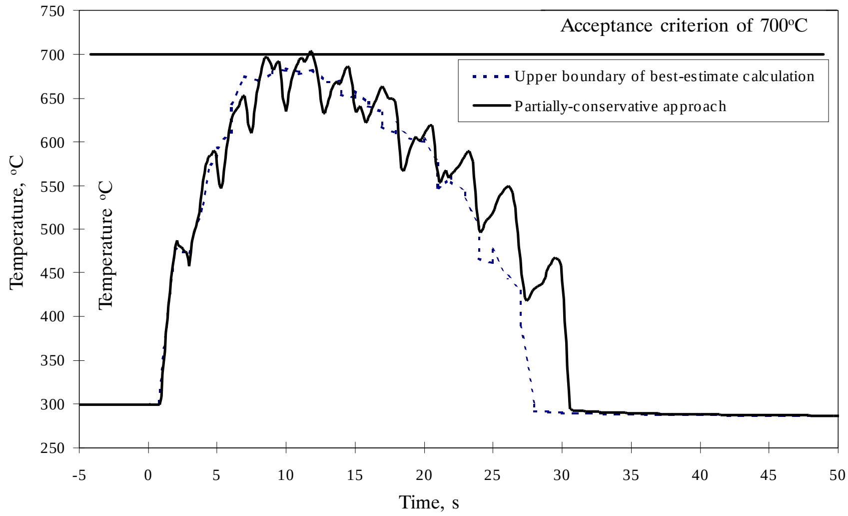

Fig. 7. Comparison of the peak fuel cladding temperature in the maximum loaded channel in case of "partially-conservative" and best-estimate calculations with uncertainty analysis

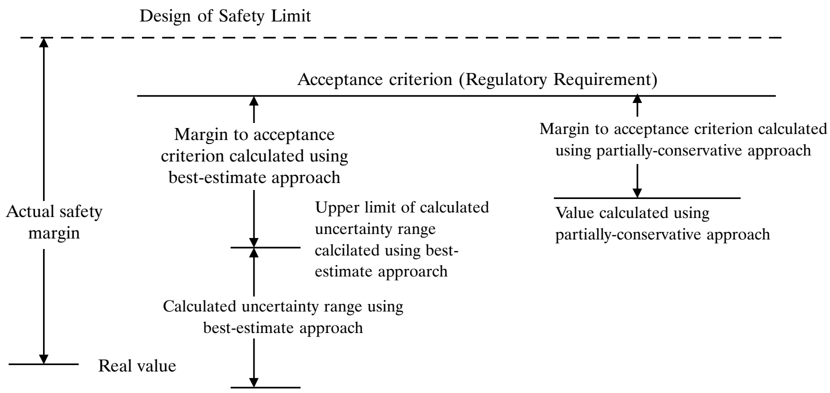

Fig. 8. Illustration of the margin to the acceptance criterion

The obtained results have shown that the parameters for best-estimate calculation taking into account the uncertainty analysis are slightly lower in comparison with partially-conservative calculation as shown in Fig. 8. It is also obvious that in this case it is not possible to use only the partially-conservative methodology, since results obtained by this methodology do not meet the acceptance criterion. With such results it is necessary to perform a complete analysis using the best-estimate approach.

6. CONCLUSIONS

The coolant flow blocking in the piping connecting MCP pressure header and GDH was analysed using the best-estimate and partially-conservative approaches. The analysis showed that the calculation results depend mostly on the selection of the mixture level tracking and thermal front tracking model and the initial thermal power of the reactor. The results of best-estimate calculation plus uncertainty analysis show that no acceptance criteria are exceeded. A comparison of both calculations shows that the peak fuel cladding temperatures for best-estimate calculation taking into account the uncertainty analysis are slightly lower in comparison with those obtained by partially-conservative calculation. That enables to draw a conclusion that in most cases both approaches, best-estimate and partially-conservative, can be applied. 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, the partially-conservative approach takes considerably less computational time. However, when the results obtained by this methodology do not meet the acceptance criteria (as in this particular case), a complete analysis by employing the best-estimate approach is necessary.

Received
11 February 2003

ABBREVIATIONS

BAZ	Fast Acting Scram System
CCFL	Counter Courant Flow Limitation
CHF	Critical Heat Flux
CPS	Control and Protection System
DS	Drum Separator
ECCS	Emergency Core Cooling System
FC	Fuel Channel
GDH	Group Distribution Header
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
NPP	Nuclear Power Plant
SAR	Safety Analysis Report
SUSA	System for Uncertainty and Sensitivity Analysis
PWR	Pressurised Water Reactor
RBMK	Russian abbreviation for Large Power Boiling Reactor

SUBSCRIPTS

e – electrical
th – thermal

References

1. Almenas K., Kaliatka A., Uspuras E. Ignalina RBMK-1500. A Source Book. Extended and updated version. Lithuanian Energy Institute, Kaunas, 1998.
2. The RELAP5 Code Development Team: RELAP5/MOD3 Code Manual. Volume 1. Code Structure, System Models, and Solution Methods. NUREG/CR-5535. INEL-95/0174. 1995.
3. Kloos M., Hofer E. SUSA Version 3.2. User's Guide and Tutorial. 1999.
4. Glaeser H. G. Uncertainty evaluation of thermal-hydraulic code results // Proceedings of International Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000). Washington, DC, USA, November 2000.
5. Glaeser H. G. GRS method for uncertainty and sensitivity evaluation of codes results: description, characteristics and applications // IAEA Regional Training Course on the Use of System Analysis Codes for Accident Analysis. Zagreb, Croatia, 2000.
6. D'Auria F., Leonardi M., Pochard R. Methodology for the evaluation of thermalhydraulic codes accuracy // Proceedings, Int. Conference "New Trends in Nuclear System thermal hydraulics". 1994. P. 467–477.
7. D'Auria F., Giannotti W. Development of a code with the capability of internal assessment of uncertainty // Nuclear Technology. 2000. Vol. 131. P. 159–196.
8. D'Auria F., Mavko B., Prošek A. Fast Fourier transform based method for quantitative assessment of code predictions of experimental data // Proceedings, Int. Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000). 2000.
9. D'Auria F. D., Galassi G. M., Leonardi M., Galetti M. R. Application of fast Fourier transform method to evaluate the accuracy of SBLOCA data base // Proceedings of NURETH-8. 1997. P. 358–366.
10. Boyack B. E. et al. Quantifying reactor safety margins. Part 1: an overview of the code scaling, applicability, and uncertainty evaluation methodology // Nuclear Engineering and Design. 1990. Vol. 119. P. 1–15.
11. Wilson G. E. et al. Quantifying reactor safety margins. Part 2: characterization of important contributors to uncertainty // Nuclear Engineering and Design. 1990. Vol. 119. P. 17–31.
12. De Crécy A. CIRCÉ: a tool for calculating the uncertainties of the constitutive relationships of CATHARE 2 // Proceedings, NURETH-8. 1997. P. 349–357.
13. Chojnacki E., Ounsy A. Application of the IPSN method for uncertainty and sensitivity analysis to the CATHARE2 code for a SBLOCA transient (LSTF SB-CL-18) // Proceedings of NURETH-8. 1997. P. 340–348.
14. Glaeser H. G., Wickett T., Chojnacki E., D'Auria F., Perez C. L. OECD/CSNI uncertainty methods study for "Best estimate" analysis // Proceedings, Int. Meeting on "Best-Estimate" Methods in Nuclear Installation Safety Analysis (BE-2000). 2000.

Rolandas Urbonas, Algirdas Kaliatka, Eugenijus Ušpuras

ŠIUOLAIKINIŲ ANALIZĖS METODŲ PALYGINIMAS, REMIANTIS POSTULUOJAMO GPK BLOKADOS ĮVYKIU IGNALINOS AE

S a n t r a u k a

Straipsnyje pristatyta Ignalinos AE postuluojamo šilumnešio srauto blokados grupiniame paskirstymo kolektoriuje analizė. Čia pateikti skaičiavimai, pritaikius dviejų tipų analizės metodus – geriausio įverčio bei pusiau konservatyvų. Geriausio įverčio metodas yra pagrįstas geriausio įverčio programų paketų naudojimu su realiomis pradinėmis ir kraštinėmis sąlygomis bei neapibrėžtumo analize. Tuo tarpu pusiau konservatyvus metodas – geriausio įverčio programų paketų naudojimu su konservatyviomis pradinėmis ir kraštinėmis sąlygomis bei konservatyviomis prielaidomis. Abiem metodais gautų rezultatų palyginimas rodo, kad pikinės kuro apvalkalo temperatūros, gautos geriausio įverčio metodu, yra truputį žemesnės, nei gautos pusiau konservatyviu metodu. Skaičiavimo sąnaudos pusiau konservatyviam metodui yra gerokai mažesnės. Tačiau kai šia metodika gauti rezultatai netenkina priimtinumo kriterijų (kaip parodyta šiame straipsnyje), būtina atlikti išsamią geriausio įverčio metodu pagrįstą analizę.

Raktažodžiai: grupinis paskirstymo kolektorius, termohidraulika, neapibrėžtumo ir jautrumo analizė

Роландас Урбонас, Альгирдас Калятка, Еугениюс Ушпурас

СРАВНЕНИЕ СОВРЕМЕННЫХ ПОДХОДОВ АНАЛИЗА НА ПРИМЕРЕ ПОСТУЛИРУЕМОЙ БЛОКИРОВКИ РГК ИГНАЛИНСКОЙ АЭС

Р е з ю м е

В статье представлен анализ постулируемого события – блокады теплоносителя в раздаточном групповом коллекторе на Игналинской АЭС. Представлены два типа анализа – наилучшей оценки и полуконсервативный. Методика анализа наилучшей оценки основана на использовании кодов наилучшей оценки с реалистическими начальными и граничными условиями и анализом неопределенности. В то время полуконсервативная методика основана на тех же кодах наилучшей оценки, но консервативных начальных и граничащих условиях, а также консервативных предпосылок. Сравнение результатов обоих методов показывает, что пиковые температуры оболочки топлива, полученные при расчетах с методикой наилучшей оценки и анализом неопределенности, несколько ниже, чем при расчетах с полуконсервативной методикой. Полуконсервативная методика не нуждается в столь больших вычислительных ресурсах. Но когда результаты, полученные с помощью этой методики, не удовлетворяют критерии приемлемости (как это было показано в статье), необходимо провести полный анализ по методике наилучшей оценки.

Ключевые слова: раздаточный групповой коллектор, теплогидравлика, анализ неопределенностей и чувствительности