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Uncertainty analysis of one Main Circulation Pump trip event at the Ignalina NPP

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One Main Circulation Pump trip event is an anticipated transient with expected frequency of approximately one event per year. There were a few events when one MCP was inadvertently tripped. The throughput of the rest running pumps in the affected Main Circulation Circuit loop increased, however, the total coolant flow through the affected loop decreased. The main question arises whether this coolant flow rate is sufficient for adequate core cooling.

This paper presents an investigation of one MCP trip event at the Ignalina NPP. According to international practice, the transient analysis should consist of deterministic analysis by employing best-estimate codes and uncertainty analysis. For that purpose, the plant's RELAP5 model and the GRS (Germany) System for Uncertainty and Sensitivity Analysis package (SUSA) were employed. Uncertainty analysis of flow energy loss in different parts of the Main Circulation Circuit, initial conditions and code-selected models was performed. Such analysis allows to estimate the influence of separate parameters on calculation results and to find the modelling parameters that have the largest impact on the event studied. On the basis of this analysis, recommendations for the further improvement of the model have been developed.

Key words: RBMK-1500, Main Circulation Pump, uncertainty analysis

1. INTRODUCTION

One MCP trip event can be assigned to anticipated transients with an expected frequency of approximately one event per year. This paper presents a benchmark analysis of one MCP trip event in the Ignalina NPP. A similar analysis has been presented in the preceding works [1, 2]. In contradistinction to the preceding works, this paper presents an uncertainty analysis of these events. For investigation of one MCP trip event, the RELAP5 model of the Ignalina NPP was used.

The Ignalina Nuclear Power Plant is a twin-unit with two RBMK-1500, graphite moderated, boiling water, multichannel reactors. The MCC consists of two identical halves – the left and the right loops. The schematic representation of one MCC loop is shown in Fig. 1. For the cooling water forced circulation through the RBMK-1500 reactor at the Ignalina NPP eight Main Circulation Pumps are employed. The MCPs (5) are joined in groups of four pumps each (three for normal operation and one on standby). The MCPs feed the common pressure header (8) on each side of the reactor. Each pressure header provides the coolant to 20 Group Distribution Headers (9), each of which in turn feeds 38 to 43 fuel channels (11). The coolant flow rate through individual fuel channels is regulated by Isolating and Control Valves mounted in the lower water communication lines (10). The coolant passing through the fuel channels is boiled and part of the water is evaporated. Steam–water mixture through steam–water communication lines (12) flows to the Drum Separators. Steam separated in the DS through steam lines (13) is supplied to turbines. In case of one MCP trip the throughput of two running pumps in the affected MCC loop increased, however, the total coolant flow through the affected loop decreased. The reactor power was decreased down to 60% from maximal in response to one MCP trip signal. A detailed description of RELAP5 nodalization scheme is presented in [1].

For validation of the RELAP5 model of the Ignalina NPP, a benchmark analysis of the forced circulation phenomenon was performed. Data on operational occurrence measured at the Ignalina NPP were compared to RELAP5 calculations.

Five methods for calculating the uncertainty in the predictions of advanced best estimate thermal-hydraulic codes were compared in [3]. The Pisa

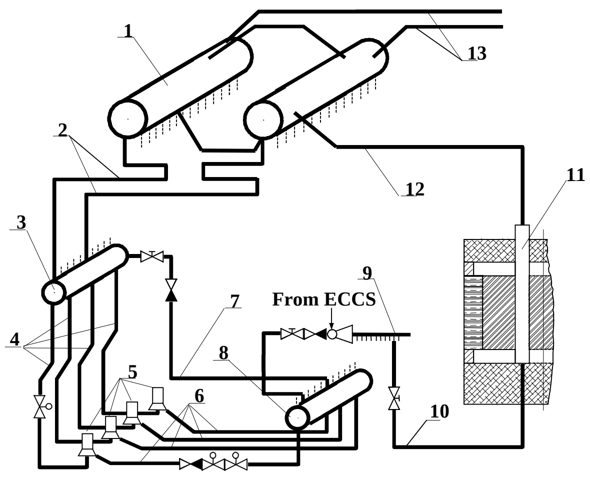

Fig. 1. Schematic representation of one loop of the RBMK-1500 Main Circulation Circuit: *1* – drum separator, *2* – downcomers, *3* – MCP suction header, *4* – MCP suction piping, *5* – MCPs, *6* – MCP discharge piping, *7* – bypass line, *8* – MCP pressure header, *9* – GDHs, *10* – lower water communication line, *11* – fuel channel, *12* – steam-water communication line, *13* – steam lines

method is based on extrapolation from integral experiments. The GRS, IPSN and ENUSA methods use subjective probability distributions. The AEAT method performs bounding analysis.

The GRS methodology and the developed package SUSA 3.2 [4] were selected for uncertainty calculation in this paper. The GRS method considers the effect of uncertainty of input parameters, application specific input data and solution algorithms on the results of calculations. The method is based on statistical tools and provides information in a form useful to decision makers. In the guidelines on choice of uncertainty analysis methods presented in [3] the GRS method is recommended for improving the knowledge of the predicted quantity most effectively and to reach the understanding of the interactions between the important processes.

The agreement of the calculation results obtained using the RELAP5 code taking into account uncertainties with the real plant data was evaluated using the Adequacy standard presented in the Guideline for performing code validation and issued by the DOE International Nuclear Safety Center [5]. The agreement is judged to be excellent when the code exhibits no deficiencies in modelling a given behaviour; major and minor phenomena and trends are predicted correctly; calculation results are judged to agree closely with the real plant data. The agreement is judged to be reasonable when the code exhibits minor deficiencies, although it provides an acceptable prediction; all major trends and phenomena are correctly predicted, but differences between the calculation and measured data are greater than those deemed acceptable for excellent agreement. According to the standard, both excellent and reasonable agreement of the calculation results and the real plant data are considered as acceptable.

2. MAIN CIRCULATION PUMP TRIP EVENT MODELLING

One MCP trip event analysis was performed using the best estimate system code RELAP5 [6]. According to the international practice, if the best estimate code is used for the analysis, the code and model uncertainties should be evaluated. Such analysis allows to find the modelling parameters that have the greatest impact on the events studied.

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

• Initial and boundary conditions (coolant pressure, flow rate, feed water temperature, amount of steam for the in-house needs, reactor power, flow energy loss in different MCC components). These values may be influenced by measurement errors.

• RELAP5 code models, assumptions and correlations.

In Table are presented the values of the parameters selected for the uncertainty analysis. Their initial values may have the greatest impact on the simulation results, as indicated by the knowledge of earlier performed benchmarking calculations.

From the Ignalina NPP documentation the deviation values are known for some of these parameters. For example, the measurement error of the coolant flow rate through pump measurement devices is equal to 2%, the reactor thermal power is determined with a 3% error. These values were used for uncertainty and sensitivity analysis as possible deviations for selected parameters. The mean deviation is determined according to the formula:

$$s = (a - i)/4,$$

where: s – mean deviation, a – maximal value of the parameter, i – minimal value of the parameter.

On the basis of RELAP5 description, recommendations for users and former benchmark analyses of RELAP5 code modelling parameters used in the basic case calculation were determined. To cover the possible code discrepancies, user effects, etc., for uncertainty analysis a wider range of parameter variation was used. 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 modelling elements. The areas with single-phase condi-

Table. **Parameters, which 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	Explanation
		Min. (i)	Max. (a)				
	Initial conditions						
1	Pressure in DS, Pa	$6.4315 \cdot 10^6$	$7.1085 \cdot 10^6$	$6.77 \cdot 10^6$	$1.6925 \cdot 10^5$ [5%]	Normal	Measurement error
2	Coolant flow rate through MCP, m^3/h	7546* 7703**	7854* 8017**	7700* 7860**	77.0 [2%]* 78.5 [2%]**	Normal	Measurement error
3	Feed water temperature, K	458.52	467.78	463.15	2.32 [1%]	Normal	Measurement error
4	Steam flow rate for the in-house needs, kg/s	82.418	85.782	84.1	0.841 [2%]	Normal	Measurement error
5	Reactor thermal power, W	$3.298 \cdot 10^9$	$3.502 \cdot 10^9$	$3.400 \cdot 10^9$	$5.1 \cdot 10^7$ [3%]	Normal	Measurement error
	Modelling parameters in the core						
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	Non homogeneous media	0 (non-homogeneous model)	1 (homogeneous model)	0 (non-homogeneous model)	–	Non parametric	Model assumption
	Flow energy loss						
13	Coefficient, which define ICV position	0.9	1.1	1.0	0.05 [10%]	Normal	Model assumption

* – coolant flow rate through each MCP of affected MCC loop.
** – coolant flow rate through each MCP of intact MCC loop.

tions are excluded due to the fact that these parameters have an insignificant impact in comparison to the two-phase area. In the basic case of calculations, some of the code models were disabled, as they did not have any impact on the results. However, in the uncertainty analysis none of the potential contributors to the uncertainty of the results can be excluded.

The coolant flow rate through individual channels depends on the resistance of pipelines between GDH and DS. Since pipelines connecting fuel channels with GDH and DS for all channels are identical, the coolant flow rate through individual channels will differ due to the different thermal power and different ICV positions. Therefore, it is obvious that variation of the ICV position will influence changes in the coolant flow rate through the channels.

3. RESULTS OF BEST-ESTIMATE ANALYSIS WITH UNCERTAINTY EVALUATION

On May 14, 1996 one MCP at the Ignalina Unit 2 was inadvertently tripped [7]. Before this event the reactor operated at the power level of 3400 MW_{th}. One turbine generator was operated in a pressure maintenance mode and the other in the power control mode. Six MCPs were in operation, providing a coolant flow of 7700 m^3/h to one and 7866 m^3/h to the other MCC loop through each pump.

The initiating event was the switch off of the two preferred electrical buses. It led to a trip of one of the six MCPs. As the flow through the pump dropped to zero (approximately 5 seconds after the beginning of the accident), the check valve downstream of this MCP closed, preventing a reverse flow through the tripped pump. The emergency protection signal AZ-4 was generated due to a loss of power to one MCP. The Control Protection System rods started moving. The reactor power was reduced to 60% from the design power within approximately 16 seconds after the switching off the MCP. The turbine generator (which before the accident operated in the power control mode) switched from power control mode to DS pressure maintenance mode.

The pressures calculated using the RELAP5 model *versus* the real measured plant data are presented in Fig. 2. As is seen from the figures, the divergence of the initial values of pressure can be explained by measurement errors: the initial measured pressure in DS and in Pressure Header is higher than that calculated by the RELAP5 model (basic case in Figure), and the initial measured pressure in the Suction Header is lower than the calculated one. To cover these two extremes, uncertainty analysis is performed using a two-sided tolerance limit (with 0.95 probability and 0.95 confidence). According to Wilk's formula [8], it is needed to perform at least 93 code runs. In this case 100 runs were performed. In Fig. 2 through Fig. 5, Maximum and Minimum curves bound values of all performed 100 calculations: the Maximum curve represents the maximum values and the Minimum curve – the minimum values of all 100 runs.

A comparison between the calculated flow rates obtained by the RELAP5 model and the real measured data are presented in Figs. 3, 4 and 5. According to the data, measured at the Ignalina NPP the initial coolant flow rates through MCPs of different MCC loops are different. Therefore two different initial

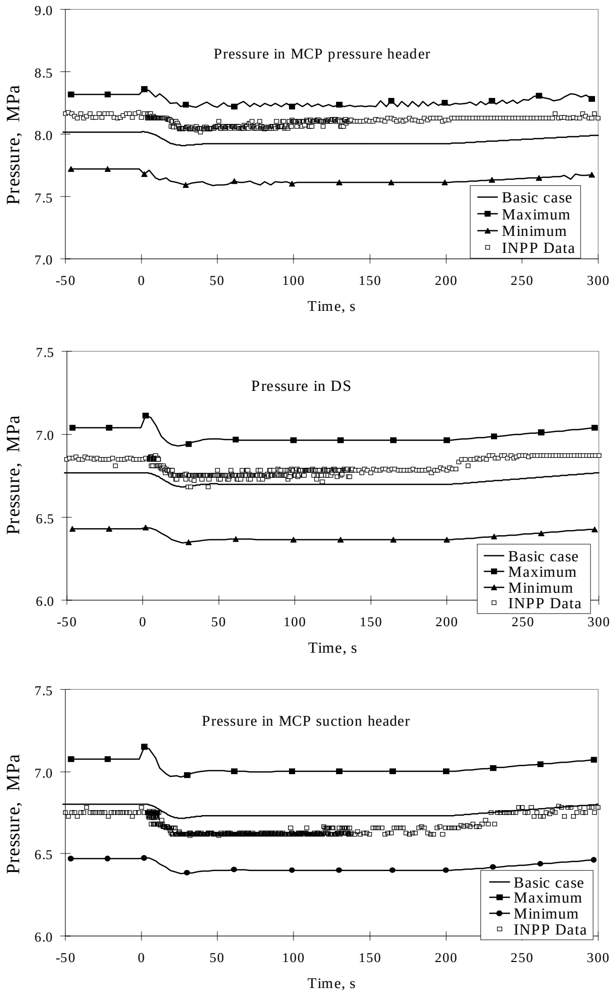

Fig. 2. Pressure in the Main Circulation Circuit

coolant flow rate values were used in the calculations (see Table). After an MCP trip the coolant flow rate through this pump is dropped to zero approximately within 5 seconds (see Fig. 4). The coolant flow rate through MCPs of an intact MCC loop after one MCP trip increases by ~400 m^3/h (see Fig. 3). This increase is due to a decrease of the reactor core resistance to the coolant flow after decreasing the reactor power. The throughput of each of the two running pumps increased by ~1500 m^3/h (see Fig. 5). However, the total coolant flow through the affected loop decreased from 23500 m^3/h to 19000 m^3/h. At the Ignalina NPP, measurement of the flow rate through MCPs is based on measuring pressure losses in the throttling devices. In Figs. 3–5 the observed spread of measured coolant flow rates through MCPs can be explained by imperfection of the measuring devices and information processing system.

In the case of one MCP trip the main question is whether core cooling by forced circulation through running MCPs is reliable. Therefore, from computational results in the case of one MCP trip, for analysis

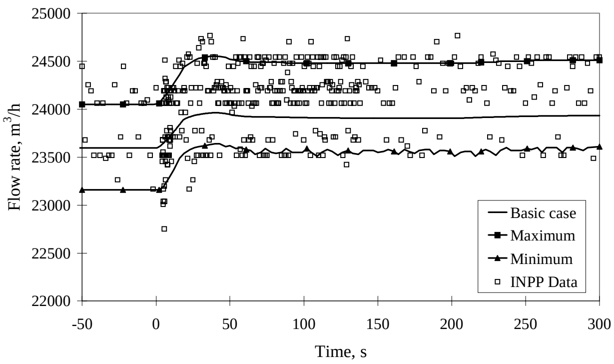

Fig. 3. Coolant flow rate through the Main Circulation Pumps of the intact Main Circulation Circuit loop

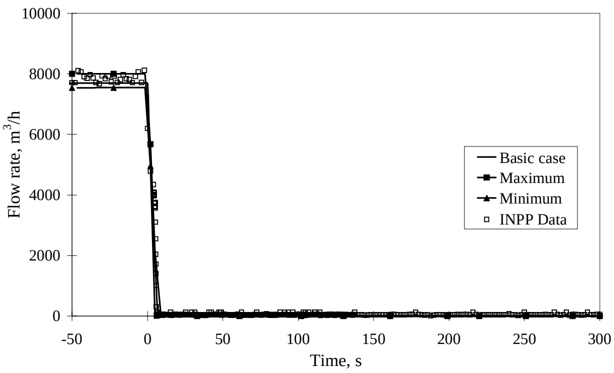

Fig. 4. Coolant flow rate through the tripped Main Circulation Pump

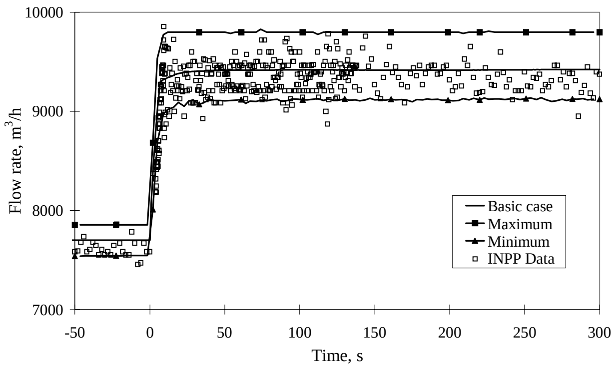

Fig. 5. Coolant flow rate through the Main Circulation Pump of the affected Main Circulation Circuit loop

the coolant flow rate through one running MCP of an affected loop was selected. In Fig. 5, the maximum and minimum values of all performed calculations, basic case calculation results and Ignalina NPP data are presented. As is shown in Figs. 2–5, the differences between the calculation results and the real data of the plant are greater than those deemed necessary for an excellent agreement. However, all the presented calculated parameters are in reasonable agreement with the real plant data, because most of the main measured parameter values are within the calculated uncertainty range (Figs. 2–5).

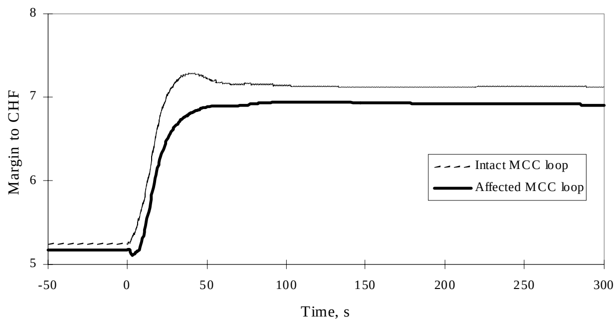

Fig. 6. Margin to the Critical Heat Flux

As is seen in Fig. 6, the margin to CHF in an intact MCC loop increases after a decrease of the reactor power. The margin to CHF in the affected MCC loop also increases, however, this increase is a somewhat less than in an intact loop. It is due to coolant flow rate decrease in the affected MCC loop. A decrease of the coolant flow rate and reactor power affects CHF in an opposite manner. However, the influence of power decrease is stronger than the influence of coolant flow rate decrease. The changes in the margin to CHF demonstrate changes in the reactor cooling regime changes after one MCP trip. Therefore, uncertainty analysis was performed for two states of reactor: steady state conditions and a trip of one MCP.

In the steady state conditions, the strongest impact on the calculation results has the initial flow rate through the MCPs (Par. 2, see Figs. 7 and 8). The transient analysis shows that the strongest impact on the calculation results has the Isolating and Control Valve position (Par. 13, see Figs. 7 and 8). Flow energy loss in ICV has a great impact on the coolant flow rate through the fuel channels. That is the reason for ICV position influence on the uncertainty of results. The other significant parameter is the initial flow rate through the MCPs (Par. 2).

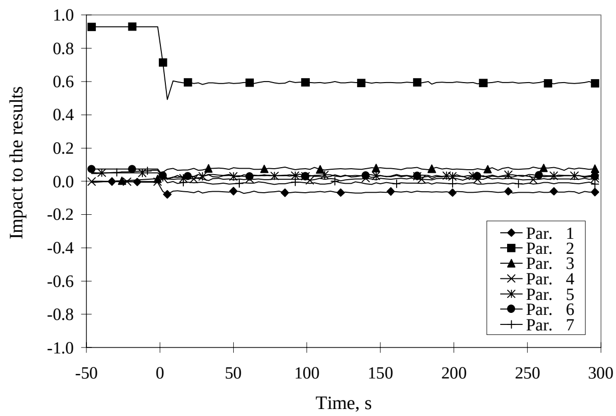

Fig. 7. Impact of the parameters Nos. 1–7 on the coolant flow rate through one running MCP of affected MCC loop in case of one MCP trip

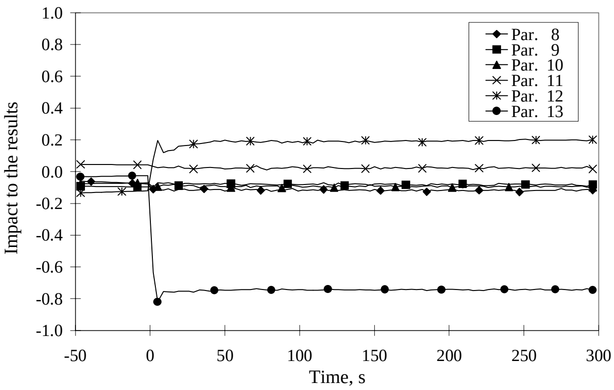

Fig. 8 Impact of the parameters Nos. 8–13 on the coolant flow rate through one running MCP of the affected MCC loop in case of one MCP trip

Analysis shows that after one MCP trip the coolant flow rate through the affected MCC loop is within the interval of 18000–19600 m^3/h, taking into account uncertainties of initial conditions and code assumptions. An increased margin to the critical heat flux (Fig. 6) in the worst case (lowest flow rate) shows that the reactor core is reliably cooled by forced circulation in case of one MCP trip.

4. CONCLUSIONS

Uncertainty analysis of flow energy loss in different parts of the Main Circulation Circuit, initial unit conditions and code-selected models was performed for one Main Circulation Pump trip event. Because the strongest impact on calculation results has the flow energy loss in the Isolating and Control Valve, it is recommended in cases of forced circulation to model the reactor core in a more detailed way, *i.e.* the core must be represented by possibly more channel groups. The performed uncertainty analysis has demonstrated that the reactor core is reliably cooled in the case of one Main Circulation Pump trip event, even taking into account the possible sources of uncertainties (measurement errors of initial plant conditions, code selection models, users errors).

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Abbreviations

AZ-4	emergency protection
CHF	critical heat flux
DS	drum separator
GDH	group distribution header
ICV	isolating and control valve
MCC	main circulation circuit
MCP	main circulation pump
NPP	nuclear power plant
RBMK	Russian acronym for “channeled large power reactor”

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VIENO PAGRINDINIO CIRKULIACIJOS SIURBLIO SUSTOJIMO IGNALINOS AE NEAPIBRËÞTUMO ANALIZË

S a n t r a u k a

Vieno pagrindinio cirkuliacijos siurblio (PCS) sustojimas yra tikëtinas pereinamasis ávykis, kurio tikëtinas daþnis – maþdaug vienas ávykis per metus. Ignalinos AE buvo keletas vieno PCS savaiminio sustojimo atvejø. Tokio ávykio metu ðilumneðio srautas per kitus veikianèius avarinës pagrindinio cirkuliacijos kontûro pusës siurblius padidëja, taèiau suminis ðilumneðio debitas per avarinæ pusæ sumaþëja. Iðkyla klausimas, ar tokio ðilumneðio srauto pakanka patikimai auðinti aktyviàjà zonà.

Straipsnyje pateikiama vieno PCS sustojimo Ignalinos AE atvejo analizë. Pagal tarptautinæ praktikà pereinamøjø procesø analizë turi susidëti ið deterministinës analizës, naudojant geriausio áverèio programinius paketus, ir neapibrëþtumo analizës. Ðiam tikslui panaudoti RELAP5 programinio paketo pagrindu sukurtas elektrinës modelis ir GRS (Vokietija) sukurta sistema SUSA, skirta neapibrëþtumo analizei. Atlikta srauto energijos praradimo ávairiose pagrindinio cirkuliacijos kontûro vietose, pradiniø sàlygø ir programiniame pakete naudojamø modeliø pasirinkimo jautrumo analizë. Tokia analizë suteikia galimybæ ávertinti tam tikrø parametrø átakà skaièiavimo rezultatams ir nustatyti modeliavimo parametrus, labiausiai veikianèius iðtyrinëtà ávyká. Ðios analizës pagrindu buvo pateiktos rekomendacijos, skirtos modeliui patobulinti.

Raktaþodþiai: RBMK-1500, pagrindinis cirkuliacijos siurblys, neapibrëþtumo analizë

Виргиниюс Вилейнишкис, Альгирдас Калятка, Еугениюс Ушпурас

АНАЛИЗ НЕОПРЕДЕЛЕННОСТИ СЛУЧАЯ ОСТАНОВА ОДНОГО ГЛАВНОГО ЦИРКУЛЯЦИОННОГО НАСОСА НА ИГНАЛИНСКОЙ АЭС

Р е з ю м е

Останов одного главного циркуляционного насоса (ГЦН) является ожидаемым переходным событием с ожидаемой частотой, приблизительно равной одному событию в год. На Игналинской АЭС было несколько случаев непреднамеренного останова одного ГЦН. В этом случае расход через оставшиеся работать насосы аварийной половины контура многократной принудительной циркуляции увеличивается, но суммарный расход через аварийную половину уменьшается. Возникает вопрос, достаточно ли такого расхода теплоносителя для надежного охлаждения активной зоны реактора.

В статье представлен анализ останова одного ГЦН. Согласно международной практике, анализ переходных процессов должен включать в себя детерминистический анализ с использованием программных пакетов наилучшей оценки и анализ неопределенности. Для этой цели использовались модель станции, созданная на основе программного пакета RELAP5, и созданная в GRS (Германия) система анализа неопределенности SUSA. Выполнен анализ чувствительности потери энергии теплоносителя в различных частях контура многократной принудительной циркуляции, начальных условий и выбора используемых в программном пакете моделей. Такой анализ позволяет оценить влияние отдельных параметров на результаты вычислений и определить параметры моделирования, имеющие наибольшее влияние на исследуемое событие. На основе данного анализа представлены рекомендации по улучшению модели.

Ключевые слова: главный циркуляционный насос, принудительная циркуляция, анализ неопределенности