Uncertainty and sensitivity analysis of Elektrogorsk-108 test facility RELAP5 model

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An evaluation of experimental data obtained at the Russian Elektrogorsk-108 (E-108) test facility is presented. The E-108 facility is a scaled model of the Russian RBMK design reactor. An attempt to validate the state-of-the-art thermal hydraulic code on the basis of E-108 test facility was made. Originally this code was developed and validated for BWRs and PWRs. Since this state-of-art thermal hydraulic code is widely used for simulation of RBMK reactors, further implementation and validation of the code is required.

The facility was modelled by employing the RELAP5 (INEEL, USA) thermal hydraulic system analysis best estimate code. The results show a dependence on the number of nodes used in the heated channels, the initial and boundary conditions, and code models. The obtained oscillatory behaviour is determined by the density wave and critical heat flux. It is shown that the codes are able to predict thermal hydraulic instability and sudden heat structure temperature excursion, when the critical heat flux is approached.

The uncertainty analysis of one of the experiments was performed by employing System for Uncertainty and Sensitivity Analysis (SUSA) developed by the GRS. It was one of the first attempts to use this statistics-based methodology in Lithuania.

Key words: E-108 test facility, thermal-hydraulic, uncertainty and sensitivity analysis

1. INTRODUCTION

An evaluation of experimental data obtained at the Russian Elektrogorsk-108 (E-108) test facility is presented. The E-108 facility is a scaled model of the Russian RBMK design reactor. The facility consists of six full-height parallel heated tubes, each of them simulating a single RBMK fuel channel. One of the features studied with the E-108 was that of identifying the system conditions at which flow instabilities between the parallel channels developed as a function of the variations in the thermal hydraulic conditions.

On the other hand, an attempt to validate the state-of-the-art thermal hydraulic codes on the basis of E-108 test facility was made. Originally these codes were developed and validated for BWRs and PWRs. Since the state-of-art thermal hydraulic codes are widely used for simulation of RBMK reactors, a further implementation and validation of the codes is required. The phenomena associated with channel type flow instabilities were found to be an important step in the frame of the overall effort of state-of-the-art validation and application for RBMK reactors.

Of particular importance for the validation of state-of-art thermal hydraulic codes are experimental investigations performed at facilities, which address the specific features of the RBMK. It is nearly impossible to conduct flow-instability experiments at a full scale which incorporates all details of the of the two-phase flow loop RBMK. Therefore the results of instability testing at properly scaled test loops must be analysed and used for setting the limitations at a full scale.

The facility was modelled by employing the RELAP5 (INEEL, USA) thermal hydraulic system analysis best estimate code. The results show a dependence on the number of nodes used in the heated channels, frictional and form losses employed. The obtained oscillatory behaviour is determined by the density wave and critical heat flux.

The uncertainty analysis of one of the experiments was performed by employing the German System for

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Uncertainty and Sensitivity Analysis (SUSA). It was one of the first attempts to use this statistic-based methodology in Lithuania. The obtained knowledge will be used in the licensing process of the Ignalina Nuclear Power Plant.

2. DESCRIPTION OF THE ELEKTROGORSK TEST FACILITY

The Elektrogorsk E-108 facility is a scaled model of the Russian RBMK design reactor. It was constructed in the town of Elektrogorsk, Russia, and operated by the Elektrogorsk Research and Engineering Centre of NPP Safety (EREC) to gain understanding of the thermal hydraulic behaviour of the full-scale RBMK reactor. The prime aspect of the facility is six full-height parallel heated channels, each of them simulating a single RBMK pressure tube. Test programs were conducted at the E-108 facility in 1982, 1984 and 1985 [1]. One of the features studied with the E-108 model was identification of the system conditions at which flow instabilities between the parallel channels develop. This was accomplished by sequentially reducing the forced flow rate of the coolant to the heated channels for a given inlet subcooling level. Tests were conducted for a range of system pressures (10 to 70 bar) and heated channel power input (0.06 to 0.77 MW/m^2) and inlet subcooling. Additionally, and of particular importance with respect to the RBMK pressure channels, the relative flow resistance of the piping sections was varied through the use of flow orifice.

The test loop has six full-height parallel-heated channels representing the pressure channels of the RBMK. They are connected to pumps, headers, heat exchangers and separators (see Fig. 1). The main components are listed in Table 1.

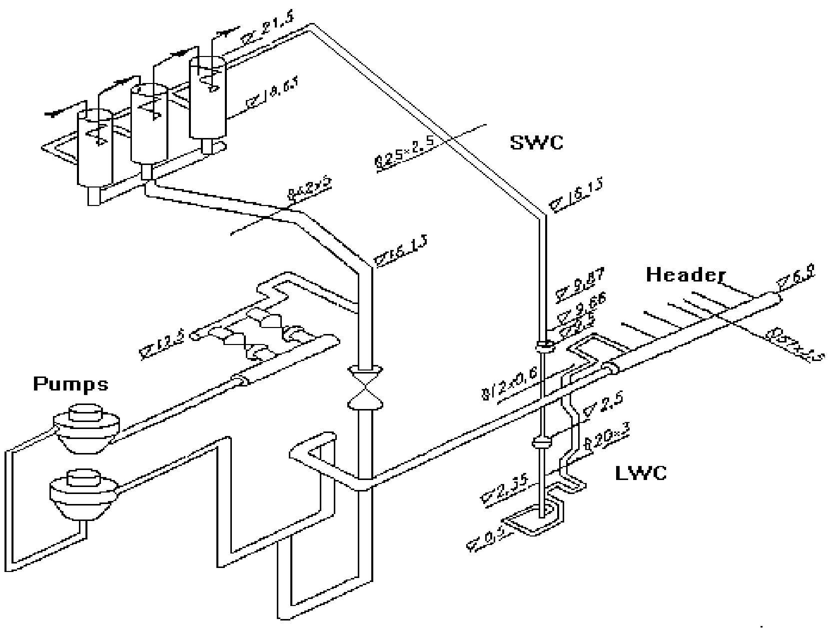

Fig. 1. E-108 facility configuration

2.1. RBMK-1500 and Elektrogorsk 108 facility scaling

According to Proshutinsky [2], the necessary conditions for the proper modelling of two-phase flow processes in parallel channels are to preserve the following thermal-hydraulic parameters:

- Reynolds ($Re = \rho \cdot v \cdot x/\mu = = v \cdot x/\nu$) and Froude ($Fr = w^2/g \cdot L$) numbers
- Dimensionless heat load N_f and dimensionless initial subcooling number N_h;
- Values of ξ_j, $(f \cdot L/d)_j$, K_j, L_j/L_2, α_j

Table 1. **Main components of the Elektrogorsk E-108 facility**

Component	Parameter
Inlet header	Inner diameter 50 mm
	Thickness 3.5 mm
	Length 0.6 m
Lower water communications piping	Inner diameter 10 to 14 mm
	Length 21.571 m
Heated channels	Inner diameter 10 or 10.8 mm
	Thickness 1 or 0.6 mm
	Length 7.3 m
Steam water communications piping	Inner diameter 20 mm
	Thickness 2.5 mm
	Length 30.387 m
Steam-water gravity separators	Inner diameter 144 mm
	Thickness 12 mm
	Height 3.5 m
Condensers/coolers	Inner diameter 12 mm
	Thickness 2 mm
Two centrifugal pumps	V_{nom}= 30 m^3/h
	ΔP_{nom}= 7.35 bar
Electric heaters	500 kW output

- Slip factor, $S = v_g/v_l$.

An important condition for the model is to reproduce axial energy distribution. It is very difficult to satisfy all of the above conditions in an experimental facility. Additionally, it is almost impossible to obtain geometrical simulation for all elements of the system, *e.g.*, identical friction factors in separate parts of the system, and to reproduce axial energy distribution.

The experiments in the Elektrogorsk E-108 facility have been conducted in three variations of resistances [1]:

- $\xi_{LWC} = 25$ and $\xi_{SWC} = 3.5$, which closely corresponds to the RBMK-1000 conditions
- $\xi_{LWC} = 25$ and $\xi_{SWC} = 14$, which closely corresponds to the RBMK-1500 conditions with closed check-valve
- $\xi_{LWC} = 75$ and $\xi_{SWC} = 14$, which closely corresponds to the RBMK-1500 conditions under nominal operation.

Table 2. **Scaling comparisons of the RBMK *versus* the Elektrogorsk E-108 facility**

Parameters	RBMK-1500 [3]	Elektrogorsk E-108 [1]
L_{LWC}, m	33.97	21.581
L_{HC}, m	7.00	7.00
L_{SWC}, m	33.93	30.387
L_{SWC}/L_{HC}	5.360	2.430
H_{SWC}/L_{HC}	1.940	1.310
$\xi_{SWC} \cdot K_{SWC}^{2.}$	3.380	10.760
$(K_{SWC}^{2} \cdot L/d)_{SWC}$	186.0	128.0
$(V_{LWC} + V_{HC} + V_{SWC}) \cdot 10^3$, m^3	257.529	9.542
d_{HC}, mm	8.57	10; 10.8
$A_{HC} \cdot 10^4$, m^2	22.730	0.785; 0.916
Pressure, bar	70	10 – 70
Flow rate (per channel), kg/s	1.24–1.28	0.15*–1.37
Heat flux, MW/m^2	0.3–1.25	0.06–0.77

• – natural circulation simulation

The Elektrogorsk E-108 facility corresponds reasonably well to the RBMK-1500 (see Table 2). This facility is aimed to model the parallel channel instability and the critical heat flux phenomena in the fuel channels, which are important phenomena in RBMK plants. The geometry (height of the heated section and the total height), thermal-hydraulic of the channels for some boundary conditions, as well as local resistances of the Elektrogorsk E-108 facility correspond exactly to the same parameters of the RBMK-1500. However, the length of the pipes and the parameter L_j/L_2 which corresponds to the friction loss coefficient in steam water communications is slightly different from those in the RBMK-1500.

A more detail description of the Elektrogorsk E-108 test facility is presented in [1] and in some other references.

2.2. RELAP5 model of E-108 facility

Model for the Elektrogorsk E-108 facility employs all available geometrical information between the lower and upper distribution headers. The main geometrical and thermal data are described in section 2.1 and [1]. The available data enabled to model, using the RELAP5 state-of-the-art code [4], the part of the Elektrogorsk E-108 facility loop between the inlet header and the steam-water gravity separator. The modelled components are the inlet header, lower water communications, heated channels and steam water communications. In Fig. 2 the nodalisation of the model is shown. The lower water communication section was subdivided into six parts. Each of them has a different number of control volumes. The first part, 9.056 m long, has four control volumes, the second has three control volumes, and the rest have two control volumes each. The lower water communication section is not heated and should not affect calculation accuracy very much. In calculations one heated channel was divided into 12 control volumes

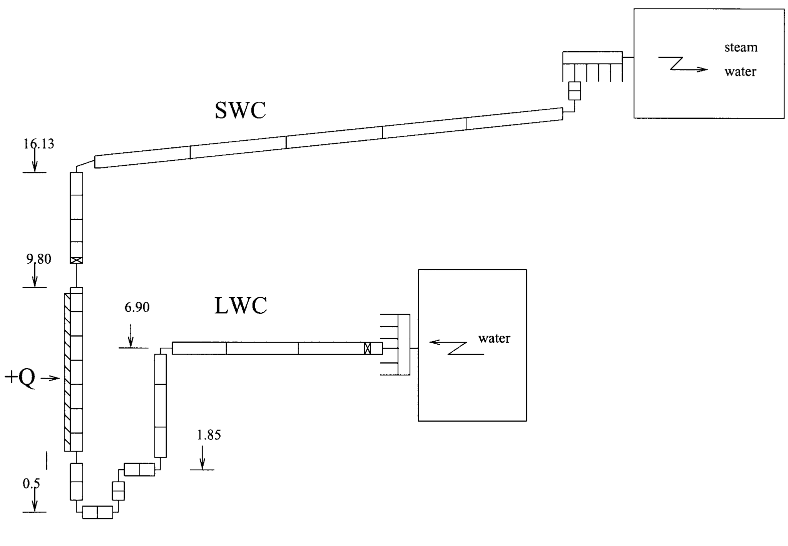

Fig. 2. E-108 test facility model

to match the measured heated tube temperature excursion. Steam water communications were divided into three parts with four, five and one control volumes, respectively.

3. INTRODUCTION TO THE GRS METHOD OF UNCERTAINTY EVALUATION

Up to now, conservative approaches were used for the nuclear power plant safety analysis. Now this approach is gradually replaced by the best estimation approach. In the case the best estimated methodology is used, the code and model uncertainty should be evaluated. Separate values of unknown accuracy should be presented for comparison with acceptance limits. Code predictions are uncertain due to a number of uncertainty sources such as code models, initial and boundary conditions, plant state, scaling and numerical solution algorithm.

The aim of the GRS uncertainty analysis [5, 6] is to identify and quantify all potentially important uncertainty parameters. The state of knowledge about all uncertain parameters should be described by ranges and subjective probability distribution.

In order to get information about the uncertainty of computer code results, a certain number of code runs must be performed. In each calculation, all uncertain parameters should be varied simultaneously.

The different steps of the uncertainty analysis are supported by the GRS software system SUSA (Uncertainty and Sensitivity Analyses) [5, 6].

3.1. Selection of input parameters that may cause calculation uncertainty

For the analysis, the forced circulation test was chosen, which represents the RBMK-1500 operation under closed check valve conditions. During this test the inlet coolant flow rate was decreased gradually at a nearly constant heater power. The simulated coolant flow rate decrease through all heated channels is shown in Fig. 3. The gradual step-shape coolant flow rate decrease leads to flow rate instabilities approximately

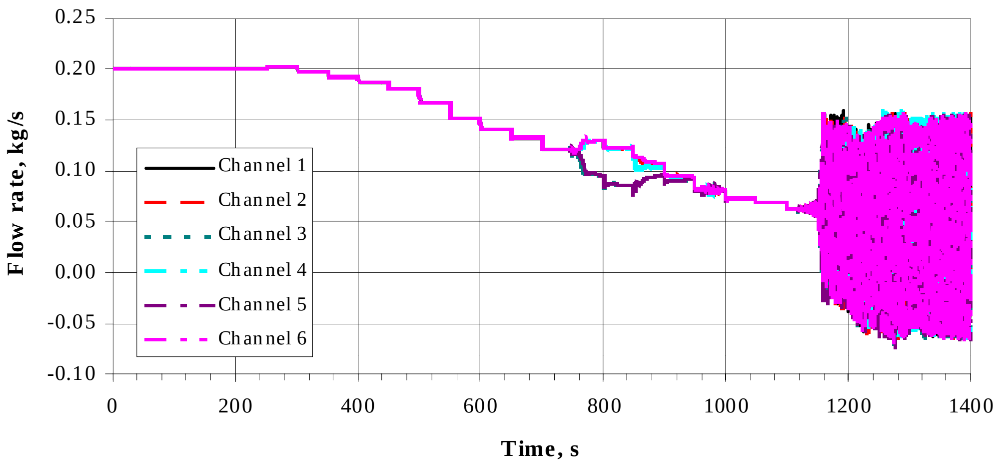

Fig. 3. Flow rate at the inlet of the channels

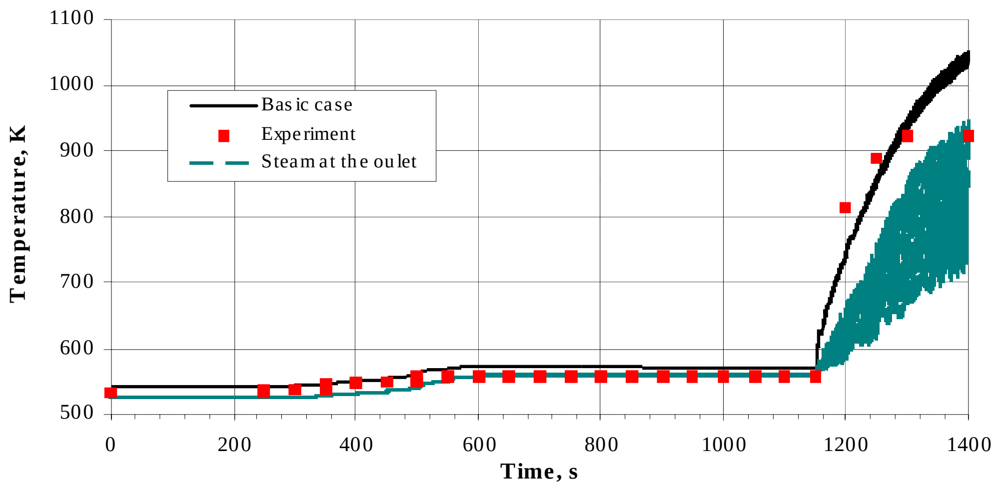

Fig. 4. Behaviour of heated tube wall surface peak temperature

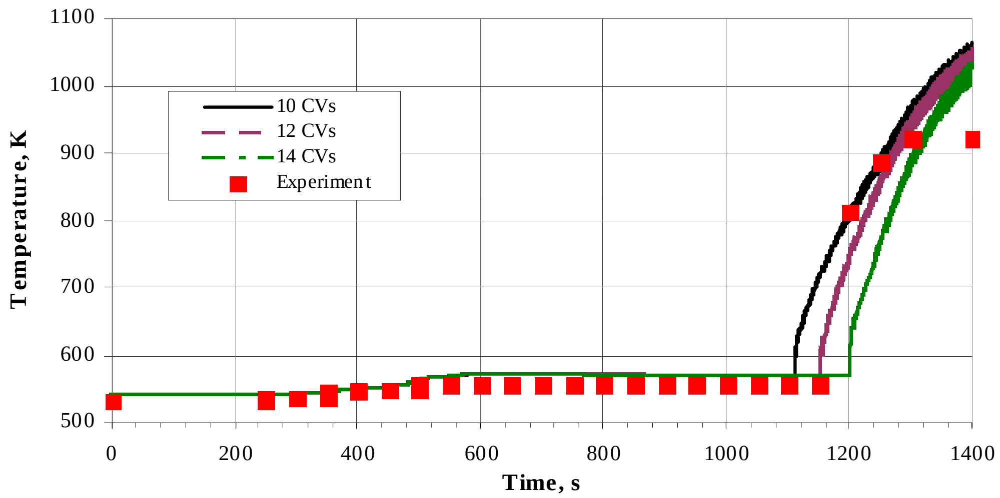

Fig. 5. Influence of the number of axial nodes on the tube wall surface peak temperature

1100 s after beginning of the transient. Approximately 1150 s after the beginning of the transient (coolant flow rate is about 0.06 kg/s), the heat flux generated by the electrical heater exceeds the critical heat flux. It leads to the heated tube wall surface temperature excursion (Fig. 4). Fig. 4 shows that the simulated heated tube wall temperature throughout the simulation (0–1150 s) is higher than the measured one, perhaps because of some measurement problems, as in the time span 600 to 1150 s the measured heated tube wall temperature corresponds well to the simulated steam temperature at the outlet of the heated channel.

Table 3. **Selection of input parameters that may affect the calculation results**

No	Parameter	Ranges		Reference	Distribution	Explanation
		Min.	Max.			
Initial conditions						
1	T_0 (rated temperature)	0.95	1.05	1.0	Normal	Measurement error
2	G_0 (rated flow rate)	0.936	1.064	1.0	Normal	Measurement error
3	Q_0 (rated power)	0.932	1.068	1.0	Normal	Measurement error
4	P_0 (rated pressure)	0.981	1.019	1.0	Normal	Measurement error
RELAP5 models						
5	Water packing	0	1	0	Non-parametric	Model assumption
6	Stratification	0	1	0	Non-parametric	Model assumption
7	Non-equilibrium	0	1	0	Non-parametric	Model assumption
8	PV term	0	1	0	Non-parametric	Model assumption
9	CCFL	0	1	0	Non-parametric	Model assumption
10	Non homogeneous	0	1	0	Non-parametric	Model assumption

Table 4. **Parameter values of the design runs calculated by SUSA**

	Index parameter									
No.	1	2	3	4	5	6	7	8	9	10
1	1.04	0.986	1.02	1.00	1	0	1	1	1	1
2	0.956	1.00	0.999	0.988	0	0	0	0	0	0
3	1.00	1.03	0.987	1.01	0	0	1	1	0	1
…	…	…	…	…	…	…	…	…	…	…
59	0.986	0.992	1.02	1.01	1	0	1	1	1	0
60	0.952	0.980	0.981	1.02	0	1	1	1	1	1

The parameters that may cause the calculation uncertainty can be divided into two main groups:

• Initial conditions (coolant pressure, temperature and flow rate or heater power. These values may be affected by measurement errors during the experiment)

• RELAP5 model assumptions (in the model different correlations for the calculation of friction loss, CHF and heat transfer may be used) [4].

The nodalisation matter was excluded from uncertainty analysis. The results of sensitivity analysis of model nodalisation are shown in Fig. 5. One can see that the axial nodalisation influences the time of CHF occurrence. From a number of possible heated section divisions, the nodalisation of 12 control volumes in a heated section was selected as a basic case to match the experimental data.

A list of important parameters has been prepared (see Table 3). The state of knowledge about all uncertain parameters is described by ranges and selected probability distributions. Using this input information, the computer code SUSA generates the table of selected parameter values for the design runs (see Table 4). Because the one-sided evaluations for probability 0.95 and confidence 0.95 were selected, the number of RELAP5 runs should be at least 59 [5, 6]. In the analysed case 60 RELAP5 runs were performed.

3.2. Uncertainty and Sensitivity Analysis

With the SUSA 3.2 package three types of analysis can be done [5]:

- Index-dependent uncertainty analysis,
- Index-dependent sensitivity analysis,
- Scalar sensitivity analysis.

From a number of RELAP5 output result parameters, only one important parameter, tube wall surface peak temperature, was selected. This result describes the best of our selected phenomena – the critical heat flux. All SUSA analyses were performed with respect to this tube wall surface peak temperature results. Time-dependent peak temperatures for 60 runs are shown in Fig. 6 which shows that the CHF occurrence time as well as the peak heated tube wall temperature varies within a considerable range.

In the uncertainty analysis there were set tolerance limits (probability 0.95 and confidence 0.95). Maximum, minimum and mean values were compared with the experiment data (Fig. 7). Experimental data after CHF occurrence are within the simulated maximum and minimum values. In the time span 0–1150 s, experimental data represent the steam temperature (see Figs. 4 and 7). In Fig. 8, a comparison of one-sided tolerance limit, basic case and measured data is presented.

In the sensitivity analysis, the parameters that most strongly influence the results (peak temperature) were identified. In the case of the E-108 model (Table 3) there are several parameters such as parameter No. 1 – coolant inlet temperature, parameter No. 2 – coolant flow rate, parameter No. 3 – heater power, and parameter No. 7 – selection of non-equilibrium option. Figures 9 and 10 show that the results vary with time. It might have both a positive (*i.e.* supports the temperature increase) and a negative (*i.e.* damps the temperature increase) effect on the results.

When one looks at the coolant flow rate (parameter 2) variation, one can indicate the trend of dependence (Fig. 11): the lower the coolant flow rate, the higher the heated tube wall temperature.

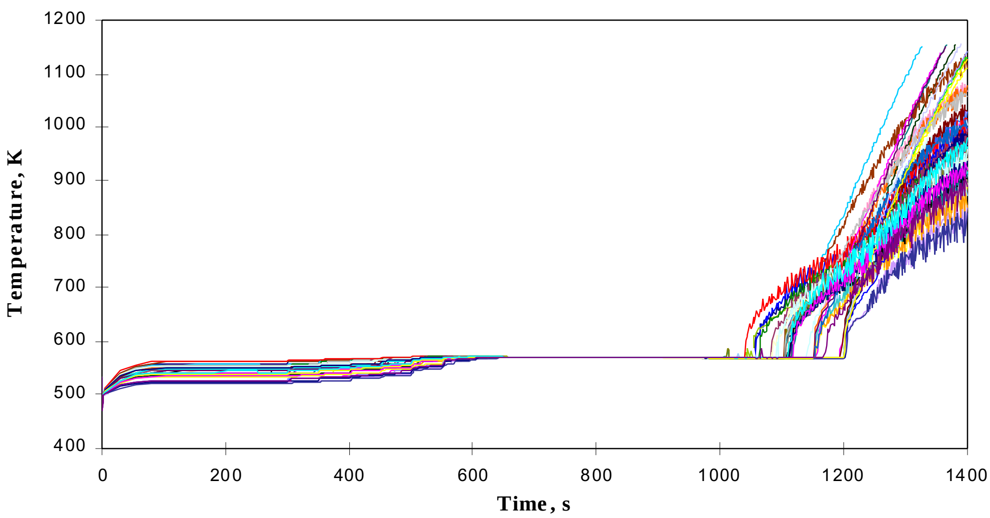

Fig. 6. Simulated and SUSA generated tube temperatures

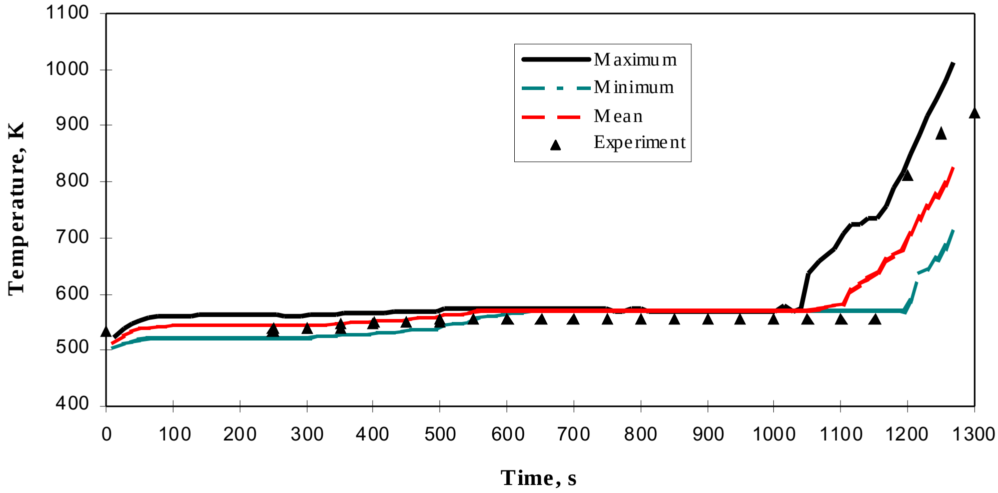

Fig. 7. Comparison of maximum, minimum, mean values of the performed runs (peak temperature of heated tube wall surface) with the temperature measured

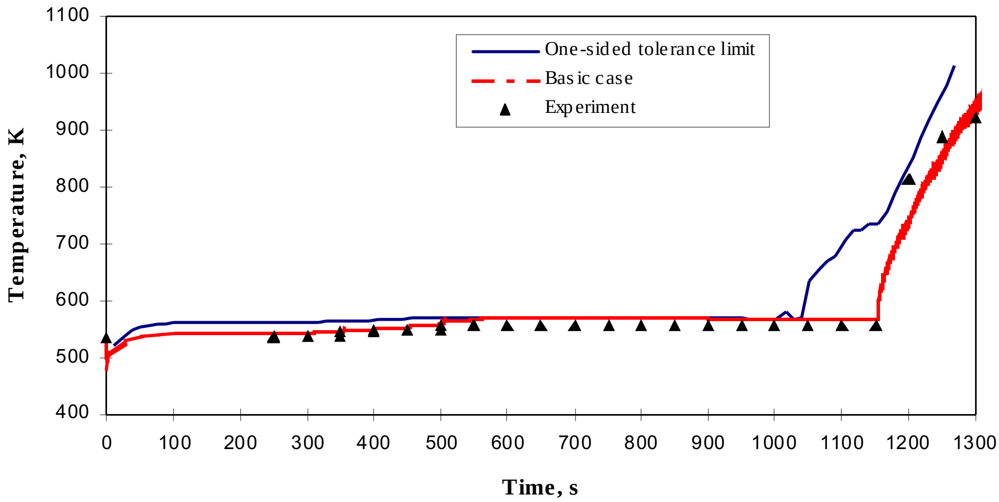

Fig. 8. Comparison of uncertainty boundaries of one-sided tolerance limit with the basic case and measured temperature

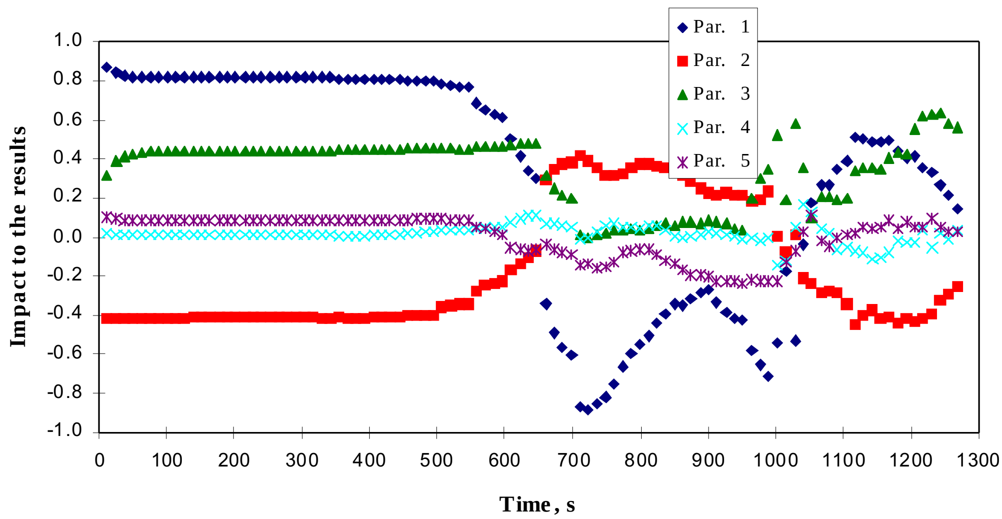

Fig. 9. Correlation of the first five parameters with selected results (peak temperature of heated tube wall surface)

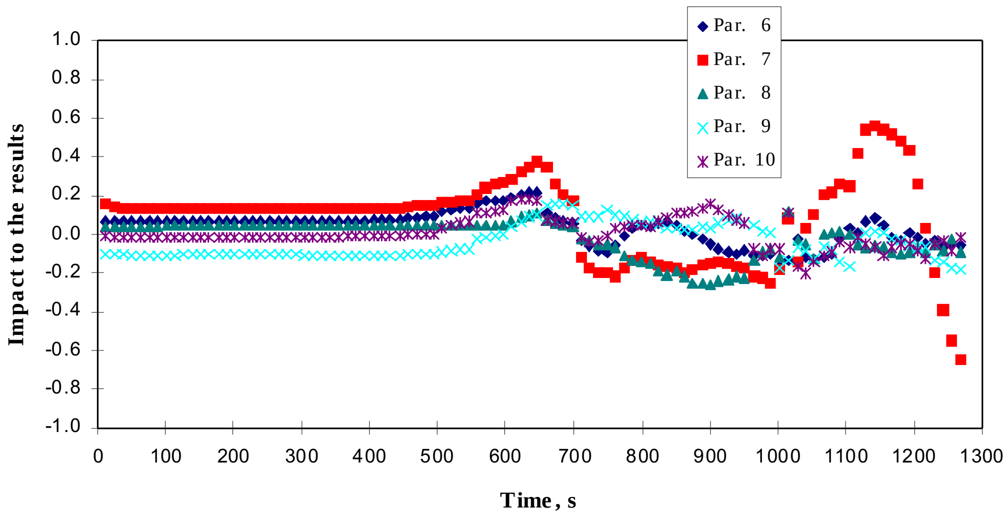

Fig. 10. Correlation of the second five parameters with selected results (peak temperature of heated tube wall surface)

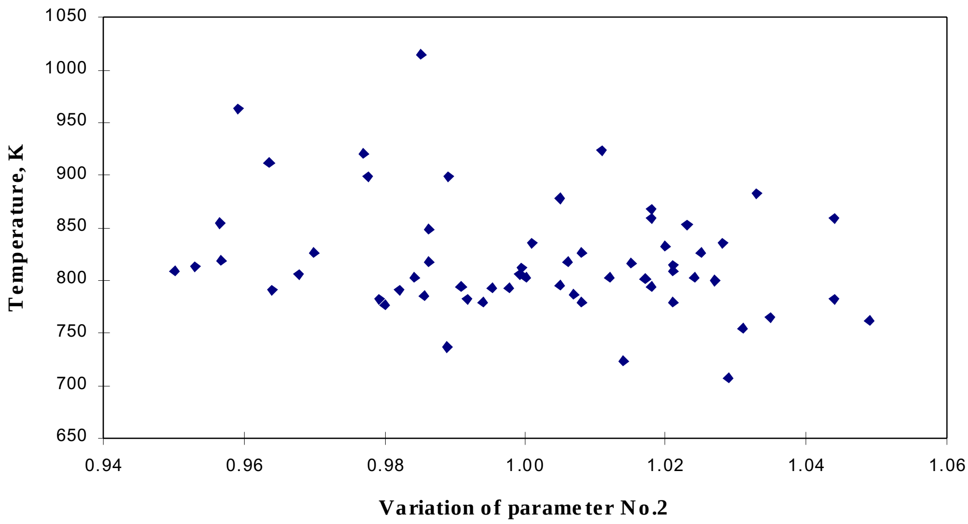

Fig. 11. Influence of coolant flow rate variation at the very end of the transient

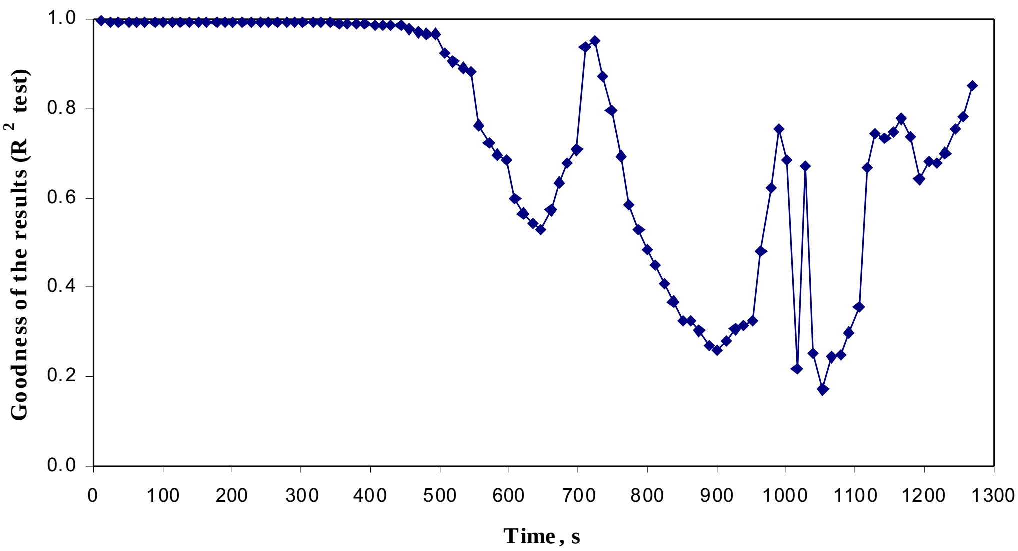

Fig. 12. Goodness of the results (R^2 test)

The obtained results were checked by the R^2 test. The results were accepted with a very high confidence (0.95) at the beginning of simulation (Fig. 12). However, R^2 dropped down to 0.2 for a short time and later recovered to the 0.6–0.8.

The uncertainty and sensitivity method applied by SUSA does not limit the number of uncertain parameters and that is not computationally costly.

The method provides sensitivity measurements of the influence of the identified input parameter uncertainties on the results. The measures permit an uncertainty importance ranking. This provides information where to improve the state of knowledge in order to reduce the output uncertainties.

4. CONCLUSIONS

An analysis of the Elektrogorsk E-108 test facility data is presented. Simulations with RELAP5 code showed that the code allows a good representation of the experimental data.

The sensitivity and uncertainty analysis is demonstrated. The analysis allowed to identify the parameters, that most strongly influence the results.

The obtained knowledge will be applied while performing best estimate calculation for the Ignalina NPP. That will be an important issue in the licensing process, in safety analysis for modification of diverse shutdown system and other analyses important to the safety. The uncertainty evaluation methodology developed by the GRS can be used not only for the RELAP5 calculations, but also for other thermal-hydraulic analyses as well as for the analyses beyond the thermal hydraulic field.

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Abbreviations

BWR	Boiling Water Reactor
PWR	Pressurised Water Reactor
RBMK	Russian abbreviation "Large Power Boiling Reactor"
CHF	Critical Heat Flux
CCFL	Counter Current Flow Limitation
HC	Heated Channel
LWC	Lower Water Comunications
SWC	Steam Water Comunications
ξ	hydraulic resistance coefficient, (–)
K_j	resistance coefficient, (–)
f	friction coefficient, (–)
α_j	angle, ^{o}
V_g	velocity of gas, m/s
V_f	velocity of fluid, m/s
P	rated pressure, (–)
h_{fg}	latent heat, J/kg
i	enthalpy
N_f	dimensionless heat load number, $N_f = Q/G \cdot h_{fg}$
N_h	dimensionless initial subcooling $N_h = \Delta i/h_{fg}$

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ELEKTROGORSKO E-108 EKSPERIMENTINIO STENDO DUOMENŲ NEAPIBRĖŽTUMO IR JAUTRUMO ANALIZĖ

S a n t r a u k a

Straipsnyje pateiktas Elektrogorsko eksperimentiniame stende E-108 atliktų bandymų įvertinimas. E-108 yra sumažintas RBMK tipo reaktoriaus modelis.

Eksperimentų, atliktų šiame stende, pagrindu buvo pabandyta validuoti geriausio įverčio programinį paketą RELAP5. Šis programinis paketas buvo sukurtas ir validuotas BWR ir PWR tipo reaktoriams. Pastaruoju metu šis termohidraulinis programinis paketas yra plačiai naudojamas RBMK tipo reaktorių skaičiavimams, todėl reikalinga tolimesnė jo plėtra ir validacija.

Bandymai, atlikti E-108 eksperimentiniame stende, buvo sumodeliuoti, naudojantis RELAP5 programiniu paketu. Eksperimentų metu išmatuotų ir apskaičiuotų rezultatų palyginimas parodė, kad kaitinamojo kanalo sudalinimas, pradinės ir kraštinės sąlygos bei kai kurių programinio paketo modelių panaudojimas turi didelę įtaką. Gauti tekėjimo nestabilumai buvo sąlygoti tankio bangos bei virimo krizės. Yra parodyta, kad programinis paketas gali gerai aprašyti termohidraulinius tekėjimo nestabilumus bei staigų šiluminių struktūrų temperatūros padidėjimą.

Papildomai vienam iš eksperimentų buvo atlikta neapibrėžtumo analizė, panaudojant GRS kompanijos sukurtą sistemą neapibrėžtumo ir jautrumo analizei (SUSA). Tai buvo vienas pirmųjų bandymų Lietuvoje taikyti šią statistine analize pagrįstą metodologiją.

Raktažodžiai: E-108 eksperimentinis stendas, termohidrodinamika, neapibrėžtumo ir jautrumo analizė

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АНАЛИЗ НЕОПРЕДЕЛЕННОСТЕЙ И ЧУВСТВИТЕЛЬНОСТИ ДАННЫХ ЭЛЕКТРОГОРСКОГО Е-108 ЭКСПЕРИМЕНТАЛЬНОГО СТЕНДА

Р е з ю м е

В статье представлена оценка опытных данных, полученных на Электрогорском Е-108 экспериментальном стенде. Экспериментальный стенд Е-108 – это уменьшенная модель реактора типа РБМК.

На основе экспериментальных данных, полученных на этом стенде, была сделана попытка провести валидацию системного кода наилучшей оценки RELAP5 (INEEL,США). Этот код был создан и развит для реакторов типов BWR и PWR. В настоящее время этот код все больше используется для анализа реакторов типа РБМК, поэтому необходимо проводить дальнейшее его улучшение и валидацию.

Опыты, проведенные на экспериментальном стенде Е-108, были смоделированы с помощью термогидравлического системного кода наилучшей оценки RELAP5. Сопоставление измеренных и рассчитанных данных показало, что результаты зависят от нодализации в нагреваемом канале, начальных и граничных условий, а также моделей кода. Полученное колебательное поведение потока было вызвано волной плотности и кризисом кипения. Показано, что код может хорошо описать явления термогидравлических колебаний и внезапного повышения температуры в тепловых структурах.

Дополнительно с помощью системы анализа неопределенностей и чувствительности (SUSA) был проведен анализ неопределенностей одного из экспериментов. Это одна из первых попыток в Литве использовать вышеупомянутую методологию, основанную на статистике.

Ключевые слова: экспериментальный стенд Е-108, термогидродинамика, анализ неопределенности и чувствительности